LIMERICK GENERATING STATION UNITS 1 & 2

50-352
50-353

ENVIRONMENTAL REPORT - OPERATING LICENSE STAGE

REVISION 12 PAGE CHANGES

The attached pages, tables, and figures are considered part of a controlled copy of the Limerick Generating Station EROL. This material should be incorporated into the EROL by following the instructions below.

<u>After the revised pages are inserted, place the page that follows these instructions in the front of Volume 1.</u>

REMOVE	INSERT
VOLUME 2	
Pages 2.4-7 thru -10	Pages 2.4-7 thru -10
Page 3-vii	Page 3-vii
Figure 3.1-3	Figure 3.1-3
Page 3.4-3 & -4	Page 3.4-3 & -4
Figure 3.4-5	Figure 3.4-5
Pages 3.9-1 & -2	Pages 3.9-1 & -2
----	Figure 3.9-8
VOLUME 4	
Pages 7-i, -ii, -iv, -v	Pages 7-i, -ii, -iv, -v
Pages 7.1-1 thru -8	Pages 7.1-1 thru -8
Pages 7.1-21 thru -27	Pages 7.1-21 thru -38
----	Tables 7.1-22 thru -26
Figure 7.1-1 & -2	Figures 7.1-1 thru -7
VOLUME 5	
Page E290.16-1 thru Figure E290.16-1	Page E290.16-1 thru Figure E290.16-1
Page E310.10-5	Pages E310.10-5 & -6
Pages E450.1-1 thru .4-2	Pages E450.1-1 thru .4-1

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carbonate, and like the Schuylkill contains high concentrations of major cations and anions. The major cations and anions are at their highest concentrations July through November (Table 2.4-13). The essential plant nutrients are present in high concentrations in Perkiomen Creek water and pose quality problems as discussed in Section 2.2. All transition series elements are found in low concentrations (Table 2.4-13).

2.4.7.1.3 Chemical Characteristics of the East Branch Perkiomen Creek

Water quality studies of the East Branch in relation to LGS were initiated in May 1974. While data were collect at four stations (Section 6.1), only two, the upper, E32300, and the lower, E2800, will be used in this discussion. Table 2.4-14 is a summary of water quality data from E32300 covering the period 1975 through 1978 and Table 2.4-15 is a summary of data from E28000 covering the same period. The water quality of the East Branch ranges from good at E32300 to highly degraded at E2800. This shift in quality is a result of allochthonous inputs from sources to mouth as described in Section 2.2. The ionic base of the uppwer East Branch is carbonate and shifts to sulfate in the lower reaches. The East Branch has high concentrations of major cations and anions in the middle and lower reaches (Table 2.4-15); especially July through November when flow becomes intermittent. Th lower reaches also have high concentrations of the ions considered essential plant nutrients and of certain transition series element (i.e. iron, manganese, zinc, copper, and chromium). The quality of the upper East Branch is not unlike that of the Delaware River at Point Pleasant while the quality of the lower East Branch is similar to that of the Schuylkill near LGS.

2.4.7.1.4 Chemical Characteristics of the Delaware River

Water quality studies of the Delaware River in relation to LGS were initiated in May 1974. A summary of the program is given in Section 6.1. The water quality of the Delaware 1975 through 1978 is summarized in Table 2.4-16. Data in this table was collected at A11263 and depiet a moderately hard warmwater stream with a carbonate ionic base. The quality of Delaware water is relatively good in that it is well buffered and does not contain excessively high concentrations of major cations and anions or ions considered essential plant nutrients (Table 2.4-16). Lead and zinc are the only transition series elements present in significant quantities. While temporal changes in Delaware water quality do occur, they are not as severe as the shifts on smaller streams because of the greater flow.

2.4.7.2 Water Temperatures

A summary of the monthly maximum, minimum, and average temperatures of the Schuylkill River water at Pottstown for the period 1957-74 is given in Table 2.4-17.

2.4.7.3 Sediment Characteristics

Records of suspended sediment discharge in the Schuylkill River at Manayunk (river mile 14.2) are available from 1947. This station is about 34 miles downstream of the Limerick plant site. The maximum and minimum suspended sediment concentration in the river flow at this station has varied from 4910 mg/l (December 30, 1948) to 1 mg/l (frequently). The observed maximum and minimum daily sediment loads since 1947 are 650,000 tons on August 19, 1955, and less than 0.05 ton on September 2, 1966, respectively. Daily suspended sediment discharge, and mean concentration for water year 1975-76 are shown in Table 2.4-18. A duration table for suspended sediment concentration at Manayunk is given in Table 2.4-19. A double mass curve of cumulative annual suspended sediment discharge against cumulative annual water discharge at Manayunk, for the period 1948-76, is shown in Figure 2.4-9 (Ref 2.4-1). There has been a marked decrease in the rate of suspended sediment transported by the Schuylkill River since 1955. This is due to restoration activities in the upper catchment conducted by the Commonwealth of Pennsylvania. These restoration activities included dredging of the river channel, construction of "on-stream desilting basins," and regulation of coal mining activities in the basin.

2.4.8 WATER IMPOUNDMENTS

There are no major lakes or ponds in the site vicinity. A spray pond has been constructed at the site to serve as the ultimate heat sink for the plant. The bottom of this pond is at 241 feet (MSL). The pond was constructed by excavation only. At the bottom, it is a 600-foot by 400-foot rectangle, with semicircular ends of a radius of 200 feet. From this spray pond, water is pumped to the plant for the residual heat removal and emergency service water systems. After circulation through coolers and heat exchangers, warm water is returned to the spray pond through a network of spray nozzles. The water surface area of the pond at the operating elevation of 251 feet (MSL) is 9.9 acres. An additional 7.6 acres of the surrounding area, including roads, cut surfaces, and natural terrain, drain toward the pond. Runoff from the cut faces is intercepted by a drainage ditch along the outside edge of a peripheral service road at 255 feet (MSL), and is diverted to culverts that discharge into the spray pond. Along the north edge of the pond, the roadway slopes to 252 feet (MSL) for a length of 60 feet, with a 9% upward slope at either end to connect it to 255 ft (MSL). This depressed portion of the roadway is designed to function as the crest of an uncontrolled

emergency spillway. The spill is directed to a draw that drains northward into Sanatoga Creek.

Water, in small quantities, will spill over the emergency spillway from the spray pond to Sanatoga Creek. The capacity of the spray pond is such that these spills would occur only if a storm of severity higher than a 25-year, 24-hour storm occurred at a time when the pond was at the normal operating level of 251 feet (MSL).

2.4.9 CONCLUSION

Information regarding the ambient water quality of surface water bodies in the site vicinity has been presented. Low flow characteristics of the Schuylkill River, which is the stream receiving the blowdown discharge from the plant, are described. The locations of downstream users who could be affected by plant discharges, and of river control structures that may affect the dilution and travel time of effluents, are identified. This information would be useful in analyzing the transport of effluents in water required to meet the criteria of 10 CFR 50, Appendix I.

2.4.10 GROUNDWATER HYDROLOGY

Investigation of regional and local groundwater conditions indicates that the construction and normal operation of the Limerick Generating Station will have no adverse effects upon the groundwater resources in the region and at the site.

2.4.10.1 Description of Aquifers

Groundwater in the region occurs in sedimentary rocks of the Triassic Newark Group. This group includes the Stockton Formation and overlying Lockatong, Hammer Creek, and Brunswick lithofacies (Ref 2.4-11).

Although other units provide some groundwater in the region, the Brunswick is the most widespread source and the only significant aquifer at the plant site. The Stockton Formation is at great depth beneath the plant site, and is not of hydrologic importance in the site area.

The Brunswick, Hammer Creek, and Lockatong lithofacies are time equivalent units. The Hammer Creek lithofacies do not occur in the site area, and the Lockatong lithofacies only ocur in the northern part of the plant site.

The Brunswick is composed of red shale, sandstone, and siltstone locally interbedded with a few thin zones of the Lockatong, a dark gray argillite. Bedding ranges from a few inches to a few feet thick, with an average thickness of about 2 feet.

The rocks of the Brunswick lithofacies are very fine-grained, and primary permeability due to porosity is small. Most of the ground water movement within the Brunswick follows secondary openings, primarily fractures and joints. The fractures that parallel the bedding planes are usually tight and, probably, contribute little to the permeability; most important are the nearly vertical joint planes. Where present, joints provide an interconnected series of channels through which ground water can flow, giving the material low to moderate permeability (Ref 2.4-12). The number and width of the joints vary; consequently, the permeability differs from one location to another. For example, in a series of beds 100 feet thick there may be only a few beds in which the joints are well-developed.

In the Brunswick, unconfined water is encountered at shallow depth; deeper wells may encounter water under confined conditions. Yields from wells that penetrate the Brunswick vary widely because of lateral and vertical variations in lithology, uneven spacing of joints, and locally complex structure. Fault zones in the Triassic rocks have been found to be barriers to the flow of ground water; wells located near them generally have very low yields. The median yield of drilled municipal and industrial wells in the Brunswick is about 110 gallons per minute (gpm); the median transmissivity is 1100 gpd/foot. Yields in excess of 300 gpm are rare, and obtained from wells that intersect a larger number of water-bearing zones (Ref 2.4-12).

Recharge to the Brunswick occurs when infiltration of precipitation into the relatively impervious soil percolates down through weathered rock. The water table generally follows the profile of the land surface; groundwater flow is from high to low topographic areas. Groundwater movement is prevalent only in the upper portion of the Brunswick, where the fracture density is greatest. Poor water quality, and low yields in wells below a depth of about 600 feet, indicate little groundwater movement below that depth.

2.4.10.2 Site Groundwater Occurrence

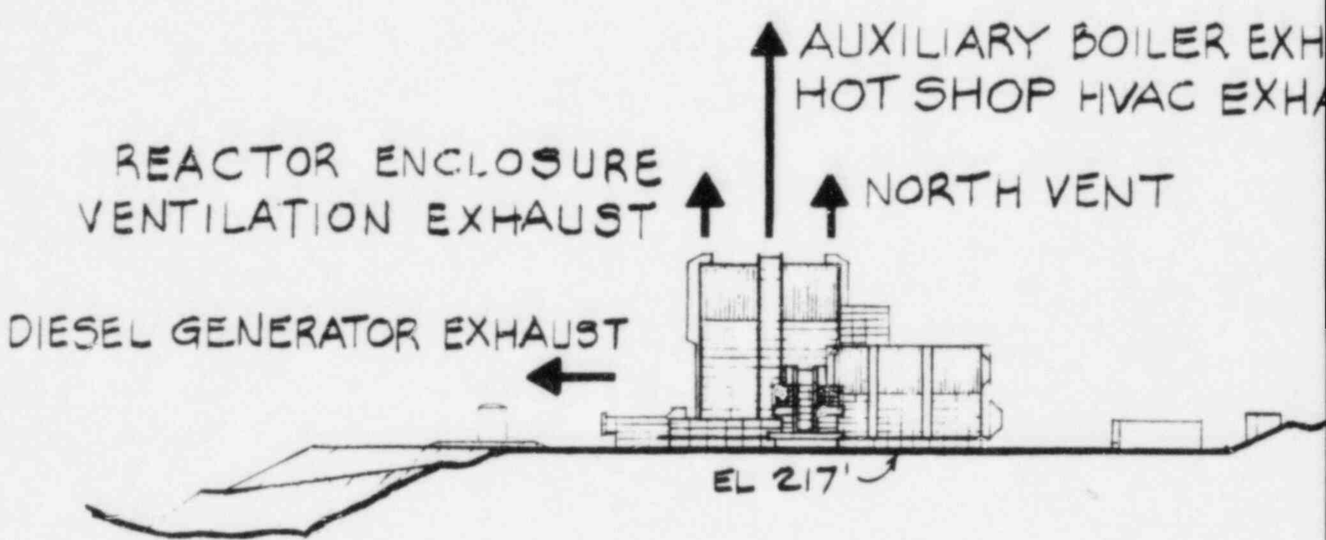
Water levels are monitored in observation wells at the spray pond and in the power block area as part of a continuing program to monitor the direction of groundwater movement and water table elevations. The locations of observation wells are shown in Figure 2.4-11.

Shallow borings completed in the upper weathered zone encounter unconfined groundwater. Deeper borings (more than about 150 feet) penetrate sandstone layers with fine-grained interbeds that contain water under confined conditions. A pumping test at the site indicated that hydraulic connection between sandstone layers is very poor, depending upon open, interconnected fractures of the fine-grained interbeds. Static water levels

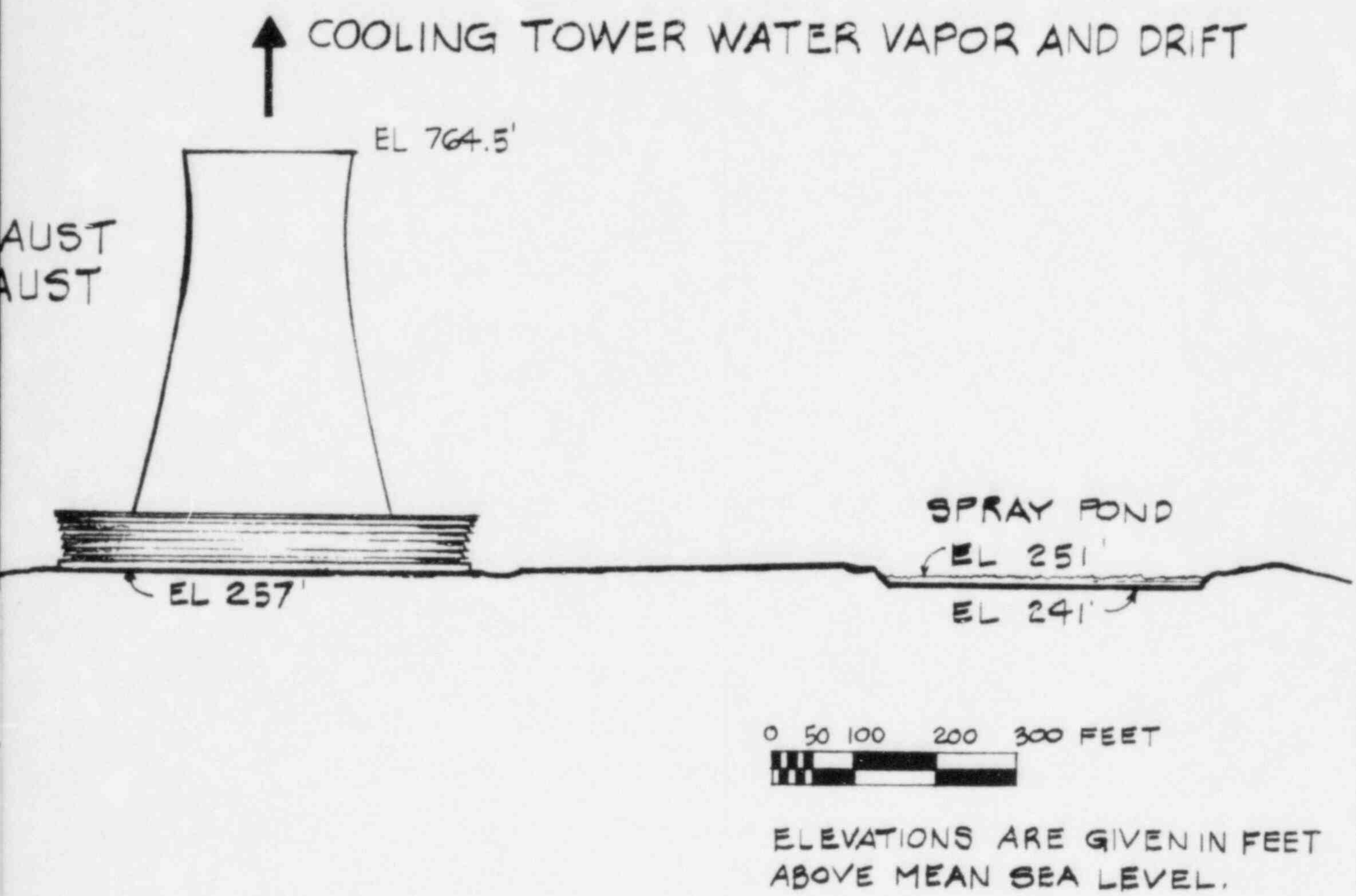
CHAPTER 3

FIGURES

Figure No.	Title
3.4-11	Perkiomen Makeup Water System General Plan and Profile
3.4-12	Perkiomen Pump Structure General Arrangement
3.4-13	Perkiomen Pump Structure Plan
3.4-14	Perkiomen Pump Structure Section
3.5-1	Liquid Waste Management Subsystems
3.5-2	Offgas System Process Flow Diagram
3.5-3	Flow Diagram of the Solid Radwaste System
3.6-1	Schematic Of Station Waste Water Effluent
3.7-1	The Sewage Treatment Facilities
3.7-2	Underground Storm Drainage System
3.9-1	Power Generation Schematic Outlets
3.9-2	Detail Illustrations of Transmission Lines (16 pages)
3.9-3	500 kV Single Circuit Lattice Steel Structure
3.9-4	Vertical and Triangular Conductor Configuration on Single Circuit Structure
3.9-5	Double Circuit Vertical Tubular Structure
3.9-6	Wide Flange Type Steel Structure
3.9-7	Wide Flange Type Structure Showing Extension
3.9-8	Transmission Line Routing



EAST ELEVATION



LIMERICK GENERATING STATION
UNITS 1 AND 2
ENVIRONMENTAL REPORT

EAST ELEVATION OF STATION

FIGURE 3.1-3

REV. 19, 04/83

to three 50% capacity centrifugal dry pit service water pumps, rated at 18,000 gpm each. These pumps, located within the circulating water pump structure, convey the service water to a supply header for distribution to the various heat exchangers. The average service water temperature rise in the heat exchangers is less than the circulating water temperature rise in the main condenser at full power operation. The service water from the heat exchangers is collected in a return header and piped via a 36-inch carbon steel pipe to the top of the cooling tower fill ring. The service water then becomes thoroughly mixed with the circulating water in falling through the cooling tower fill and into the basin. A normally closed, 20-inch service water bypass line terminating in the basin can be used to bypass the fill during cold weather startup, when the circulating water system is not operating. The service water system contributes less than 5% of the total heat dissipation to the atmosphere at full power; the circulating water system is the main source of the heat to be rejected.

