NUREG-1074

Draft Environmental Statement
related to the operation of Hope Creek Generating Station

Docket No. 50-354

Public Service Electric and Gas Company
Atlantic City Electric Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

June 1984

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ABSTRACT

This Draft Environmental Statement contains an assessment of the environmental impact associated with the operation of the Hope Creek Generating Station pursuant to the National Environment Policy Act of 1969 (NEPA) and Title 10 of the Code of Federal Regulations, Part 51, as amended, of the Nuclear Regulatory Commission regulations. This statement examines the environmental impacts, environmental consequences and mitigating actions, and environmental and economic benefits and costs associated with station operation. Land use and terrestrial and aquatic ecological impacts will be small. No operational impacts to historic and archeological sites are anticipated. The effects of routine operations, energy transmission, and periodic maintenance of rights-of-way and transmission facilities should not jeopardize any populations of endangered or threatened species. No significant impacts are anticipated from normal operational releases of radioactivity. The risk of radiation exposure associated with accidental release of radioactivity is very low. Socioeconomic impacts of the project are anticipated to be minimal. The action called for is the issuance of an operating license for Hope Creek Generating Station, Unit 1.

Further information may be obtained from

Mr. David H. Wagner, Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
301 492-8525

SUMMARY AND CONCLUSIONS

This Draft Environmental Statement (DES) was prepared by the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation.

(1) This action is administrative.

(2) The proposed action is the issuance of an operating license to Public Service Electric & Gas Company for operation of the Hope Creek Generating Station (Docket Number 50-354), located on the Delaware River Estuary, in Lower Alloways Creek Township, Salem County, New Jersey.

The unit employs a General Electric boiling water reactor to produce a core thermal power of 3,293 (megawatts thermal) (MWt). A steam turbine generator will use this energy to produce a net electrical output of approximately 1,067 megawatts electric (MWe). The exhaust steam in this closed-cycle system will be condensed in the station condenser. The station condenser will dissipate excess heat to the atmosphere through a natural draft cooling tower.

(3) The information in this statement represents an assessment of the environmental impacts of station operation pursuant to the Commission's regulations as set forth in Title 10 of the Code of Federal Regulations, Part 51 (10 CFR 51), which implements the requirements of the National Environmental Policy Act of 1969 (NEPA). After receiving, in February 1970, an application to construct the facility and subsequent amendments thereto, the staff reviewed the impacts that would occur during construction and operation. That evaluation was issued as the Final Environmental Statement - Construction Permit phase (FES-CP) in February 1974. After this environmental review, a safety-review, and an evaluation by the Advisory Committee on Reactor Safeguards, the Nuclear Regulatory Commission issued Construction Permit No. CPPR-120 on November 4, 1984 for construction of the facility. The applicant submitted an application for an operating license by letter dated March 1, 1983. The NRC conducted a predocketing acceptance review and determined that sufficient information was available to start detailed environmental and safety reviews. The applicant's operating license application was docketed on June 29, 1983.

(4) The staff has reviewed the activities associated with the proposed operation of the facility and the potential impacts of such operation, both beneficial and adverse. The staff's conclusions are summarized as follows:

(a) In December 1981, Unit 2 of the proposed dual-unit facility was cancelled. This cancellation most notably resulted in the elimination of the Unit 2 reactor building and cooling tower. Elimination of the cold water bypass system and cancellation of Unit 2 resulted in a reduction in the amount of water withdrawn from the Delaware River for cooling purposes. (Sections 4.2.1 and 4.2.3)

(b) Consumptive surface water use by Hope Creek Unit 1 during periods of river flow below 85 m^3/s (3,000 ft^3/s), as measured at Trenton, New Jersey, is to be compensated for under a ruling of the Delaware River Basin Commission made after the FES-CP was issued. The applicant is participating in the development of a supplementary reservoir for this purpose. (Section 4.2.3.2)

(c) The Hope Creek site occupies 300 ha (741 acres)* on Artificial Island, an extension of the New Jersey mainland created by the deposition of dredge spoils. The agricultural quality of the soil is low. Agricultural activities and important wildlife habitats were absent before facility construction. (Sections 4.2.2 and 4.3.4)

(d) One offsite transmission line will connect Hope Creek with the existing grid. Evidence examined to date indicates that operation of this transmission line will have no effect on the health of humans, animals, and plants. (Sections 4.2.7 and 5.5.1.3)