3.4.3 NATURAL DRAFT EVAPORATIVE COOLING TOWERS

Each generating unit is served by one cross-flow natural draft evaporative cooling tower that cools circulating and service water by dissipating heat to the atmosphere at approximately 8 billion Btu per hour during full power operation. The cooling towers are also used, if in service, during shutdowns to cool normal service water, RHRSW, and ESW. The heat dissipation rate during shutdown is less than 10% of the heat dissipation rate during full power operation. Each cooling tower has been designed to meet the following conditions: 7.9 billion Btu/hr heat rejection, 75°F wet-bulb temperature, 66% relative humidity, 13.9°F wet-bulb approach, 33.5°F temperature range, and 476,600 gpm water flow.

Figure 3.4-2 shows the major features of one of the two identical cooling towers. The warm water falls through distribution orifices in the bottom of the distribution flume on top of the fill ring, and is broken into droplets as it cascades through the fill. The water droplets are cooled by the air flowing horizontally through the fill, which is constructed of PVC (ASTM C221-67, Type B), comprising about 5.3 million square feet of wetted surface area per tower. The natural draft, caused by pressure and density differences between entering and exiting air, passes about 50 million cfm of air through each tower during full power operation. Cooling tower air flow data are given in Table 3.4-1. The cooling is achieved in part by sensible heat transfer, but mostly by evaporating a portion of the water. Evaporation rates by month are given in Table 3.3-1. A small portion of the water droplets (drift) is carried through the drift eliminators with the water vapor. Although the drift rate is guaranteed for a maximum of 0.2% of the recirculating water flow, about 0.03% (200 thousand gpd per tower) is actually

expected. The rest of the water is collected in the full diameter basin for recycling and blowdown.

Blowdown is taken continuously from the cooling tower basin, which is the coolest water in the recirculating system. The blowdown temperature varies seasonally because of ambient atmospheric conditions, and is shown in Table 3.4-2. Performance curves for the cooling tower are shown in Figure 3.4-3. The maximum cooling tower blowdown temperature will be 94°F. Blowdown will flow from the cooling tower basin over a 20-foot-long weir crest at elevation 262.4 feet MSL. The blowdown rate equals the rate of pumping makeup water minus the rate of evaporation and drift, as shown in Table 3.3-1. Evaporation curves for the cooling tower are shown in Figure 3.4-4. A small heat load (usually less than 1% of the total two-unit full-power heat rejection) is rejected to the Schuylkill River because the blowdown temperature is warmer than the river temperature. The environmental effects of heat dissipation are discussed in Section 5.1.

3.4.4 EMERGENCY SPRAY POND

The spray pond is an emergency cooling system that is used during plant shutdown if the cooling towers are not available for heat dissipation. Although the spray pond is not intended for use during normal operation or normal shutdown, it is used during loss of offsite power or loss-of-coolant accident (LOCA). Environmental impacts due to infrequent testing, and emergency operation of the spray pond are insignificant.

The spray pond system is common to both Units 1 and 2, and is shown in Figure 3.4-5. The system consists of the spray pond pump structure, spray pond spray nozzles, and associated piping and valves. The spray pond pump structure, located at the south edge of the spray pond, contains four RHRSW wet-pit turbine pumps rated at 9000 gpm each, and four ESW wet-pit turbine pumps rated at 6500 gpm each. Each pump is installed in its own bay. A removable screen is placed at the entrance of each of the bays. The spray pond has a capacity of 29.6 million gallons, a surface area of 9.9 acres, and a depth of 10 feet, with a normal water surface elevation of 251 feet MSL. The spray system consists of two spray networks for each of the two safeguard divisions. For winter startup, spray network bypass lines are provided so that when low ambient temperatures exist, the total flow can be routed directly to the pond without passing through the spray network.

The spray pond performance (for a once-in-40-year DBA) has been analyzed to ensure that the design spray pond volume is adequate for 30 days of cooling, and that the cooling water temperature does not exceed the design limit during design meteorological conditions. The transient analyses for performance evaluation assumed that the spray pond is subjected to the heat load from a LOCA on one unit, and emergency shutdown and cooldown of the

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3.9 TRANSMISSION FACILITIES

3.9.1 DESCRIPTION OF TRANSMISSION FACILITIES

As described in Section 3.2 of the Environmental Report-Construction Permit Stage and 3.7 of the Final Environmental Statement, five outlets for generation will be provided as shown schematically in Figures 3.9-1 and 3.9-8. The existing Peach Bottom to Whitpain 500-kV line will be routed through the Limerick 500-kV substation where the line will be cut and reconnected to provide two generation outlets. A 500-kV Limerick to Whitpain line will be constructed entirely on existing rights-of-way (ROW). This line is referred to in Sections 3.9.1.1 and 3.9.2.1. Two 230-kV Limerick to Cromby lines will be constructed along two existing railroad ROWs. These lines are referred to in Sections 3.9.1.2 and 3.9.2.2.

In addition to these previously described transmission facilities, two new 230-kV lines are required. A new 230-kV line from Cromby to North Wales will be constructed on existing ROW. This line is discussed in greater detail in Sections 3.9.1.3 and 3.9.2.3. A new 230-kV line from Cromby to Plymouth Meeting will be constructed using a combination of existing and railroad ROW. This is discussed in greater detail in Sections 3.9.1.4 and 3.9.2.4.

Figure 3.9-2 provides a detailed illustration of the transmission facilities associated with the Limerick Generating Station.

3.9.1.1 Limerick to Whitpain 500-kV Line

The Limerick to Whitpain 500-kV line was discussed in Section 3.2 of the Environmental Report-Construction Permit Stage and Section 3.7 of the Final Environmental Statement. In accordance with NRC Regulatory Guide 4.2 and 10 CFR 51, no further discussion is necessary.

3.9.1.2 Two Limerick to Cromby 230-kV Lines

The two Limerick to Cromby lines were discussed in Section 3.2 of the Environmental Report-Construction Permit Stage and Section 3.7 of the Final Environmental Statement. In accordance with NRC Regulatory Guide 4.2 and 10 CFR 51, no further discussion is necessary.

3.9.1.3 Cromby to North Wales 230-kV Line

The proposed Cromby to North Wales 230-kV transmission line will be approximately 16 miles in length. Philadelphia Electric Company owns, or has easement for, 100% of the proposed ROW for this line. The ROW varies between 150 and 300 feet in width. At the present time, this ROW contains a 138-kV lattice tower

transmission line. Most properties adjacent to the ROW are farms and much of the ROW is farmed. For this reason, tree trimming for the Cromby-North Wales line will be minimal. Less than 5% of the ROW is wooded. No changes in land usage are anticipated. The new line will cross the Schuylkill River, Perkiomen Creek, and the northeast extension of the Pennsylvania Turnpike.

The route selection for this line was based upon using an existing ROW. The existence of this ROW makes further consideration of alternative routes for this line impractical, as discussed in Section 10.9.

The new line will be supported on gray, single-circuit, triangular-configuration, tubular steel structures (Figure 3.9-4) for a distance of approximately 15 miles from Cromby to West Point Pike in Upper Gwynedd Township. The conductor configuration will change from triangular to vertical where sharp turns in the ROW are encountered.

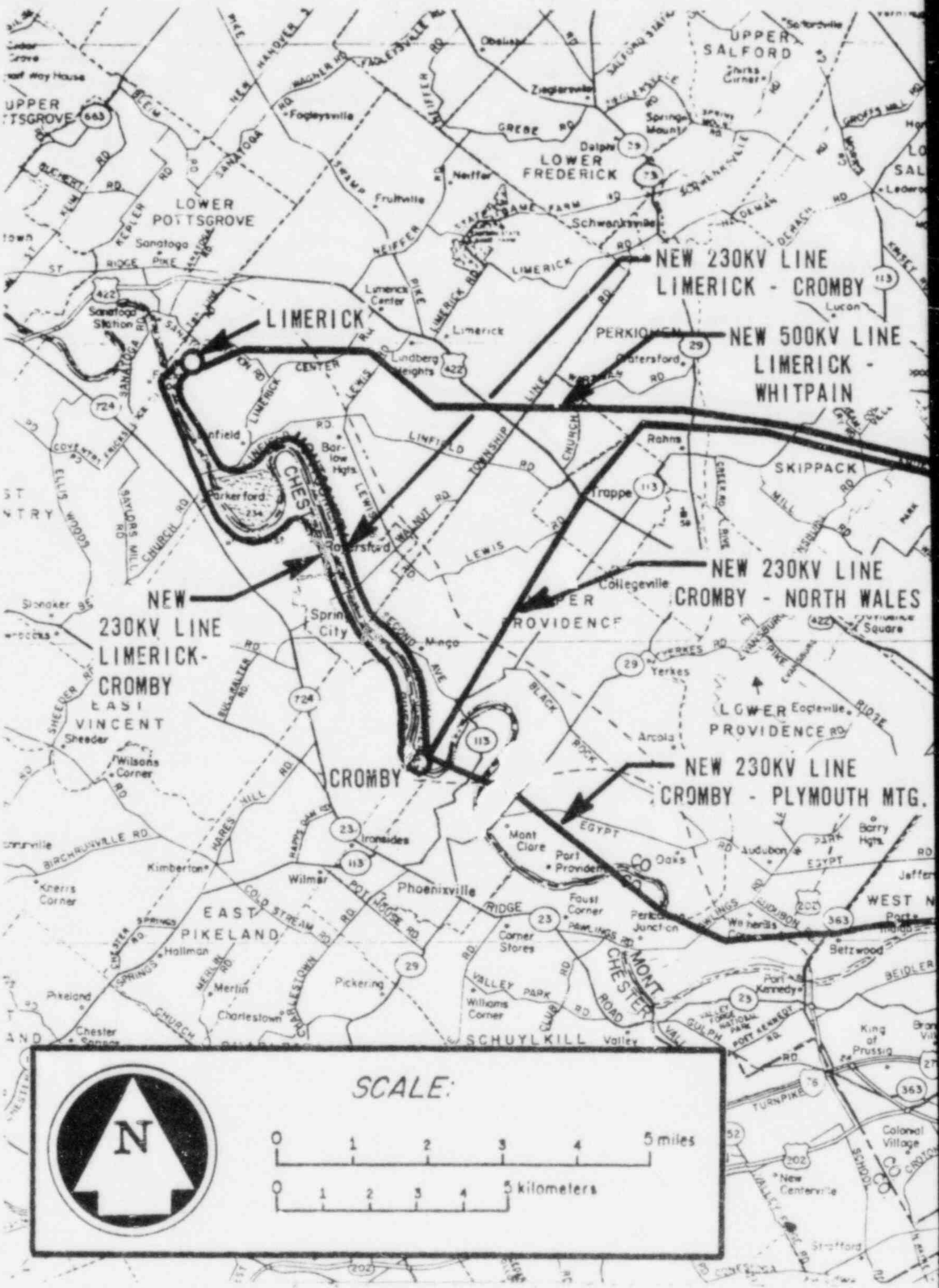
The last mile of the line requires installation of double-circuit vertical tubular structures (Figure 3.9-5). These structures will carry the new line and the existing Whitpain-North Wales line, which must be relocated, to new bus takeoff positions at North Wales Substation. The double-circuit vertical structures are needed because of the narrowness of the ROW in this area. These structures will also be painted gray.

The Cromby-North Wales line will be a high-capacity, 230-kV line with two 1590-kcmil (1.545 inches in diameter) ACSR conductors per phase. This line will have a summer normal rating of 1200 mVA and an emergency rating of 1400 mVA. The ruling span for this line will vary between 600 and 1200 feet depending upon terrain. All clearances will meet or exceed the minimum requirements of National Electric Safety Code (NESC) Section 23. The line will be designed to maintain a minimum vertical clearance to the ground of 25 feet at a maximum conductor temperature of 140°C, (284°F). This temperature is the conductor temperature used to establish clearances for ACSR conductors. The maximum electric field strengths anticipated for typical spans are indicated on the ROW cross sections (Figure 3.9-2).

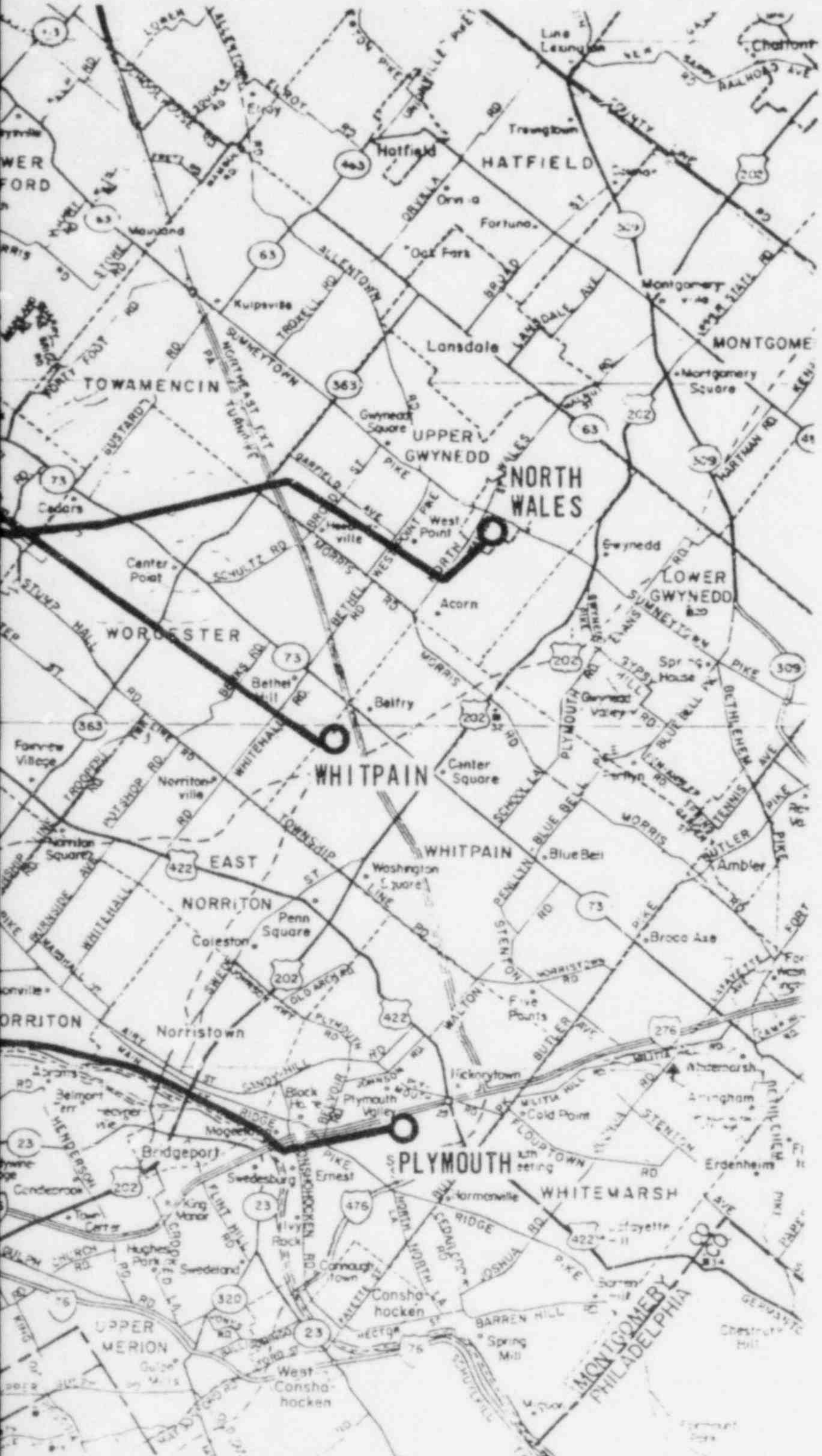
The visual impact of the new line will be minimized by locating the new structures next to the existing line towers. This procedure takes full advantage of existing foliage which now shields the line towers from view and ensures that no structures will be placed where the general public has become accustomed to seeing only the conductors.

3.9.1.4 Cromby to Plymouth Meeting 230-kV Line

The proposed Cromby to Plymouth Meeting 230-kV transmission line will be approximately 14.5 miles long and will be constructed on existing Conrail and Philadelphia Electric Company ROW. The



NEW 230KV LINE
LIMERICK - CROMBY
LIMERICK
NEW 500KV LINE
LIMERICK - WHITPAIN
NEW 230KV LINE
CROMBY - NORTH WALES
NEW
230KV LINE
LIMERICK-
CROMBY
CROMBY
NEW 230KV LINE
CROMBY - PLYMOUTH MTG.
UPPER SALFORD
LOWER FREDERICK
LOWER POTTSGROVE
SKIPPACK
PROVIDENCE
LOWER PROVIDENCE
EAST VINCENT
EAST PIKELAND
Phoenixville
SCALE:
N
0 1 2 3 4 5 miles
0 1 2 3 4 5 kilometers



LIMERICK GENERATING STATION
UNITS 1 AND 2
ENVIRONMENTAL REPORT

TRANSMISSION LINE ROUTING

FIGURE 3.9-8

REV. 12, 04/83

CHAPTER 7

ENVIRONMENTAL EFFECTS OF ACCIDENTS

TABLE OF CONTENTS

Section	Title
7.1	STATION ACCIDENTS INVOLVING RADIOACTIVITY
7.1.1	Approach to the Analysis of Class 1-8 Accidents
7.1.2	Models and Data Used to Evaluate the Environmental Consequences of Class 1-8 Accidents
7.1.2.1	Radiation Dose Models and Data for Class 1-8 Accidents
7.1.2.2	Source Term Models and Data for Class 1-8 Accidents
7.1.2.3	Atmospheric Diffusion Estimates for Class 1-8 Accidents
7.1.3	Class 1-8 Accident Analysis
7.1.3.1	Class 1 - Trivial Incident Inside Primary Containment
7.1.3.2	Class 2 - Small Releases Outside Primary Containment
7.1.3.3	Class 3 - Radwaste System Failure
7.1.3.3.1	Equipment Leakage or Malfunction
7.1.3.3.2	Offgas Treatment System Failure
7.1.3.3.3	Release of Waste Sludge Tank Contents
7.1.3.4	Class 4 - Fission Products to Primary System (BWR)
7.1.3.4.1	Fuel Cladding Defects
7.1.3.4.2	Off-Design Transients that Induce Fuel Failure
7.1.3.5	Class 5 - Fission Products to Primary and Secondary Systems (PWR)
7.1.3.6	Class 6 - Refueling Accidents
7.1.3.6.1	Fuel Assembly Drop
7.1.3.6.2	Heavy Object Drop onto Fuel in Core
7.1.3.7	Class 7 - Spent Fuel Handling Accidents
7.1.3.7.1	Fuel Assembly Drop in Fuel Storage Pool
7.1.3.7.2	Heavy Object Drop onto Fuel Racks
7.1.3.7.3	Fuel Cask Drop
7.1.3.8	Class 8 - Accident Initiation Events Considered Design Basis Evaluation in Safety Analysis Report
7.1.3.8.1	Loss of Coolant Accidents (LOCA)
7.1.3.8.2	Control Rod Accidents
7.1.3.8.3	Steam Line Break Accidents
7.1.3.9	Summary of Environmental Consequences and Public Risk of Class 1-8 Accidents
7.1.4	Approach to the Analysis of Severe Accidents
7.1.4.1	Models and Data
7.1.4.1.1	Source Term Description and Associated Frequencies
7.1.4.1.2	Consequence Model

CHAPTER 7

TABLE OF CONTENTS (Cont'd)

Section	Title
7.1.4.1.3	Uncertainty
7.1.4.2	Analysis
7.1.4.3	Results
7.1.4.3.1	CCDFs
7.1.4.3.2	Risk Considerations
7.1.4.4	Conclusions
7.1.5	References
7.2	TRANSPORTATION ACCIDENTS INVOLVING RADIOACTIVITY
7.3	OTHER ACCIDENTS
7.3.1	Storage and Use of Oil
7.3.2	Storage of Condensate and Refueling Water
7.3.3	Storage and Use of Acid and Caustic
7.3.4	Storage and Use of Chlorine
7.3.5	Storage and Use of Compressed Gases
7.3.6	Summary

CHAPTER 7

TABLES (Cont'd)

No.	Title
7.1-17	Class 8.2(b) Accident - Radioactivity Released as a Result of a Rod Drop Accident
7.1-18	Class 8.3(b) Accident - Radioactivity Released as a Result of Steam Line Break - Small Pipe
7.1-19	Class 8.3(b) Accident - Radioactivity Released as a Result of a Steam Line Break - Large Pipe
7.1-20	Summary of Maximum Exclusion Area Boundary Doses Resulting From Accidents
7.1-21	Summary of Population Doses Resulting From Accidents
7.1-22	Source Term Characteristics - Point Estimate
7.1-23	Frequencies of Table 7.1-22 Source Terms
7.1-24	Activity in the Limerick Reactor Core at 3293 MWt
7.1-25	Permanent Resident Population for the Limerick Site
7.1-26	Average Values of Environmental Risks Due to Accidents Per Reactor-Year
7.2-1	Environmental Impact of Transportation of Fuel and Waste

CHAPTER 7

FIGURES

No.	Title
7.1-1	Schematic Outline of Consequence Model
7.1-2	Median CCDF of Bone Marrow Dose Greater than 200 Rem
7.1-3	Median CCDF of Population Exposure
7.1-4	Median CCDF of Acute Fatalities
7.1-5	Median CCDF of Latent Cancer Fatalities
7.1-6	Median CCDF of Ex-Plant Costs
7.1-7	Median Individual Risk of Early Fatality as a Function of Distance

CHAPTER 7

ENVIRONMENTAL EFFECTS OF ACCIDENTS

7.1 STATION ACCIDENTS INVOLVING RADIOACTIVITY

The purpose of this section is to consider the potential radiological effects on the environment of accidental events and to compare these potential effects with those of normal station operation and natural background radiation. Radiological effects that result from normal station operation are discussed in Section 5.2, and natural background radiation is discussed in Section 6.4.

A detailed accident and safety analysis is a normal part of the design and licensing of each power station. The results of this analysis are presented to the NRC in the form of safety analysis reports (SARs). These reports contain detailed descriptions of the facility and station site, as well as a highly conservative analysis of the effects of normal and abnormal plant conditions. In addition to the analysis presented in the SAR, further examination of the environmental effects of normal and abnormal station conditions, based upon realistic parameters, is required to be presented in this Environmental Report. An assessment of the risks associated with the Limerick plant from accidents more severe than included in the design bases for the station was undertaken and is required to be presented in Section 7.1.4.

There are two main aspects of station safety: prevention of station accidents, and containment of radioactivity in the event of an accident. Prevention of station accidents begins with conservative design of the reactor and its control system, and conservative engineering of the reactor installation. Starting with this base, the designer seeks to anticipate the possible sources of malfunction, and to make provisions for mitigating their effects in the design. A strict quality assurance program ensures high component and system reliability.

Radioactive materials produced in the core of the reactor are contained within the station by a number of successive barriers that are incorporated in the station design. These barriers are the fuel material, zircaloy fuel cladding, the steel wall of the reactor vessel, and the primary and secondary containment systems. Containment of radioactivity in the event of an accident also involves the incorporation of engineered safety

features (ESF) in the station design, such as radiation shields, emergency cooling systems, and air filtration systems.

In considering the environmental effects of postulated station accidents, several important distinctions must be made from other station environmental effects. The estimated effects are potential rather than certain. As a result of measures taken, or prevention of accident through design, manufacture, and operation, occurrences of accidental events in operating nuclear power plants have been rare. The improbability of accidental events in operating nuclear plants has been maintained at this low level through design review, operating limits, and quality assurance procedures. Therefore, the environmental effects of these potential events must be considered in conjunction with their probability of occurrence.

7.1.1 APPROACH TO THE ANALYSIS OF CLASS 1-8 ACCIDENTS

In the Federal Register of June 13, 1980 (45FR 40101), the Nuclear Regulatory Commission published a statement of interim policy regarding accident considerations. This statement withdrew the proposed annex to Appendix D of 10CFR50 and suspended the rulemaking procedures associated with it. It also put forward the Commission's interim policy that "...Environmental Impact Statements shall include consideration of the site-specific environmental impacts attributable to accident sequences that can result in inadequate cooling of the reactor fuel and in melting of the reactor core. In this regard, attention shall be given both to the probability of occurrence of such releases and to the environmental consequences of such releases."

Accordingly, Section 7.1.4 describes an analysis of the public risk associated with these severe accidents.

Although, as is described above, the proposed annex was subsequently withdrawn, the information for accidents formerly designated as Class 1-8 is given in Sections 7.1.1 to 7.1.3. The public risk associated with these accidents is summarized in Section 7.1.3.9.

The occurrence of abnormal station conditions and accidental events must be considered in design, licensing, and operation of nuclear power plants. In technical terms, an accident is an unexpected chain of events (i.e., a process rather than a single event). In SARs, the basic events involved in various possible

station accidents are identified and studied with regard to the adequacy of the performance of the engineered safety features (ESF). In addition, the potential radiological effects of station accidents are analyzed by the evaluation of physical factors involved in each chain of events that might result in radiation exposures to humans. These factors include the meteorological conditions existing at the time of the accident, radionuclide uptake rates, and exposure times and distances, as well as the many factors that depend upon station design and the mode of operation. In these analyses, the factors affecting the consequences of each accident are identified and evaluated, and uncertainties in their values are discussed. Because some degree of uncertainty always exists in the prediction of these factors, it has become general practice in SARs to assume conservative values in making calculated estimates of radiation doses.

As a result of the highly conservative analysis, the radiation exposure levels calculated in SARs are not actually expected to be reached, even if the event initiating the accident occurs. In fact, the calculated exposures resulting from a DBA are generally far in excess of what would be expected, and do not provide a realistic means of assessing the radiological effects of postulated station accidents. In the analyses presented here, the radiation exposures associated with station accidents have been analyzed on a more realistic basis, as specified in the proposed annex to Appendix D of 10 CFR Part 50, which is referenced by NRC Regulatory Guide 4.2, Rev. 2 (Ref 7.1-1). In many cases, the assumptions are still conservative in that the most probable assumptions would result in even lower radiation exposure.

The effectiveness of measures that have been taken for accident prevention is judged by the frequency at which the accident occurs; that is, the accident probability. The effectiveness of the measures taken in containment of radioactivity can be judged by the calculated values of the radiological exposures associated with each accident. As discussed in the Federal Register (36 FR 22851) for the proposed annex to Appendix D of 10 CFR Part 50, the determination of the environmental impact of potential accidents requires the consideration of both the potential exposures, and the probabilities of receiving these exposures.

The environmental impact of the postulated accidents is evaluated for eight accident classes identified in Table 7.1-1. These classes are defined in the proposed annex to Appendix D of 10 CFR Part 50.

7.1.2 MODELS AND DATA USED TO EVALUATE THE ENVIRONMENTAL CONSEQUENCES OF CLASS 1-8 ACCIDENTS

Maximum individual dose estimates are based upon a receptor located at the exclusion area boundary. Man-rem dose estimates are based upon the year 2000 population projections. The population distribution as a function of distance and sector for the year 2000 has been estimated, and presented in Section 2.1. The total population dose was determined by taking the product of the dose and the number of people receiving that dose in an area segment defined by a 22.5° sector, at a particular distance from the station, and summing the product of each 22.5° sector for a distance out to 50 miles from the station.

7.1.2.1 Radiation Dose Models and Data for Class 1-8 Accidents

The models used are based upon NRC Regulatory Guides 1.3 (Ref 7.1-2) and 1.25 (Ref 7.1-3). The following assumptions are basic to both the model for the whole-body dose due to immersion in a cloud of radioactivity, and the model for the thyroid dose due to inhalation of radioactivity:

a. Direct radiation from the station is negligible compared to whole-body radiation due to immersion in the cloud of radioactivity.

b. All radioactive releases are treated as ground level releases, regardless of the point of discharge.

c. Continuous release atmospheric dispersion factors are applicable, and cloud depletion due to ground deposition is assumed to be insignificant.

d. The dose receptor is a standard man, as defined by the International Commission on Radiological Protection (ICRP) (Ref 7.1-4).

For all distances and time periods, the semi-infinite cloud model is used to calculate the whole-body dose. The procedure results in population exposures that are conservative.

The semi-infinite, whole-body gamma dose is given by the following equation from TID-24190 (Ref 7.1-5):

$$rD_{oo} = (0.25)\,(X/Q) \sum_{i=1}^{N} (Qi)(Ei) \qquad (7.1\text{-}1)$$

where:

rDoo = gamma dose from semi-infinite cloud (rad)
X/Q = atmospheric dilution factor (sec/meter^3)
N = number of isotopes
Qi = source strength for isotope i (curies)
Ei = average gamma energy for isotope i (MeV/dis)

The thyroid dose for a given time period is obtained from the following equation:

$$D = (X/Q)(BR) \sum_{i=1}^{N} (Qi)(DCFi) \qquad (7.1\text{-}2)$$

where:

D = thyroid inhalation dose (rem)
X/Q = atmospheric dilution factor (sec/meter^3)
BR = breathing rate (meter^3/sec)
N = number of isotopes
Qi = total activity of iodine isotope i released (curies)
DCFi = dose conversion factor for iodine isotope i (rem/curies inhaled)

Table 7.1-2 lists the physical data for the radiation dose models. The half-life values were taken from the Meek and Rider Report (Ref 7.1-6), and are in general agreement with those in TID-14844 (Ref 7.1-7) and ORNL-2127 (Ref 7.1-8). The values for the gamma energies are those given in the Table of Isotopes (Ref 7.1-9). The thyroid dose conversion factors are taken from the ICRP Committee II Report (Ref 7.1-10), and the breathing rates used in the calculations of inhalation doses are based upon the average daily breathing rates assumed in the ICRP Report, which are also used in the NRC Regulatory Guide 1.3 (Ref 7.1-2).

7.1.2.2 Source Term Models and Data for Class 1-8 Accidents

It is the purpose of this section to provide the general information used for accident evaluations.

The inventories of radioactive materials in the fuel pellets and fuel rod gap spaces in the reactor core depend upon the following:

a. Core power

b. Plant capacity factor

c. Temperature distribution in the pellets

d. Length of operating time prior to the accident or shutdown

e. Diffusion rates of radioisotopes through the fuel pellet materials.

Fission product inventories for the core and gap are based upon operation at 3458 MWt for 1000 days. Activity inventories for the total core, total gap, and gap of one fuel rod are given in Table 7.1-3. Reactor coolant concentrations are given in Table 7.1-4. These coolant concentrations were calculated using the methodology of NUREG-0016 (Ref 7.1-11).

7.1.2.3 Atmospheric Diffusion Estimates for Class 1-8 Accidents

Estimates of atmospheric diffusion (X/Q) have been made at the exclusion area boundary, the outer boundary of the low population zone (LPZ), and at 0.5, 1.5, 2.5, 3.5, 4.5, 7.5, 15, 25, 35, and 45 miles for each sector. These estimates have been made for periods of 2, 8, and 16 hours, and 3 and 26 days following a postulated accident. The sector-dependent model in Draft Regulatory Guide 1.145 (Ref 7.1-12) has been used.

The calculation procedure used to determine X/Q for the appropriate time periods following a postulated accident is described in Draft Regulatory Guide 1.145. The diffusion model presented in this guide is used to determine X/Q values for the first 2 hours following the accident. X/Q values for longer time periods are determined by logarithmic interpolation between the 2-hour accident value and the annual X/Q at each receptor point.

The annual X/Q values have been calculated using the model described in Regulatory Guide 1.111 (Ref 7.1-13). The Limerick emission has been classified as a low-level release, according to the criteria of Draft Regulatory Guide 1.145. This requires that the source be treated as ground level. This assumption has also been made in the annual X/Q calculations.

Meteorological data from Limerick Weather Station No. 1, from January 1972 through December 1974, have been used in the diffusion calculations. Lapse rate wind distributions have been computed using wind speed and direction from the 30-foot level, and temperature difference from the 266-26 foot height interval. The lapse rate, wind speed, and wind direction categories are consistent with the recommendations of Regulatory Guide 1.23 (Ref 7.1-14). The wind distribution used to calculate the 2-hour accident X/Q values has been normalized by directional sector, in accordance with Draft Regulatory Guide 1.145. This distribution is shown in Table 2.3.2-2. In each sector, the total frequency of wind speed and stability categories equals 100%. The stability classes designated as 1 through 7 in this distribution refer to the Pasquill classes A through G. A wind distribution computed in the standard manner is shown in Table 2.3.2-42. This distribution was used to calculate the annual X/Q values used in the logarithmic interpolation scheme.

The dispersion parameters developed by Pasquill (Ref 7.1-15) and Gifford (Ref 7.1-16) have been used in the accident calculations. Analytical approximations to these curves, developed by Eimutis and Konicek (Ref 7.1-17), have been used for sigma-y. The approximations of Busse and Zimmerman (Ref 7.1-18) have been used for sigma-z. A building wake correction of 2298m² was used. This is equal to one-half the minimum cross-sectional area of the reactor turbine enclosure complex.

The effective probability level is an adjustment necessary to equate the directionally dependent approach of Draft Regulatory Guide 1.XXX with the 50th percentile criterion previously employed by the NRC in the directionally independent model. This parameter is calculated as follows:

$$Pe = \frac{P(N/n)}{S} \qquad (7.1\text{-}3)$$

where:

Pe = effective probability level

P = desired probability level (50%)
N = total number of hours having valid wind and stability data in the period of record
n = total number of hours having valid wind and stability data in the directional sector of interest
S = total number of directional sectors (16)

The effective probability levels calculated for each sector at the Limerick Generating Station are listed in Table 7.1-5.

Cumulative frequency distributions of X/Q for the first 2 hours following a postulated accident were computed for distances of interest in each sector. These distributions were then plotted on a log probability scale. In each plot, the data points were enveloped by a fitting function, as described by Markee and Levine (Ref 7.1-19). The accident X/Q values in each directional sector were then obtained from the intersection of this function and the effective probability level.

Accident X/Q values for periods of 8 and 16 hours and 3 and 26 days following the accident have been determined by logarithmic interpolation between the maximum 2-hour and the maximum annual X/Q at each distance. A complete summation of the estimated X/Q values for the entire duration of the postulated accident is given in Table 7.1-6 for distances up to 50 miles for each sector.

7.1.3 CLASS 1-8 ACCIDENT ANALYSIS

In the following subsections, postulated accidents are identified and analyzed, and their radiological consequences are estimated.

7.1.3.1 <u>Class 1 - Trivial Accidents Inside Primary Containment</u>

Class 1 accidents are postulated as the release of small quantities of radioactive material inside the primary containment. The various mechanisms by which this may occur include small spills and small leaks from equipment and valve packing. A low level of continuous leakage from components such as valve packing stems, pump seals, and flanges, etc, is expected. Radioactivity release events of this class are considered as part of normal operating conditions, and analyzed along with radioactivity releases due to normal operation in Sections 3.5 and 5.2.

b. Large pipe break.

For these postulated breaks, considering the most probable operating conditions prior to the break and using realistic assumptions, the calculated two-phase mixture level in the reactor pressure vessel does not reach the steam line before isolation is complete. Therefore, only steam will issue from these breaks for the entire transient.

Small Pipe Break (of 0.25 ft^2): The following assumptions and parameters are postulated for evaluating the environmental consequences of a main steam line break accident for a small pipe break:

a. The primary coolant activity is based on an offgas release rate of 60,000 microcuries/sec after 30 minutes delay.

b. It is assumed that the main steam line will release coolant for 5 seconds after the isolation signal is received.

c. The total amount of steam escaping from the break is 2750 lb. This quantity is the sum of a steam loss for two time periods, a 0.5-second duration prior to reactor trip, and a 5-second duration to complete closure of the MSIVs.

d. Iodine in the fluid released to the atmosphere is at one-tenth the primary system liquid concentration.

e. Fifty percent of the iodines and 100% of the noble gas in the fluid exiting through the break are assumed to be released to the atmosphere.

f. Meteorology for less than 8 hours is used because the release from this accident is expected to last for less than 8 hours.

The total activity released to the environs is given in Table 7.1-18.

Large Pipe Breaks: The assumptions and parameters postulated for evaluating the environmental consequences of a main steam line break accident for a large pipe break are identical to those given for a small pipe break, with the exception that the total amount of steam escaping from the break is 36,000 pounds. This quantity is the sum of a steam loss for two time periods, a 0.5-second duration prior to reactor trip, and a 5-second duration to complete closure of the MSIVs.

The total activity released to the environs is given in Table 7.1-19.

In making an assessment of the probability of the occurrence of typical events considered as DBAs in the FSAR, a firm numerical estimate is not possible because of the extreme rarity of such events. Quality assurance for design, manufacture, and operation, and highly conservative design considerations combine to produce piping and vessels with an extremely low probability of failure. Therefore, when the consequences are weighted by probabilities, the environmental risk is low.

7.1.3.9 Summary of Environmental Consequences and Public Risk of Class 1-8 Accidents

In the preceding discussion, a number of postulated accidents have been identified and analyzed. These selected events cover the full range of accident analyses formerly required in the NRC guidelines. The resulting estimates of potential station EAB doses as a result of each postulated accident, along with an assessment of the likelihood of each event, are listed in Table 7.1-20.

In the column giving the general assessment of the likelihood of these events and conditions, several categories have been used. Those events that could be expected to occur at frequencies of from once per station lifetime to as often as once per year are classified "occasional". Those events or conditions that would be expected to occur at frequencies less than once per station lifetime are classified "rare". Finally, there are a number of events that are considered unlikely, with projected probabilities much less than once per station lifetime. These events have been classified "extremely rare".

Table 7.1-21 shows the estimated integrated exposure from each postulated accident to the population within 50 miles of the station. When considered with the probability of occurrence, the

annual potential radiation exposure of the population from all the postulated accidents is a small fraction of the exposure from natural ba^kground radiation and, in fact, is well within naturally occurring variations in the natural background.

From the results in the accident analysis, several specific conclusions can be reached concerning offsite doses:

a. The radiation exposures that would result from the occurrence of accidents are generally lower than those expected from normal operation, and much lower than that from natural background radiation.

b. The population exposure from possible station accidents is negligible when compared to the population exposure received from just the variation in natural background radiation, which overshadows the po ential population exposure from any accident considered.

c. Most of the radiation dose levels are so low as to be undetectable, even with the most sensitive modern radiation detection instruments.

d. When these potential exposures are considered in conjunction with their predicated frequencies of occurrence, it is judged that Class 1-8 accidents are small contributors to public risk. This judgment is based on the Reactor Safety Study (Ref. 7.1-20) and a published risk assessment of Class 3-8 accidents (Ref. 7.1-21). The Class 3-8 study estimated risk to the public using methodology that is similar to that used in the RSS. The results of the study showed that Class 3-8 accidents are small contributors to public risk relative to postulated more severe accidents.

7.1.4 APPROACH TO THE ANALYSIS OF SEVERE ACCIDENTS

This analysis is being provided at the request of the NRC staff (EROL Questions E450.1, E450.2, E450.3 and E450.4) to help provide a response to the Statement of Interim Policy on severe accident considerations published by the NRC in the Federal Register on June 13, 1980 (45FR40101).

The analysis uses a comprehensive probabilistic risk assessment of the radiological consequences of accidents at the Limerick site. The assessment includes consideration of both internal and external initiators and specifically includes contributions from internal events, earthquakes, and fires. Internal and external flood, transportation, tornado, and turbine missile initiators were found to be noncontributors to risk. The analysis involves highly improbable sequences of failures that are more severe than those postulated for the design basis for protective systems and engineered safety features. The analysis treats the frequency of occurrence of these events in a systematic fashion and includes an assessment of uncertainty in the frequencies, the phenomenological analysis, and the consequence analysis. The focus of the presentation in this section is on the median results for the radiological consequences of the postulated events.

The fire analysis consists of an estimate of the frequencies of fires in various rooms in the plant and models the effects of fires on various safety-related systems. The seismic analysis consists of a detailed study of the predicted characteristics of earthquakes at the Limerick site and of the response of structures and systems. The earthquakes predicted to cause accidents at the Limerick plant that are significant contributors to public risk are highly improbable and of a severity that has not occurred in the Limerick area in historical times. Given the occurrence of such an earthquake, it is highly likely that the public consequences of the earthquake itself directly on the surrounding area would be considerably more severe than the consequences of a seismically-induced accident at the plant.

Section 7.1.4.1 contains descriptions of the models and data employed in the analysis. Section 7.1.4.2 explains how the analysis was performed. The results are presented in Section 7.1.4.3. Section 7.1.4.4 contains conclusions.

7.1.4.1 Models and Data

Section 7.1.4.1.1 describes the fission product source terms and their associated frequencies. Section 7.1.4.1.2 contains a brief outline of the consequence model (the CRAC2 code) and the necessary input data. Section 7.1.4.1.3 discusses the uncertainty analysis.

7.1.4.1.1 Source Term Description and Associated Frequencies

The magnitude and frequency of fission product source terms used in this assessment are given in Tables 7.1-22 and 7.1-23, respectively. Source term is defined in this section to mean the magnitude of the release of fission products to the atmosphere, together with associated characteristics such as the time of release, warning time, duration of release, and rate of release of heat. These source terms have been selected to characterize the release anticipated from the various events analyzed in this section. These source terms tend to be conservative estimates that, for example, exclude deposition in the primary system and in the reactor enclosure. Detailed descriptions and the basis for selection of these source terms is given in the Limerick Generating Station Severe Accident Risk Assessment (Ref. 7.1-22).

a. OXRE -- This source term includes the releases due to oxidation reactions that occur as a result of an in-vessel or ex-vessel steam explosion, or a hydrogen explosion following core melt. Fire is the most important contributor to this source term, contributing 55 percent of the point estimate frequency of 1.3×10^{-7} per year.

b. OPREL -- This source term is dominated by gross rupture of the containment, either as a result of the buildup of noncondensable gases or a hydrogen burn, following loss of coolant inventory, core melt and vessel rupture. Again, fires contribute most significantly to the point estimate frequency, given 55 percent of the total of 2.0×10^{-5} per year.

c. $C4\gamma$ -- This source term is for an ATWS sequence ending in gross rupture of the drywell. Seismic and internal initiators are roughly equal contributors, and the total point estimate frequency is 1.3×10^{-7} per reactor year.

d. $C4\gamma'$ -- This source term is for an ATWS sequence ending in gross rupture of the wetwell, without loss of the suppression pool. Seismic and internal initiators are roughly equal contributors, and the total point estimate frequency is 1.1×10^{-7} per reactor year.

e. $C4\gamma''$ -- This source term is for an ATWS sequence ending in gross rupture of the wetwell, with loss of the suppression pool. Seismic and internal initators are

roughly equal contributors, and the total point estimate frequency is $1.3x10^{-8}$ per reactor year.

f. C123γ" -- This source term is for those sequences other than C4γ" that result in a gross rupture of the containment in the wetwell with loss of the suppression pool. It has a total point estimate frequency of $1.0x10^{-6}$ per year, to which fires contribute 58 percent.

g. LEAK1 -- This source term is for core melt sequences in which the containment leaks relatively slowly without operation of the standby gas treatment system (SGTS). The leakage sizes are smaller than for the γ failure modes and preclude gross rupture. These sequences are small contributors to public risk. The most important initiator is fire, and the total point estimate frequency is $3.2x10^{-6}$ per year.

h. LEAK2 -- This source term is for core melt sequences that are similar to those in LEAK1 except that the SGTS is operating effectively. The most important initiator is fire, and the total point estimate frequency is $1.8x10^{-5}$ per reactor year.

i. RB -- This source term includes the releases that result from the collapse of the reactor enclosure as a result of an earthquake. This leads to failure of the RHR heat exchanger lateral supports, which is assumed to lead to failure of the attached piping leading from the suppression pool. The pool will drain down to the pipe, leading to an open containment while the core melts. However, the suppression pool is still available for fission product scrubbing of the melt release of fission products.

j. VR -- This is a source term for the case in which the reactor vessel fails, and the containment fails shortly thereafter.

For internal events, this source term is caused by a spontaneous vessel rupture that can cause immediate containment failure. In this case, VR has a predicted point estimate frequency of $1.4x10^{-8}$ per reactor year.

For earthquakes, this source term is dominated by events in which there is failure of the vessel upper lateral supports, causing rupture of the four main steam lines while collapse of the reactor enclosure breaks pipework connected to the suppression pool (as in the case of source term RB). In this seismic case, VR has a predicted point estimate frequency of 3.7×10^{-7} per reactor year.

k. VRH20 -- This source term is also for the case in which the reactor vessel fails, and the containment fails shortly thereafter. The only difference between this source term and VR is that, in the case of VRH20, sufficient water is assumed to remain in the bottom of the vessel so that fission products are driven rapidly out into the atmosphere when molten core falls and causes the generation of steam. In the case of VR, the vessel is assumed to be completely dry, and it takes a relatively long time to drive the fission products out into the atmosphere. For spontaneous (internal) vessel rupture, VRH20 has a point estimate frequency of 1.4×10^{-8} per reactor year. In the seismic case, VRH20 has a point estimate frequency of 4.1×10^{-8} per reactor year.

The derivation of the point estimate frequencies is presented in Reference 7.1-22 and a discussion of the methods employed in the uncertainty evaluation of frequency is given in Section 7.1.4.1.3.1.

7.1.4.1.2 Consequence Model

The CRAC2 code was used to generate the complementary cumulative distribution functions (CCDFs) that are the final product of the analysis (Figures 7.1-2 to 7.1-6). The code is discussed in the PRA Procedures Guide (Ref. 7.1-23). A schematic outline of CRAC2 is given in Figure 7.1-1. Reference 7.1-23 should be consulted for discussion of such topics as exposure pathways, dosimetric and health effects models, and protective actions. Those parts of the input data or the coding that were modified to take account of Limerick specific features are discussed below.

7.1.4.1.2.1 Curies of Fission Products and Actinides in the Core at the Initiation of the Accident

The amounts (curies) of each radionuclide released to the atmosphere for each accident sequence or release category is obtained by multiplying the release fractions specified in the definition of the source term (Table 7.1-22) by the amounts that would be present in the core at the time of the hypothetical accident. These amounts are shown in Table 7.1-24 for the Limerick reactor.

7.1.4.1.2.2 Meteorological Data

The CRAC2 input data file for Limerick contains five years of consecutive hourly values of wind speed, wind direction, stability class, and precipitation intensity. These were processed from measurements taken at the Limerick site during the years 1972 to 1976.

These five years of data were processed by CRAC2 using the bin sampling technique. This required a minor code modification to enable CRAC2 to sample from the entire five years of data. The sampling techniques used by CRAC2 are described in Reference 7.1-23. The use of five years of data and the improved sampling techniques of CRAC2 yield a more complete and representative sample than has been possible using the "stratified sampling" techniques of CRAC. The data are consistent with those used and presented elsewhere in the EROL.

7.1.4.1.2.3 Population Distributions

The population distribution around the site has been assigned to a grid consisting of 16 sectors, the first of which is centered on due north, the second on 22-1/2 degrees east of north, etc. There are also 34 radial intervals (Table 7.1-24) that contain the predicted permanent resident population for the year 2000.

The population within 50 miles was taken from Tables 2.1-5 and 2.1-12 and assigned to the finer CRAC2 grid by ratioing by area. In the 50 to 500 mile range, 1980 U.S. census data were used on a county-by-county basis, and 1981 Canadian census data were used in census tracts, which are comparable in size to U.S. counties. The population within counties or tracts was again assigned to the CRAC2 population grid by ratioing by area. Extrapolation to the year 2000 was done by using regional growth rates from the

Census Department's Bureau of Economic Affairs, for the USA, and similar regional growth rates for Canada.

7.1.4.1.2.4 Evacuation Modeling and Other Protective Measures

The site-specific offsite emergency response plans are not complete at this time. Certain features of these plans, however, are considered to be sufficiently defined so as to be used in this analysis (e.g., 360-degree evacuation of the EPZ). These features were combined with a generic evacuation model, which was developed at Sandia Laboratories, on the basis of U.S. evacuation experience. It is described in the PRA Procedures Guide. This evacuation model is used with three alternative evacuation scenarios; 1-, 3- or 5-hour delay times with relative probabilities of 30, 40 and 30 percent, and a subsequent evacuation speed of 10 mph (4.5 m/sec). This is considered to be a "best estimate" model.

The source terms considered in Tables 7.1-22 and 7.1-23 include some with contributions from earthquakes. For evacuation for these sequences, the model was modified to incorporate a 3-hour delay for the whole population and an effective evacuation speed of 0.5 m/sec.

The "best estimate" model also includes an estimate of the response of people beyond the EPZ in the range 10 to 25 miles. They are assumed to continue their normal activities for 12 hours after the passage of the cloud, at which time they are rapidly relocated. In the event of an earthquake, this period is assumed to be 24 hours. Equivalent reductions in predicted dose could be achieved by other countermeasures such as assuming that people shelter in their basements or large buildings for a day or two before relocating; that is, significant reductions in predicted dose could be achieved by a choice of simple countermeasures. The outer limit of 25 miles is chosen because, in general, calculations with CRAC2 show that, even with conservative fission product source terms, life-threatening acute doses are rarely predicted beyond this distance, even in the most adverse of weather conditions.

7.1.4.1.2.5 Economic Costs

The necessary input to the calculation of economic costs in CRAC2 includes several unit costs such as the cost of evacuating or relocating a person and the cost of decontaminating an acre of farm land or developed land. These costs are given in Reference

7.1-20 and have been updated to 1980 to allow for inflation. In addition, land use statistics, farm land values, farm product values, dairy production, and growing season information are required by CRAC2. These statistics are provided on a county-wide basis within 50 miles and on a state-wide basis for larger distances. The various economic inputs are tabulated in Reference 7.1-22.

7.1.4.1.3 Uncertainty

Reference 7.1-23 lists 51 modeling assumptions or parameter variations to which the complementary cumulative distribution functions (CCDFs) may be sensitive. However, an uncertainty analysis taking account of all 51 parameters would be prohibitively time consuming. Instead, four major sources of uncertainty were chose; (a) the frequencies of the source terms given in Table 7.1-23; (b) the magnitude and associated characteristics of the source terms; (c) the evacuation and sheltering modeling; and (d) the modeling of health effects.

Consideration of this limited set of uncertainties is sufficient to establish plausible bounds on the CCDFs; that is, more detailed uncertainty analysis would not be expected to produce results that are likely to lie outside the bounds established by the more limited uncertainty analysis. Justification for this view is given in Reference 7.1-22.

7.1.4.1.3.1 Uncertainty in Frequencies

Probability distributions on the frequencies of the source terms contributing to the various results were constructed. For accident sequences originating from internal and seismic initiating events, distributions were obtained by propagating uncertainties on input parameters to the fault tree and event tree analyses through the algebraic expressions for accident class frequencies in terms of those parameters, using Monte Carlo methods. The distributions on the input parameters were assigned in a manner that follows currently accepted practice as described, for example, in Reference 7.1-23. For initiating events originating from fires in the plant, the probability distribution on accident class frequency was constructed on the basis of a sensitivity analysis of the more important assumptions and parameters. They are discussed in detail and documented in Reference 7.1-22.

7.1.4.1.3.2 Uncertainty in Source Terms

One of the greatest sources of uncertainty in the CCDFs is the magnitude of the source terms. Sensitivity studies have been carried out to determine the effect of a range of source term magnitudes and times of release for: (a) VR and VRH20; (b) C4γ, C4γ' and C4γ" (both seismic and internal); (c) OPREL (latent effects only); and (d) RB. These source terms were chosen because, on the basis of runs of CRAC2 carried out with the source terms and point estimate frequencies given in Table 7.1-23, it was established that they represent the major contributors to public risk. Details of these sensitivity studies and their effect on the CCDFs are provided in Reference 7.1-22.

7.1.4.1.3.3 Uncertainty in Evacuation and Sheltering

The CCDF for early fatalities is particularly sensitive to the choice of evacuation delay time (Ref. 7.1-23). Sensitivity studies were carried out in which they delay time was varied from 1 to 5 hours. The evacuation velocity was varied from 2.5 to 10 mph. For seismically initiated sequences, it was assumed for the sensitivity study that evacuation assumptions would be unaffected.

The 10 to 25 mile sheltering assumptions were changed to simulate sheltering in basements for 24 hours, followed by rapid relocation. In addition, the outer 25 mile radius was changed to 50 miles.

The effect that these variations have on CCDFs is described in Reference 7.1-22.

7.1.4.1.3.4 Uncertainty in Health Effects Modeling

For early fatalities, Reference 7.1-20 provides dose-response relationships for minimal, supportive, and heroic medical treatment. In the sensitivity analysis, each of these was chosen in turn. The standard dose-response relationship used for latent cancers in CRAC2, the central estimate, was varied to allow the simple linear dose-response relationship. The effect that these variations have on the CCDFs is described in Reference 7.1-22.

7.1.4.2 Analysis

The first step in the analysis was to use the point estimate source terms and point estimate frequencies in Tables 7.1-22 and 7.1-23, respectively, in CRAC2 and to produce a single CCDF for each health or economic effect. This single CCDF is called "point estimate" because it is obtained using single or point estimates of each of the important input parameters. For each health or economic effect, the significant contributors to risk, determined by comparing the size of each contributor to the area under the point estimate CCDFs, were (a) VR and VRH20; (b) RB; (c) $C4\gamma$, $C4\gamma'$ and $C4\gamma''$; and (d) OPREL (latent effects only).

In the second step, an uncertainty analysis of the frequency of each source term was carried out as described in Section 7.1.4.1.3.1.

The third step was to establish a range of conditional CCDFs for each source term and each of the health or economic effects that are being considered. Upper and lower estimates on this range were taken as upper and lower percentiles on a lognormal distribution. The upper percentiles were chosen as the 95th or 99th, depending on how likely the estimates are expected to be, and the lower estimate was chosen to be the 5th percentile. This is sufficent to fix the two independent parameters in the lognormal distribution.

The fourth step was to use this lognormal distribution in combination with the uncertainty distribution on frequencies to given an overall uncertainty distribution on the CCDFs. The uncertainty distributions are presented in Reference 7.1-22.

The final step was to extract from the uncertainty distribution the medians that are presented in Section 7.1.4.3.

7.1.4.3 Results

The results of the analysis are given in Figures 7.1-2 to 7.1-7 and in Table 7.1-26. These results give the total contribution from all source terms for seismic, internal, and fire initiators. The CCDFs for individual source terms, as well as upper and lower estimates and point estimates, are given in Reference 7.1-22. All of the results presented here are median CCDFs.

7.1.4.3.1 CCDFs

Figure 7.1-2 contains the median CCDF for the number of people receiving a bone marrow dose in excess of 200 rems from early exposure. (Early exposure is confined to that portion of the radiation dose that is accumulated within 7 days, due to inhalation of radioactive materials, cloudshine and groundshine.) This level of dose roughly corresponds to a need for hospital treatment.

Figure 7.1-3 shows the median CCDF for the total population exposure in person-rems for the population out to 500 miles (that is, the probability per reactor year that the total population exposure will equal or exceed the values given). The figure also gives a similar CCDF for the population within 50 miles.

Figure 7.1-4 shows the median CCDF for acute fatalities, representing radiation injuries that would produce fatalities within about one year after exposure.

Figure 7.1-5 gives the median CCDFs for latent cancer fatalities. CCDFs for the total population and the population within 80 km (50 miles) are shown separately, and the latent cancers have been subdivided into that attributable to exposures of the thyroid and all other organs.

Figure 7.1-6 shows the CCDF for ex-plant costs in 1980 dollars. In general, these costs are dominated by decontamination of urban or agriculatural land. Additional economic costs include decontamination of the facility itself and the cost of replacement power. These impacts are discussed in Section 7.1.4.3.2.

7.1.4.3.2 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 broad, it is also useful to combine them to obtain average measures of environmental risk. Such averages can be particularly useful as an aid to the comparison of radiological risks associated with accidental releases, or those arising from other accidents.

A common way in which this combination of factors is used to estimate risk is to multiply the frequencies by the consequences. The resultant risk is then expressed as the number of consequence expected per unit time. Table 7.1-26 shows average values of risk associated with population dose, acute fatalities, latent fatalities, and costs for protective actions and decontamination. These average values are obtained by summing the frequency multiplied by the consequences over the entire range of the median CCDFs. They are equal to the areas under the corresponding CCDFs. Because the probabilities are on a per-reactor-year basis, the averages shown are also on a per-reactor-year basis.

The acute fatality risk of $4.1x10^{-5}$ deaths per reactor year at the median level may be put into perspective by noting that 60 fatalities from motor vheicle accidents, 24 from falls, 8 from burns, and 3 from firearms are likely to occur each year within 10 miles of the plant. These figures are based on U.S. averages.

The individual risk of acute fatality as a function of distance is displayed on Figure 7.1-7. The risk to the average individual living within one mile of the site boundary is $2.2x10^{-9}$ per reactor year. This risk is small. For comparison, the following risks of fatality per year to an individual living in the United States may be noted; $2.2x10^{-4}$ per year from automobile accidents and $1.2x10^{-5}$ per year from firearms.

The average population exposure is 70 person-rem per reactor year. This value may be compared with the annual average population exposures from routine operation given in Tables 5.2-15 and 5.2-17.

The average number of latent cancer fatalities (summing those due to thyroid dose and those in all other organs) within the population to 500 miles is 0.013 per reactor year. The equivalent average latent cancer fatalities for the population within 50 miles is 0.008 per reactor year. These figures may be put in perspective by noting that, in the population of 8,100,000 that is predicted to live within 50 miles of the Limerick reactor in the year 2000, there will be about 20,000 cancer fatalities per year from all causes. This figure was obtained by multiplying the figure for the population within 50 miles by $2.5x10^{-3}$, which, according to the Statistical Abstract of the United States, is the chance per year that an individual will die of cancer.

The ex-plant economic risk, in 1980 dollars, associated with the Limerick Generating Station is predicated to be $6,000 per reactor year at the median level. This figure is small compared with the estimated property damage caused by other accidents within 50 miles of the Limerick site (e.g., of the order of $10 million per year for automobile accidents. This figure is based on U.S. average statistics).

There are other economic impacts and risks that are not included in the calculations discussed above. These costs would be for decontamination and repair or replacement of the facility, and for replacement power. Experience with such costs is currently being accumulated as a result of the Three Mile Island accident.

It is already clear that such costs can equal or exceed the original capital cost. The cost for decontamination and restoration is in the region of $2 billion. Replacement power costs for two units at the Limerick site are estimated at $580 million per year. If it is assumed that both units on the site are out of operation for 8 years, the total cost of the accident would be $6.64 billion. The accident sequences considered in this report and shown in Table 7.1-22 would all lead to core melt and would in turn lead to costs of the size described above. The predicted median frequency of core melt is 3.0×10^{-5} per year so that the economic risk due to the accident sequences considered in this report is predicted to be $200,000 per year. This estimate is in 1980 dollars.

7.1.4.4 Conclusions

The previous sections consider the potential environmental impacts of severe accidents at the Limerick facility. These have covered a broad spectrum of hypothetical accidental releases and a range of possible health and economic impacts. The comparisons in the section on risk considerations show that the public risk associated with these impacts is small.

7.1.5 REFERENCES

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SOUR

CRAC 2

Input GROUP	T_r [2] (hr)	T_d [3] (hr)	T_w [4] (hr)	h[5] (m)	Q
OXRE	4.0	0.5	3.0	27	8
OPREL	7.0	2.0	6.0	27	8
C4γ	1.5	2.0	1.0	27	7
C4γ'	1.5	2.0	1.0	27	7
C4γ"	1.5	2.0	1.0	10	7
C123γ"	7.0	2.0	6.0	10	7
LEAK 1	7.0	2.0	6.0	27	7
LEAK 2	7.0	2.0	6.0	27	7
RB[8]	1.5	3.0	1.5	10	8
VR[9]	0.25	3.5	0.25	10	1
VRH2O[10]	0.34	0.65	0.34	10	2

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(1) The final CCDFs given in Figures 7
on the source term characteristics

(2) T_r = time of release

(3) T_d = duration or release

(4) T_w = warning time

(5) h = height of release

(6) Q = rate of release of energy

(7) 8.4(6) = 8.4×10^6

(8) Reactor building failure

(9) Vessel rupture without water in ve

(10) Vessel rupture with water in vess

TABLE 7.1-22

CE TERM CHARACTERISTICS - POINT ESTIMATE(1)

RADIONUCLIDE RELEASE FRACTIONS

(6)	XE	QI	I_2	Cs	Te	Sr	Ru	La
cal/sec)								
.4(6)(7)	1.0	3(-4)	0.20	0.06	0.50	0.007	0.40	1.0(-5)
.4(6)	1.0	3(-4)	0.11	0.09	0.016	0.01	3(-3)	3 (-4)
.0(4)	1.0	3(-4)	0.261	0.202	0.434	0.029	0.095	5.2(-3)
.0(4)	1.0	3(-4)	0.07	0.09	0.20	0.016	0.008	5.0(-3)
.0(4)	1.0	3(-4)	0.73	0.70	0.55	0.09	0.12	7.0(-3)
.0(4)	1.0	3(-4)	0.13	0.17	0.50	0.02	0.08	6.2(-3)
.0(4)	0.73	3(-4)	1.9(-2)	9.8(-3)	4.6(-2)	1.6(-3)	3.2(-3)	5.8(-4)
.0(4)	0.73	3(-4)	2.7(-3)	9.8(-5)	4.6(-4)	1.6(-5)	3.2(-5)	5.8(-6)
.4(6)	1.0	3(-4)	0.05	0.09	0.09	4.0(-3)	0.02	5.0(-3)
.4(4)	1.0	3(-4)	0.1	0.33	0.33	0.15	0.04	0.02
(6)	1.0	3(-4)	0.5	0.73	0.75	0.35	0.07	0.05

.1-2 through 7.2-6 are medians and are obtained from an uncertainty analysis

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CRAC 2		PO
INPUT GROUP	INTERNAL	
OXRE	4.4(-8)	
OPREL	7.0(-6)	
C4γ	6.4(-8)	
C4γ'	5.6(-8)	
C4γ"	6.4(-9)	
C123γ"	3.6(-7)	
LEAK 1	1.1(-6)	
LEAK 2	6.1(-6)	
RB	0	
VR	1.4(-8)	
VRH20	1.4(-8)	

TABLE 7.1-23

FREQUENCIES OF TABLE 7.1-23 SOURCE TERMS

INT ESTIMATE(YR^{-1})		MEDIAN(YR^{-1})		
SEISMIC	FIRE	INTERNAL	SEISMIC	FIRE
1.3(-8)	6.9(-8)	3.3(-8)	7.5(-10)	2.6(-8)
2.0(-6)	1.1(-5)	5.3(-6)	1.2(-7)	4.2(-6)
6.3(-8)	0	6.4(-8)	2.0(-9)	0
5.6(-8)	0	5.6(-8)	9.0(-10)	0
6.3(-9)	0	6.2(-9)	1.0(-10)	0
1.0(-7)	5.8(-7)	2.8(-7)	6.3(-9)	2.2(-7)
3.3(-7)	1.8(-6)	8.8(-7)	2.0(-8)	6.8(-7)
1.7(-6)	9.9(-6)	4.6(-6)	1.1(-7)	3.7(-6)
1.2(-6)	0	0	7.6(-9)	0
3.7(-7)	0	5.0(-9)	<1(-10)	0
4.1(-8)	0	5.0(-9)	<1(-10)	0

TABLE 7.1-24

ACTIVITY IN THE LIMERICK REACTOR

CORE AT 3293 MWt

Group/radionuclide	Radioactive inventory (million of Curies)	Half-life (days)
<u>NOBLE GASES</u>		
Krypton-85	0.57	3,950
Krypton-85m	28	0.183
Krypton-87	55	0.0528
Krypton-88	77	0.117
Xenon-133	184	5.28
Xenon-135	34	0.384
<u>IODINES</u>		
Iodine-131	83	8.05
Iodine-132	128	0.0958
Iodine-133	183	0.875
Iodine-134	202	0.0366
Iodine-135	172	0.280
<u>ALKALI METALS</u>		
Rubidium-86	0.061	18.7
Cesium-134	5.7	750
Cesium-136	1.9	13.0
Cesium-137	5.6	11,000
<u>TELLURIUM-ANTIMONY</u>		
Tellurium-127	5.8	0.391
Tellurium-127m	0.79	109
Tellurium-129	21.8	0.048
Tellurium-129m	5.8	34.0
Tellurium-131m	11.4	1.25
Tellurium-132	122	3.25
Antimony-127	6.0	3.88
Antimony-129	23.2	0.179
<u>AKALINE EARTHS</u>		
Strontium-89	102	52.1
Strontium-90	4.8	10,300
Strontium-91	130	0.403
Barium-140	163	12.8

TABLE 7.1-24 (Cont'd) (Page 2 of 2)

Group/radionuclide	Radioactive inventory (million of Curies)	Half-life (days)
<u>COBALT AND NOBLE METALS</u>		
Cobalt-58	0.0	71.0
Cobalt-60	0.0	1,920
Molybdenum-99	166	2.80
Technitium-99m	143	0.25
Ruthenium-103	114	39.5
Ruthenium-105	67	0.185
Ruthenium-106	42	366
Rhodium-105	60	1.5
<u>RARE EARTHS, REFRACTORY OXIDES AND TRANSURANICS</u>		
Yttrium-90	504	2.67
Yttrium-91	127	59.0
Zirconium-95	152	65.2
Zirconium-97	156	0.71
Niobium-95	145	35.0
Lanthanum-140	166	1.67
Cerium-141	151	32.3
Cerium-143	148	1.38
Cerium-144	90	284
Praseodymium-143	147	13.7
Neodymium-147	61	11.1
Neptunium-239	1,670	2.35
Plutonium-238	0.036	32,500
Plutonium-239	0.02	8.9×10^6
Plutonium-240	0.024	2.5×10^6
Plutonium-241	5.5	5.350
Americium-241	0.0034	1.6×10^5
Curium-242	1.1	163
Curium-244	0.013	6,630

Note: The above grouping of radionuclides corresponds to that in the Reactor Safety Study

PE

Sector (miles)	N	NNE	NE	ENE
0-0.5	0	0	0	0
0.5-1.0	61	141	27	32
1.0-1.5	108	110	40	58
1.5-2.0	432	153	55	33
2.0-2.5	243	151	128	159
2.5-3.0	297	184	157	194
3.0-3.5	316	214	191	218
3.5-4.0	365	246	220	252
4.0-4.5	472	92	187	109
4.5-5.0	527	102	210	121
5-6	1,306	585	559	345
6-7	1,544	691	660	407
7-8.5	2,761	1,236	1,181	729
8.5-10	3,295	1,476	1,410	870
10-12.5	1,280	4,739	6,146	9,828
12.5-15	1,565	5,792	7,512	12,012
15-17.5	1,850	6,845	8,877	14,197
17.5-20	2,134	7,897	10,243	16,381
20-25	20,829	97,040	10,711	27,827
25-30	25,457	118,604	13,091	34,010
30-35	21,716	85,094	14,733	11,780
35-40	25,057	98,186	16,999	13,592
40-45	11,888	17,743	24,911	18,800
45-50	13,286	19,831	27,841	21,011
50-55	6,586	32,970	30,854	53,592
55-60	17,057	16,913	64,100	105,293
60-65	30,623	17,742	66,292	171,163
65-70	35,151	15,206	62,170	272,858
70-85	155,810	36,828	296,821	2,001,226
85-100	114,867	53,596	456,449	6,070,038
100-150	271,093	258,729	1,244,443	5,114,585
150-200	482,802	568,895	1,353,835	1,802,514
200-350	1,650,580	1,194,147	3,569,922	4,813,485
350-500	818,581	5,136,991	949,375	0

TABLE 7.1-25

RMANENT RESIDENT POPULATION FOR THE LIMERICK SITE

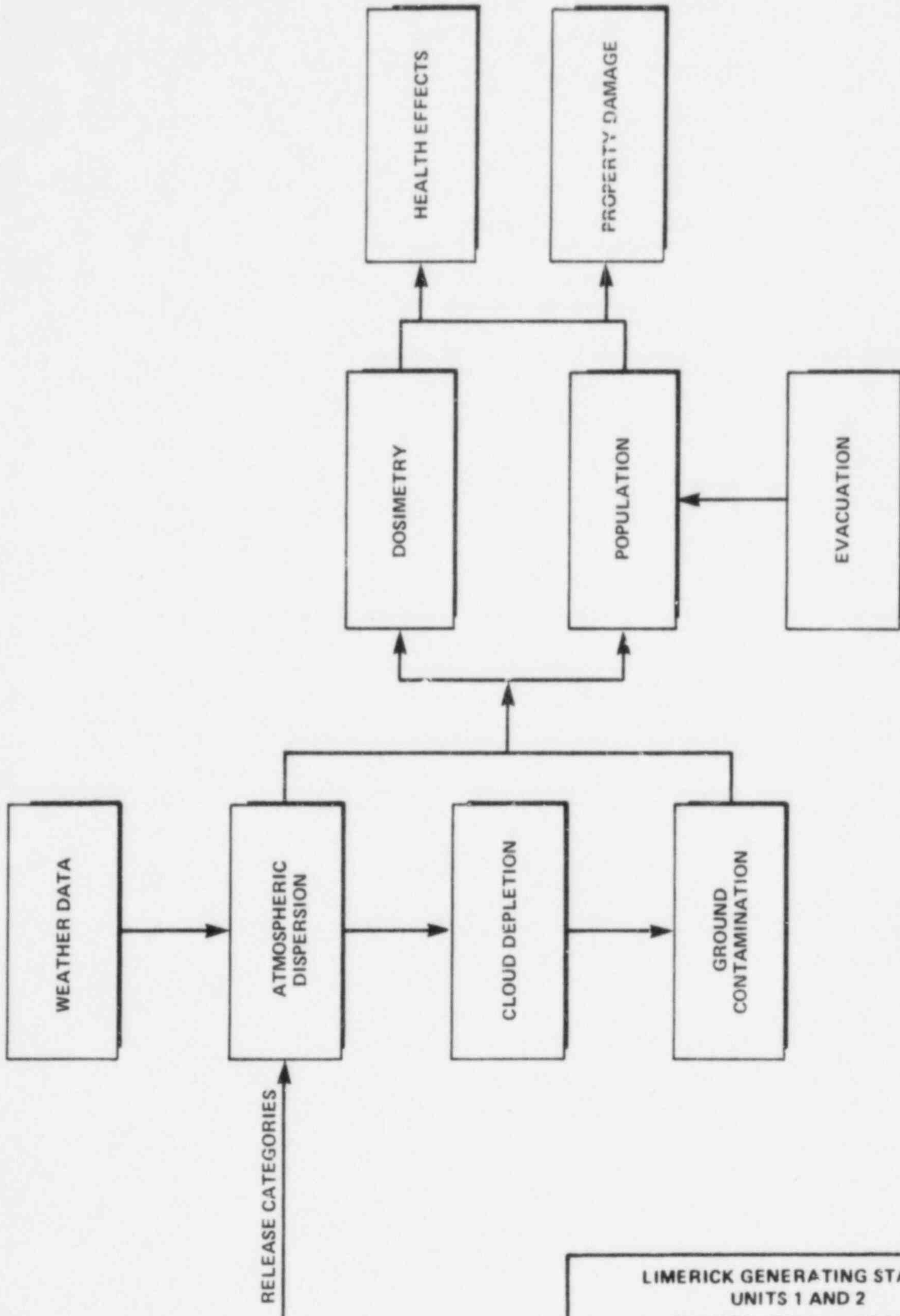
Direction											
E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
0	0	0	0	0	0	0	0	0	0	0	0
23	73	0	18	5	0	88	67	65	10	24	11
60	70	222	206	259	305	123	136	33	45	199	317
84	97	311	286	362	427	173	190	46	63	278	444
183	192	675	949	175	207	117	339	715	2,083	1,537	565
223	234	826	1,160	213	254	142	414	874	2,546	1,878	690
281	172	2,622	2,669	50	232	208	293	740	7,310	4,029	572
325	199	3,025	3,079	57	268	239	339	853	8,435	4,648	661
227	198	745	1,126	253	168	200	713	1,197	2,232	960	434
253	221	833	1,258	283	187	223	796	1,337	2,494	1,073	486
2,248	2,692	598	5,913	944	472	745	261	60	1,724	164	1,032
2,657	3,182	707	6,989	1,115	558	880	309	70	2,038	193	1,219
4,751	5,691	1,264	12,499	1,995	998	1,574	552	126	3,645	346	2,181
5,671	6,792	1,508	14,918	2,381	1,191	1,879	659	150	4,350	413	2,603
12,472	31,605	21,922	7,194	17,907	8,376	1,211	2,068	737	24,907	1,578	1,986
15,243	38,629	26,794	8,792	21,887	10,237	1,481	2,528	901	30,442	1,928	2,428
18,014	45,652	31,666	10,391	25,866	12,098	1,750	2,987	1,065	35,976	2,279	2,869
20,786	52,675	36,537	11,990	29,846	13,960	2,019	3,447	1,229	41,511	2,629	3,310
63,046	336,450	563,411	121,367	17,609	17,078	23,839	10,670	8,012	34,626	8,212	7,096
77,056	411,217	688,613	148,337	21,523	20,873	29,137	13,041	9,793	42,320	10,037	8,673
122,464	324,681	336,351	16,314	202,552	24,450	6,281	34,785	23,142	9,433	6,615	2,663
141,305	374,632	388,097	18,823	233,714	28,212	7,247	40,136	26,703	10,884	7,632	3,072
225,218	49,936	67,649	13,997	11,762	32,128	9,777	71,801	37,361	12,542	24,250	13,994
251,715	55,811	75,607	15,643	13,146	35,907	10,927	80,248	41,756	14,017	27,103	15,640
187,792	49,511	161,447	30,528	53,899	8,247	27,223	34,766	32,841	22,307	20,197	22,346
174,828	59,913	102,131	39,055	68,362	9,516	42,384	36,859	44,757	31,575	23,085	45,025
162,803	72,760	47,391	34,900	53,574	9,185	50,422	43,002	49,577	45,230	19,869	47,911
160,844	78,672	55,195	32,108	18,030	5,570	40,968	58,190	46,595	42,210	21,365	39,144
351,491	177,523	128,551	69,479	57,227	22,168	730,546	206,022	116,878	74,437	72,049	141,117
0	0	0	27,737	61,571	43,637	942,506	101,937	110,012	41,030	59,928	20,755
0	0	0	8,231	209,523	362,873	2,739,529	1,062,112	238,115	295,032	140,775	182,565
0	0	0	0	52,287	166,772	329,908	287,951	520,317	164,714	142,422	324,640
0	0	0	0	542,893	3,071,062	1,879,393	1,030,760	4,504,704	5,425,319	6,677,693	2,105,064
0	0	0	0	31,959	2,036,392	4,558,303	2,849,465	6,044,539	9,035,347	504,911	306,549

TABLE 7.1-26

AVERAGE VALUES OF ENVIRONMENTAL RISKS

DUE TO ACCIDENTS PER REACTOR-YEAR

Environmental Risk	Average/RY (Median)
Population exposure	
Person-rems within 50 miles	40
Total person-rems	70
Acute fatalities	4.1×10^{-5}
Latent cancer fatalities	
All organs excluding thyroid	0.012
Thyroid only	0.001
Cost of protective actions and decontamination	$6,000

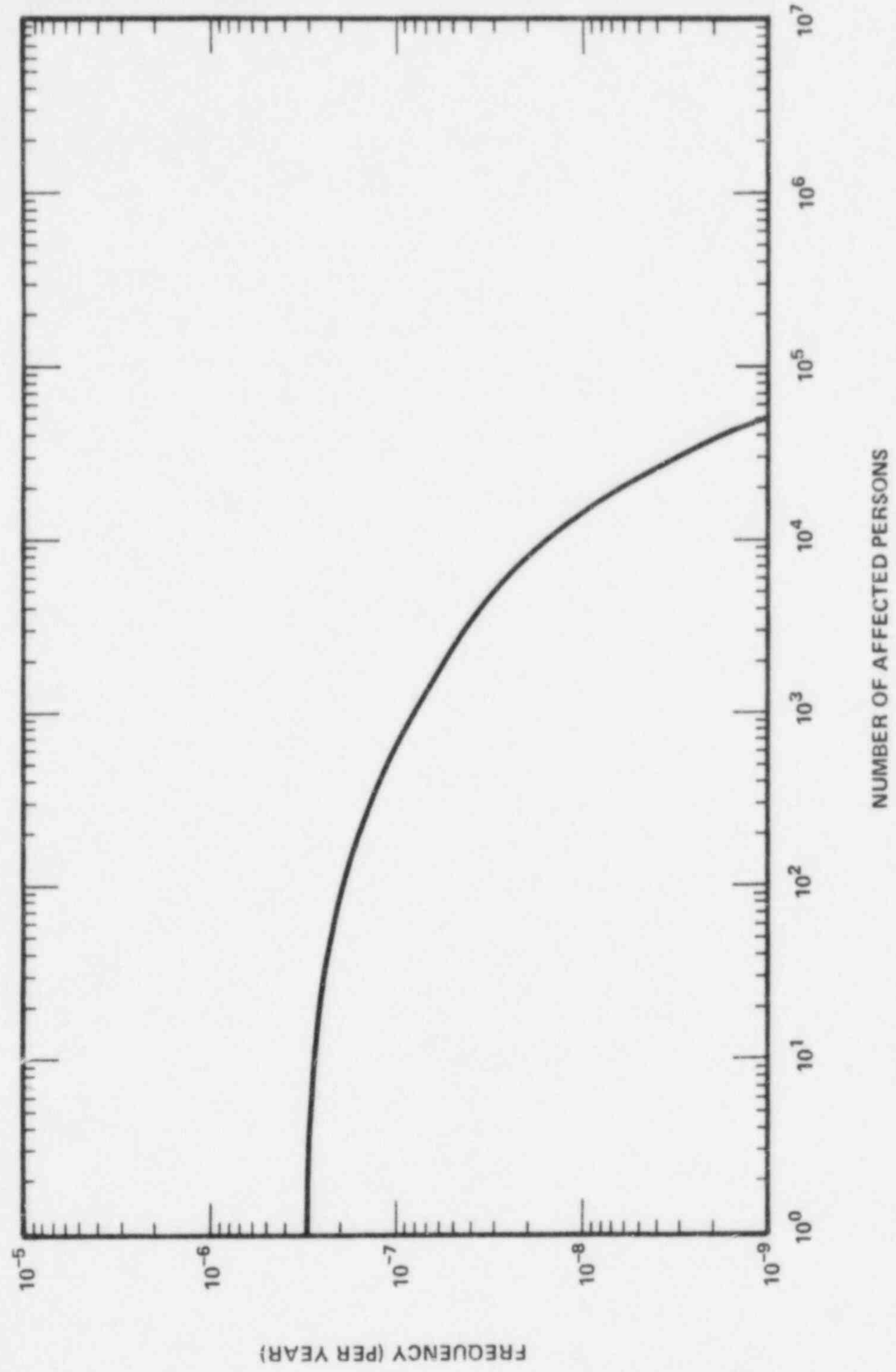


LIMERICK GENERATING STATION
UNITS 1 AND 2
ENVIRONMENTAL REPORT

SCHEMATIC OUTLINE OF CONSEQUENCE MODEL

FIGURE 7.1-1

REV. 12, 04/83



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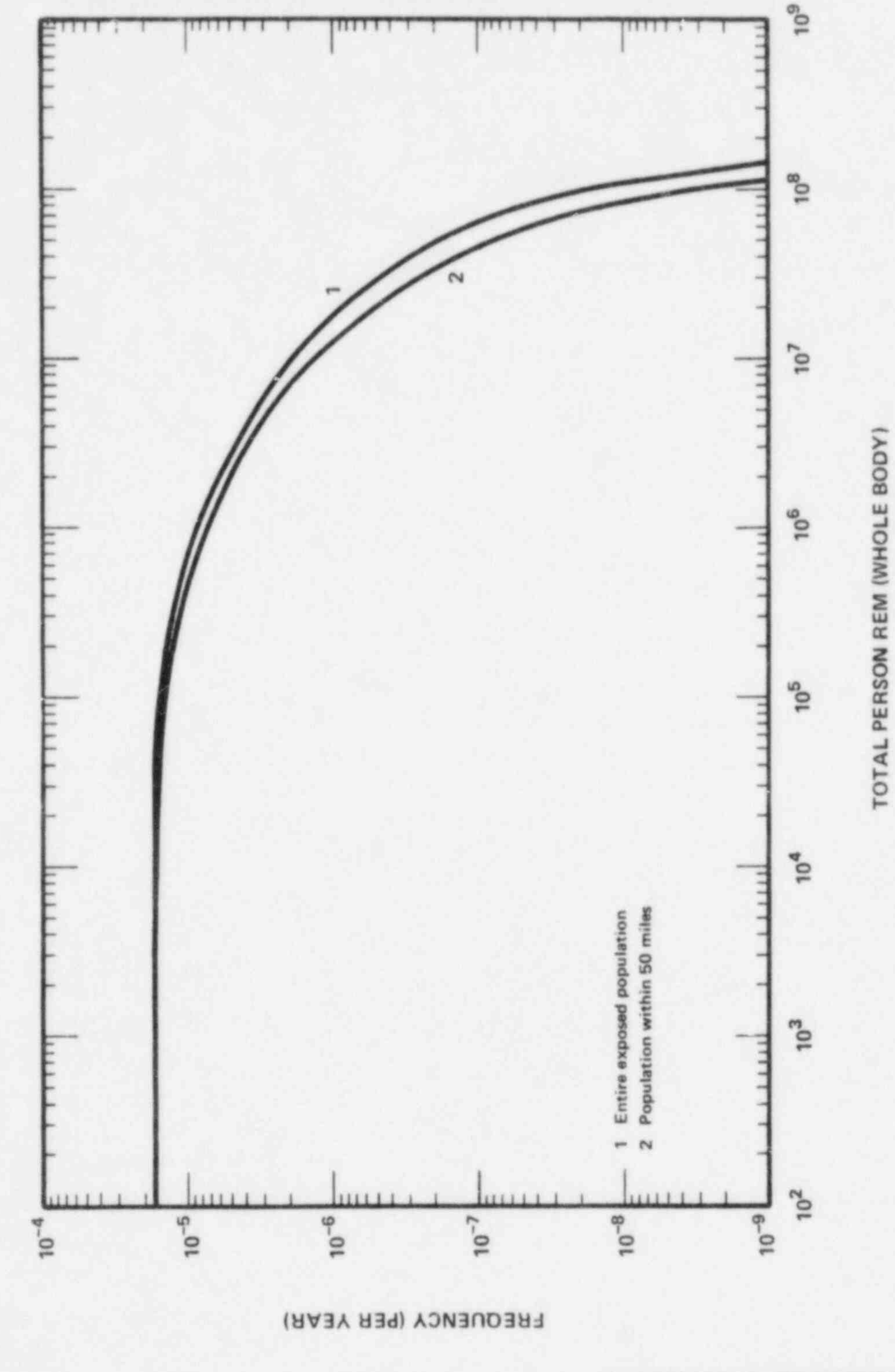
UNITS 1 AND 2
ENVIRONMENTAL REPORT

MEDIAN CCDF OF
BONE MARROW DOSE
GREATER THAN 200 REM

FIGURE 7.1-2

REV. 12, 04/83

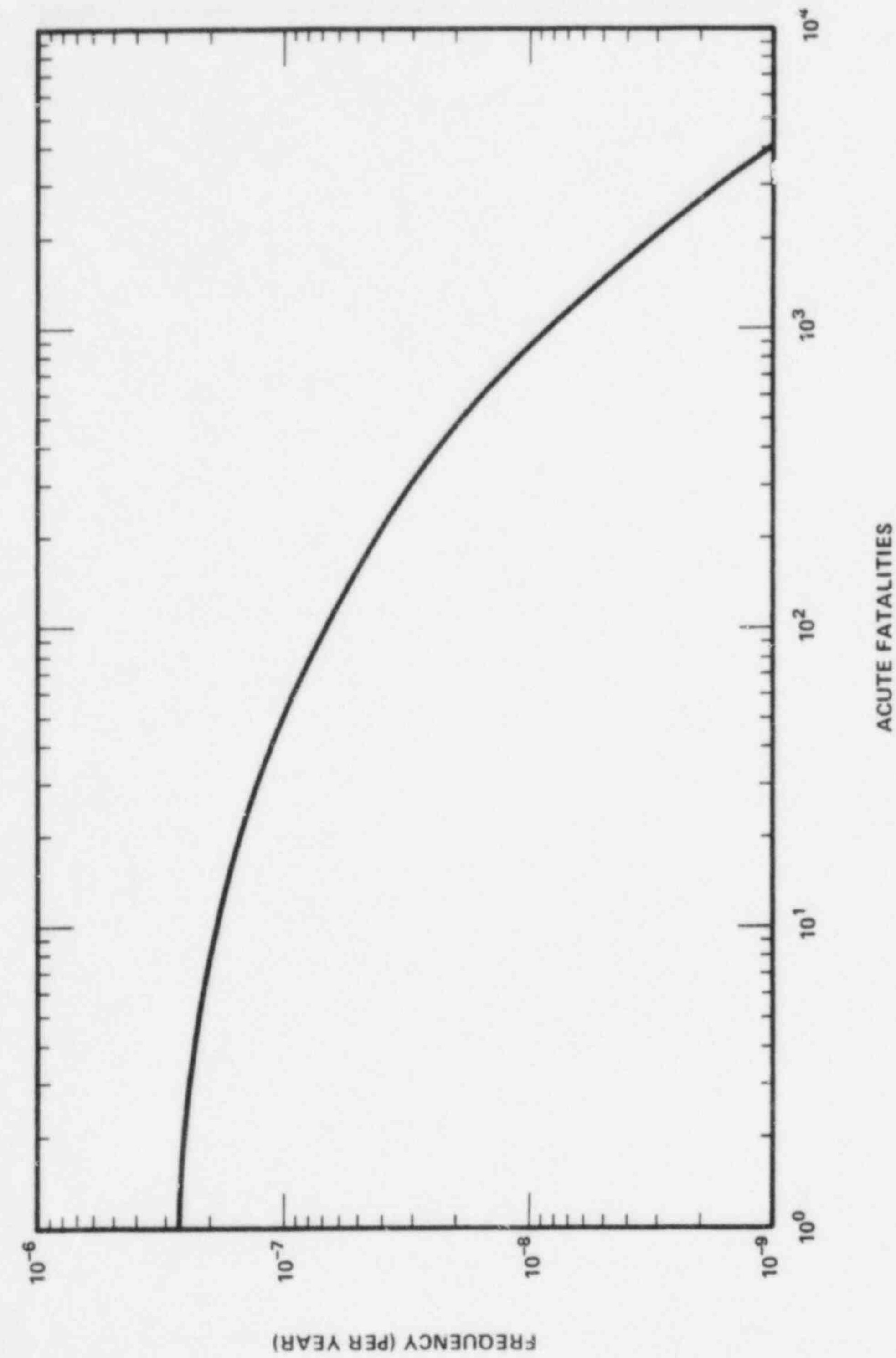


LIMERICK GENERATING STATION
UNITS 1 AND 2
ENVIRONMENTAL REPORT

MEDIAN CCDF OF
POPULATION EXPOSURES

FIGURE 7.1-3

REV. 12, 04/83



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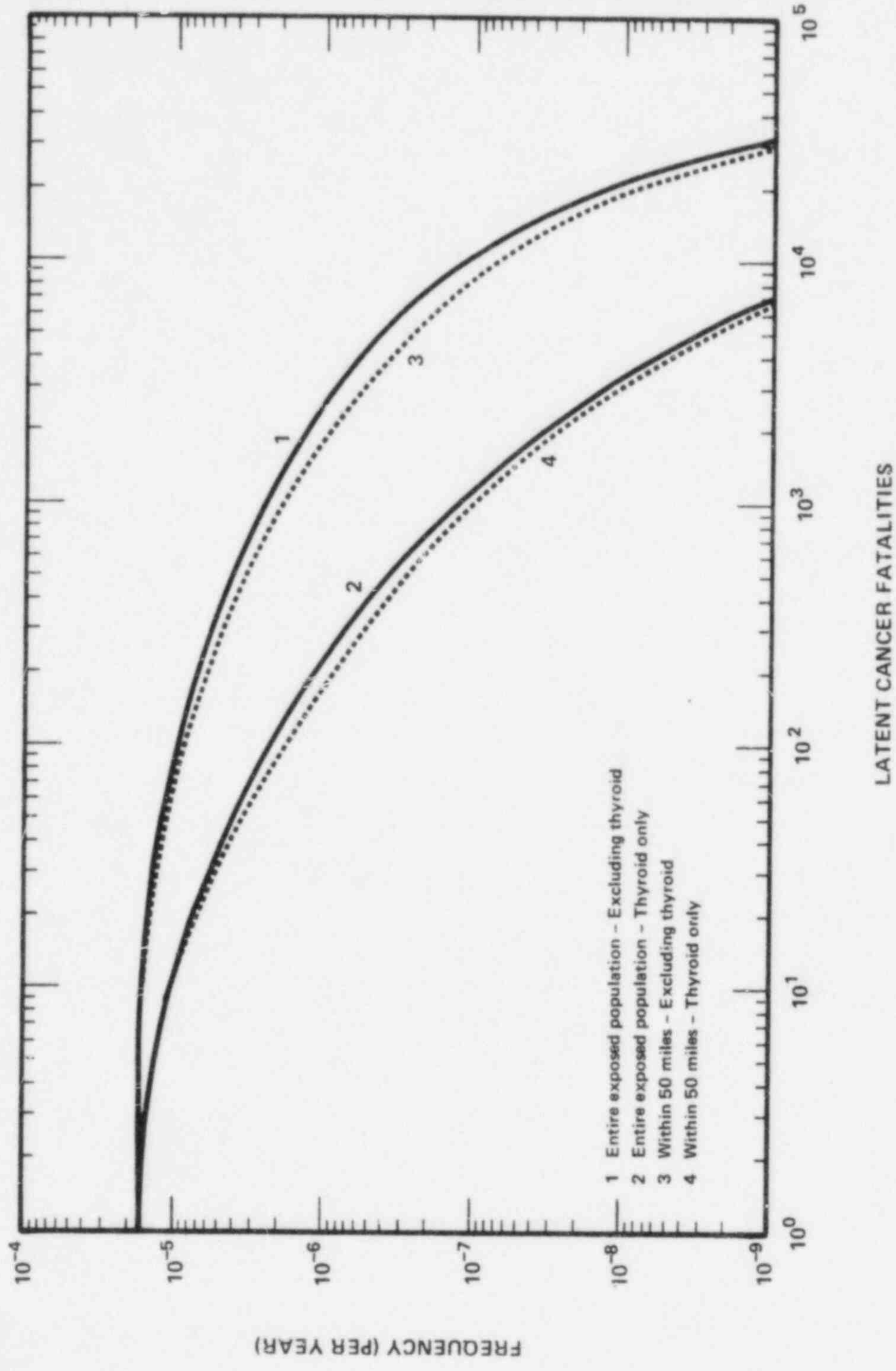
UNITS 1 AND 2
ENVIRONMENTAL REPORT

MEDIAN CCDF OF
ACUTE FATALITIES

FIGURE 7.1-4

REV. 12, 04/83



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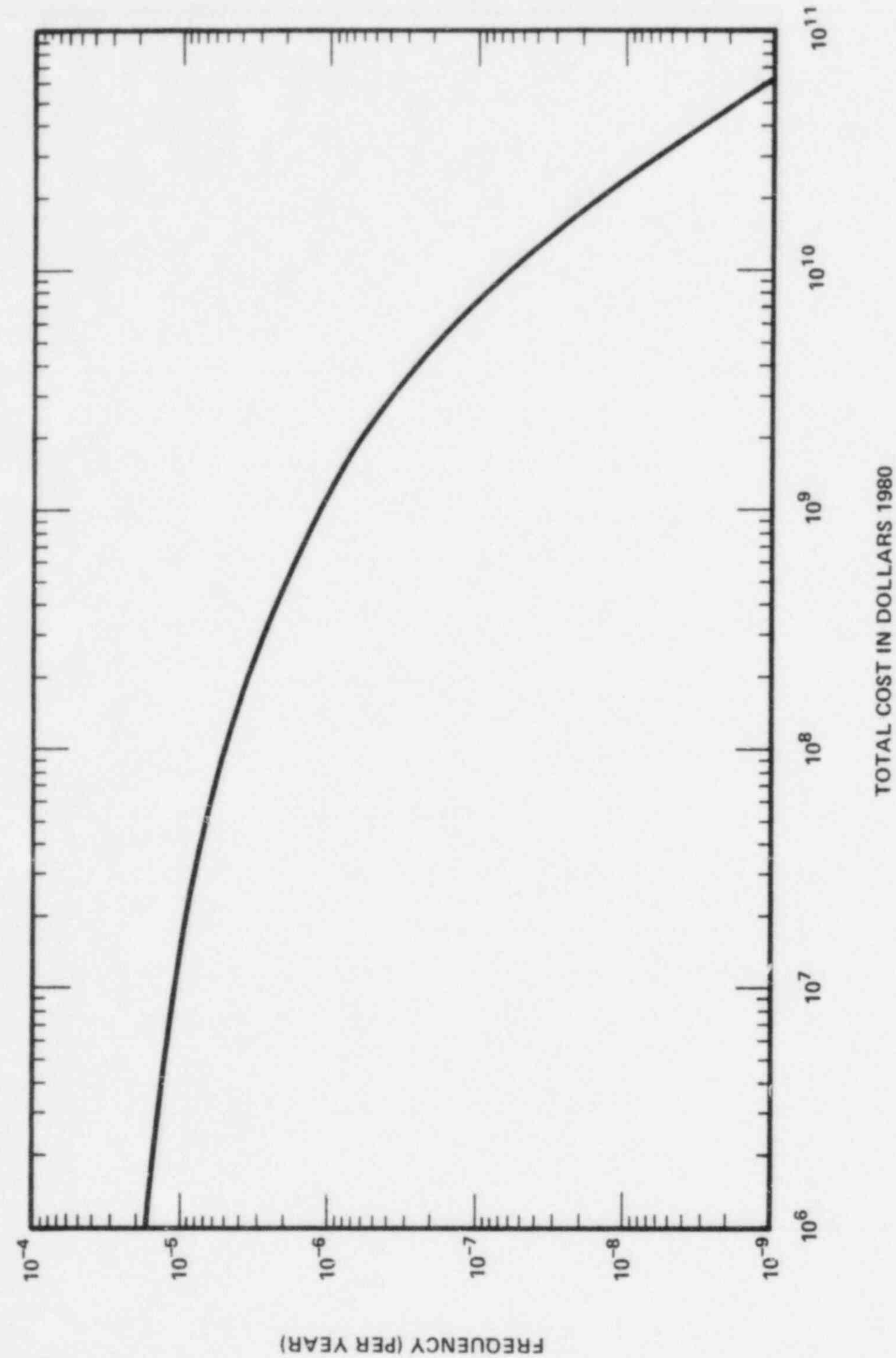
UNITS 1 AND 2
ENVIRONMENTAL REPORT

MEDIAN CCDF OF LATENT CANCER FATALITIES

FIGURE 7.1-5 REV. 12, 04/83



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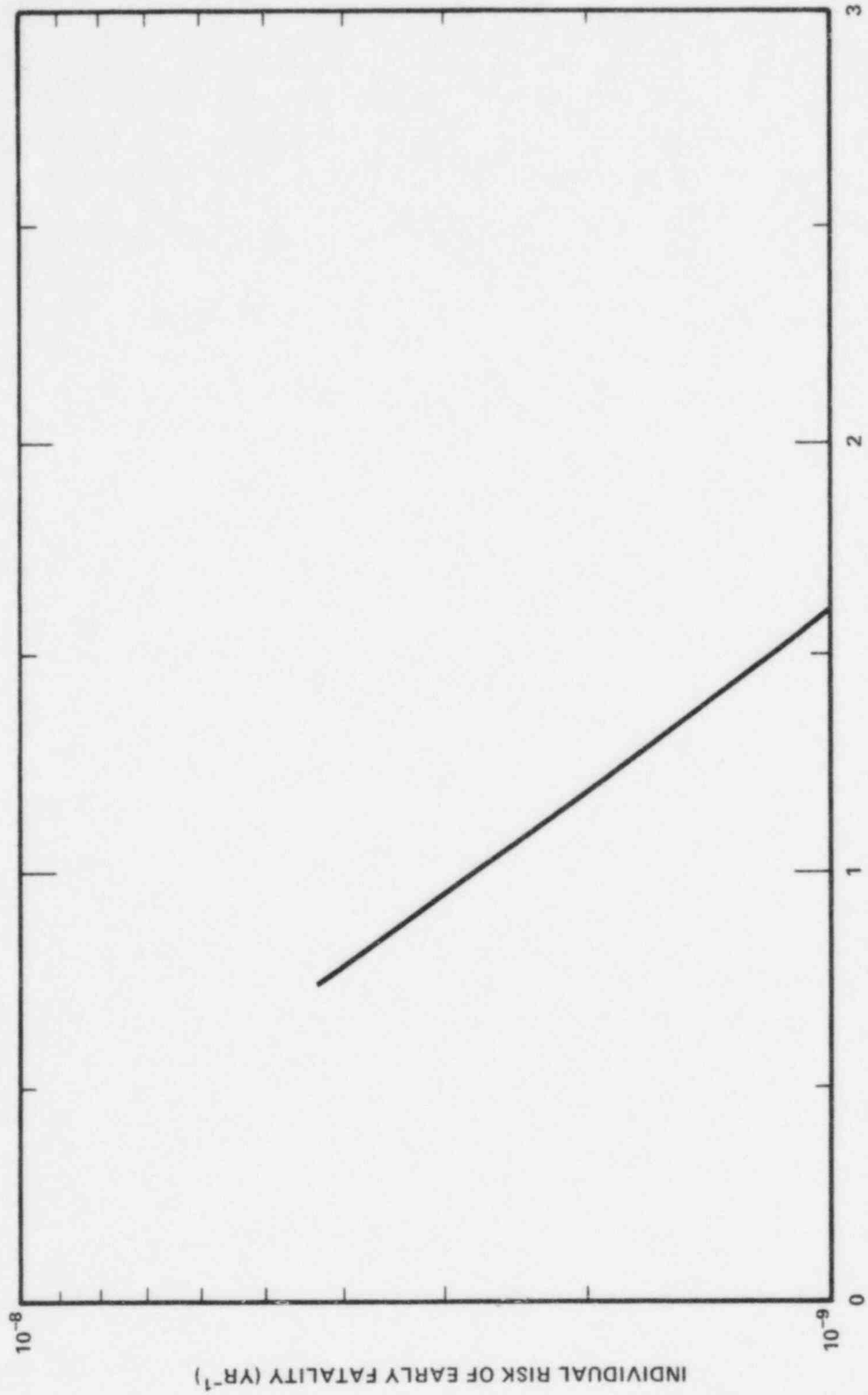
UNITS 1 AND 2
ENVIRONMENTAL REPORT

MEDIAN CCDF OF
EX - PLANT COSTS

FIGURE 7.1-6

REV. 12, 04/83



LIMERICK GENERATING STATION
UNITS 1 AND 2
ENVIRONMENTAL REPORT

MEDIAN INDIVIDUAL RISK OF EARLY FATALITY AS A FUNCTION OF DISTANCE

FIGURE 7.1-7 REV. 12, 04/83

QUESTION E290.16

Provide a figure of the site and immediate vicinity showing salt drift deposition isopleths (lbs/Ac/yr) when both natural draft cooling towers are operating. Provide detailed information on the model used to predict the deposition including information on verification of the model.

RESPONSE

A plot of annual average salt deposition isopleths in lb/acre predicted from the operation of the two natural draft cooling towers is shown in Figure E290.16-1. The predicted maximum deposition is 6.8 lb/acre/yr at a distance of 0.5 miles ESE of the plant.

The model used to calculate the isopleths for drift deposition from the two Limerick natural draft cooling towers is known as the Hosler-Pena-Pena (HPP) model. This model employs a ballistic approach to determine drift droplet trajectory, and employs the formulas of Fletcher (Ref E290.16-1) to determine droplet evaporation. The equations from the HPP model have been simplified into a series of nomograms to facilitate the calculational procedure. A complete description of the model, including the nomograms, has been given in a paper by Hosler, Pena and Pena (Ref. 290.16-2).

An extensive review of the state of the art in drift deposition modeling has been conducted by Argonne National Laboratory under sponsorship of the NRC (Ref. E290.16-3) and EPRI (Ref. E290.16-4). This review examined the theoretical foundations and assumptions upon which the various drift deposition models are based, and attempted to validate the models against the available field data.

The Argonne review necessitated computerizing the HPP model to allow easier comparison with other models, as well as field data. In the process of developing the HPP computer algorithm, improvements were made in the scheme employed to integrate the Fletcher evaporation equations, which resulted in a 30 percent increase in the accuracy of this portion of the model. These changes did not affect any of the basic model assumptions, but when extrapolating the Argonne validation results to the Limerick site, it should be recognized that the EROL calculations were based upon the earlier nomogram version of the HPP model, not the Argonne computerized version.

The only high quality data set available to validate drift deposition calculations from natural draft cooling towers was

obtained by the state of Maryland during the Chalk Point Cooling Tower Program. This data base consists of field measurements of drift deposition obtained during 1975 (Ref E290.16-5) and 1976 (Ref E290.16-6), as well as during the Chalk Point Dye Tracer experiment of 1977 (Ref E290.16-7). All of these data are limited to ground level deposition measurements at or within 1 km of the Chalk Point tower. The more refined dye tracer experiment was conducted on only one day, during conditions of high humidity, moderately high wind speeds, and stable atmospheric conditions.

The HPP model was one of three models found by Argonne to compare most favorably with the field data from Chalk Point, predicting drift deposition values that were "generally within the error bounds of the data." The HPP model was found to underpredict by factors of 2.5 at 0.5 km and 3.6 at 1.0 km during the dye tracer study. This is consistent with the data obtained during 1975 and 1976, which showed that the HPP model had a slight tendency to underpredict. Because the maximum predicted salt deposition rate of 6.8 lb/acre/year from the Limerick towers is only 50 percent of the normal background salt deposition, an underprediction of this magnitude is acceptable, and is within the accuracy limitations of the current state of the art of drift modeling.

References

E290.16-1 Fletcher N.H.: The Physics of Rainclouds, Cambridge University Press, Cambridge, Mass., 1966.

E290.16-2 Hosler C., Pena J. and Pena R.: Determination of Salt Deposition Rates from Drift from Evaporative Cooling Tower. Trans. ASME (Amer. Soc. Mech. Eng.), Ser. A, J. Eng. Power, 96(3): 283 (1974).

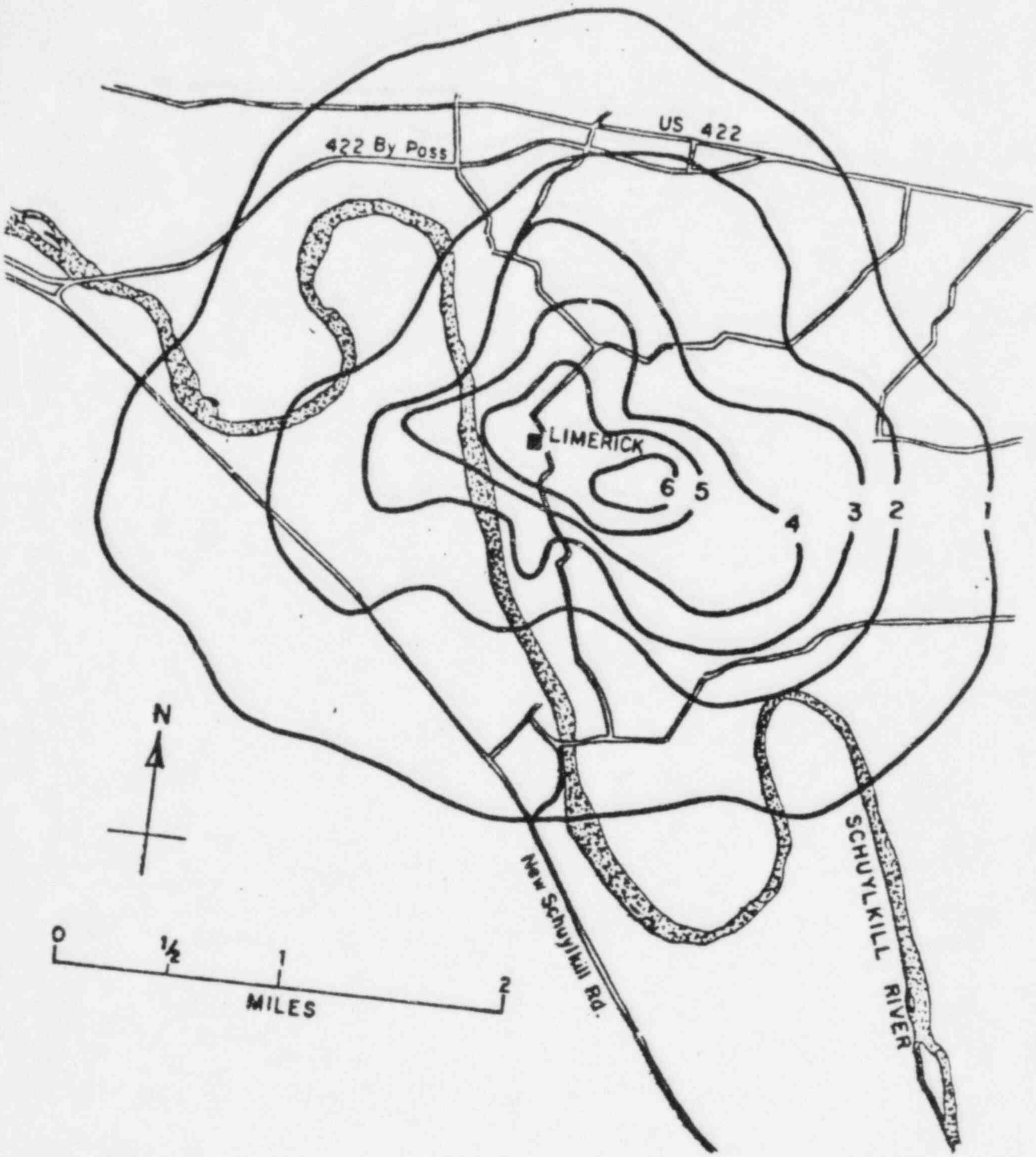
E290.16-3 Policastro A.J., Dunn W.E., Breig M.L. and Ziebarth J.P.; Evaluation of Mathematical Models for Characterizing Plume Behavior from Cooling Towers - Salt Deposition from Natural Draft Cooling Towers, NUREG/CR-1581, Vol. 2, September 1980.

E290.16-4 EPRI, Studies on Mathematical Models for Characterizing Plume and Drift Behavior from Cooling Towers, EPRI CS-1963, Vol. 1-5, January 1981.

E290.16-5 Chalk Point Cooling Tower Project, Environmental Systems Corporation's Comprehensive Status Report for the Period July 1, 1974 - October 1, 1975. Volume II. PPSP-CPCTP-9. Environmental Projects Division. Environmental Systems Corporation. Knoxville, Tennessee, May 1976.

E290.16-6 Chalk Point Cooling Tower Project, Environmental Systems Corporation's Comprehensive Project Final Report for the Period October 1, 1975 - June 30, 1976. Volumes 1 and 2, PPSP-CPCTP-12. Engineering Projects Division. Environmental Systems Corporation. Knoxville Tennessee. October 1976.

E290.16-7. Chalk Point Cooling Tower Project, Cooling Tower Drift Dye Tracer Experiment, June 16 and 17, 1977, PPSP-CPCTP-16, Vol. 2, by John Hopkins University, August 1977.



LIMERICK GENERATING STATION
UNITS 1 AND 2
ENVIRONMENTAL REPORT

ANNUAL NaC1 DEPOSITION RATE FROM COOLING TOWER OPERATION (lb/acre)

FIGURE E 290.16-1

REV. 12, 04/83

c. The transmission lines are designed so that under fair conditions audible noise is imperceptible at 230kV and only slightly perceptible at 500kV near the tower locations. During heavy rains, the 500kV design only produces 54 dB at 50 ft from the conductor (Section 3.9.3.3).

A copy of the archeological survey done by John Milner Associates on the transmission line corridors (Limerick-Cromby, Cromby-Plymouth Meeting, Cromby-North Wales, Limerick-Whitpain) and copies of the following correspondence between PECo and the State Historical Preservation Officer on this topic were provided to the NRC by letter from E. J. Bradley to A. Schwencer dated February 25, 1983.

Brenda Barrett to George N. DeCowsky, dated January 26, 1982

Greg Ramsey to Philadelphia Electric Company, dated September 3, 1982

George N. DeCowsky to Greg Ramsey, dated September 17, 1982

Greg Ramsey to Harry Bechtel, dated September 27, 1982

Greg Ramsey to George N. DeCowsky, dated November 16, 1982

Donald S. Frieman to Greg Ramsey, dated December 1, 1982

Greg Ramsey to Donald S. Frieman, dated December 8, 1982

In addition, the following correspondence was provided to the NRC by letter from E. J. Bradley to A. Schwencer dated April 4, 1983.

Greg Ramsey to John Milner Associates, dated February 7, 1983

Donald S. Frieman to Greg Ramsey, dated February 16, 1983

Donald S. Frieman to Greg Ramsey, dated February 22, 1983

Donna Williams to Donald S. Frieman, dated March 21, 1983

John Milner Associates to Donna Williams, dated March 30, 1983, transmitting the following report which addresses the concerns cited in the above correspondence.

"Norristown Design Changes and Chester County Potential Visual Effect Evaluation, A Report Supplementary to: An Investigation of Potential Visual Effects Upon Previously Recorded Historic Sites in the Vicinity of Proposed Limerick Transmission Lines, Montgomery and Chester Counties, Pennsylvania" by John Milner Associates, Inc. dated March, 1983. (The original Milner Historic Site Report, dated 1982, is also attached.)

With regard to the Point Pleasant Pumping Station, an archeological survey was conducted in 1978 by Edward M. Schortman and Patricia A. Urban. Their report entitled, "A Survey of Cultural Resources in the Area of the Proposed Point Pleasant Pumping Facilities, Combined Transmission Main, Bradshaw Reservoir, North Branch Main and Perkiomen Main, Bucks County, Pennsylvania," was provided to the NRC by letter from E. J. Bradley to A. Schwencer dated February 25, 1983.

QUESTION E450.1

In accordance with NRC's Interim Policy (45FR40101) revise Section 7.1.1 to include a probabilistic evaluation of impacts of accidents including those formerly called Class 9 accidents.

RESPONSE

The requested evaluation is provided in Section 7.1.4.

QUESTION E450.2

Please provide your assessment of accidents formerly classified as classes 3-8.

RESPONSE

The assessment of accidents formerly classified as classes 3-8 is provided in Section 7.1.

QUESTION E450.3

Figures 7.1-1 and 7.1-2 (the two CCDFs) have been superseded by subsequent PRA revisions; and therefore are no longer valid. Please provide updated information.

RESPONSE

Updated information is provided in Section 7.1.4.

QUESTION E450.4

Please provide information on the following specific items you consider appropriate to your PRA which is now recognized as part of the ER-OL and bases therefore;

a. population distribution for the plant mid-life years;

b. site specific off-site emergency response parameters such as delay time before evacuation, evacuation speed, evacuation distance etc.;

c. site-specific land-use and economic data;

d. assumption of the availability of supportive medical treatment to highly exposed individuals to reduce early fatality;

e. other categories of consequences and risk such as:

   i. delayed cancer fatality within 50-mile

   ii. person rems)
) within the 50-mile
   iii. thyroid effects)
) and the entire regions
   iv. genetic defects)

   v. offsite and onsite property damage

   vi. risks to individuals as functions of distance from the reactor, or individual risks isopleths;

f. liquid pathway considerations; and

g. comparison of risks from accidents with those from plant operation.

RESPONSE

The requested information is provided in Section 7.1.4, with the exception of Item f which is provided in the response to Question E240.21.