(e) Operation of the facility will not have any adverse impact on any terrestrial or aquatic endangered or threatened species. (Sections 4.3.5 and 5.6.2)

(f) In the 16.7-km (10-mi) area surrounding the facility, there are a total of 56 properties listed on the National Register of Historic Places. Operation and maintenance of Hope Creek and associated facilities are not expected to affect any of these properties. (Section 4.3.6)

(g) The effect of the service water intake structure and the barge slip on the 100-year floodplain of the site is negligible. (Section 5.3.3)

(h) The impact of the cooling tower on climatic conditions such as fogging and icing will be negligible. (Sections 5.4.1 and 5.5.1.1)

(i) Operation of the emergency diesel generators and the auxiliary boilers will not significantly degrade air quality in the vicinity of the plant. Additionally, the applicant has committed to the New Jersey Department of Environmental Protection to operate no more than two of the three auxiliary boilers at one time. (Section 5.4.2)

(j) Salt drift from the natural draft cooling tower will not affect native vegetation or agricultural crops in the vicinity of the facility. (Section 5.5.1)

(k) Impacts on the surface water use and quality of surface water are expected to be negligible. (Section 5.3.2)

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*Throughout the text of this document, where applicable, values are presented in both metric and English units. For the most part, measurements and calculations were originally made in English units and subsequently converted to metric. The number of significant figures given in a metric conversion is not meant to imply greater or lesser accuracy than that implied in the original English value.

(l) Ecological impacts resulting from entrainment and impingement should be negligible. Total potential fishery production lost as a result of entrainment is conservatively estimated at 0.5% of the commercial fishery finfish catch within 0-80 km (0-50 mi). Additionally, impingement of commercially important weakfish and blue crabs is estimated at less than 0.5% of the commercial fishery for the species. (Section 5.5.2)

(m) Ecological impacts resulting from discharges of thermal and chemical effluents are expected to be very small. High tidal flow past the facility will dilute such effluents to levels at which organisms either would not experience stress from these effluents or could avoid the discharge areas if necessary. (Section 5.5.2)

(n) The risks to the general public from the exposure to radioactive effluents and the transportation of fuel and wastes from annual operation of the facility are very small fractions of the estimated normal incidence of cancer fatalities and genetic abnormalities. (Section 5.9.3.2)

(o) The risk to the public health and safety from exposure to radioactivity associated with the normal operation of the facility will be small. (Section 5.9.3.2)

(p) No measurable radiological impact on the populations of biota is expected as a result of routine operation of Hope Creek. (Section 5.9.3.3)

(q) Impacts of a postulated reactor accident could be severe, but the likelihood of occurrence is small, and the risks are comparable to those at other nuclear power plants. (Section 5.9.4.6)

(r) The environmental impact of Hope Creek on the U.S. population from radioactive gaseous and liquid releases resulting from the uranium fuel cycle is very small when compared with the impact of natural background radiation. (Section 5.10)

(s) Radiation doses to the public as a result of end-of-life decommissioning activities are expected to be small. (Section 5.11)

(t) Area residents will not be affected by noise resulting from station operation; however, a potential impact to the public may be the periodic testing of the early notification system. This impact will be infrequent. (Section 5.12)

(5) This statement assesses various impacts associated with the operation of the facility in terms of annual impacts and balances these impacts against the anticipated annual energy production benefits. Thus, the overall assessment and conclusion would not be dependent on specific operating life. Where appropriate, a specific operating life of 40 years has been assumed.

(6) The Draft Environmental Statement will be made available to the public, to the Environmental Protection Agency, and to other agencies as specified in Section 8.

(7) The personnel who anticipated in the preparation of this document are identified in Section 7.

(8) On the basis of the analysis and evaluations set forth in this statement, after weighing the environmental, technical, and other benefits against the environmental costs at the operating license stage, the staff concludes that the action called for under NEPA and 10 CFR 51 is the issuance of an operating license for Hope Creek Generating Station, subject to the following conditions for protection of the environment:

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

  (b) The applicant shall carry out the environmental monitoring programs outlined in Section 5 of this statement, as modified and approved by the staff, and implemented in the Environmental Protection Plan and Technical Specifications that will be incorporated in the operating license for Hope Creek. Monitoring of the aquatic environment shall be as specified in the National Pollution Discharge Elimination System (NPDES) Permit.

  (c) If adverse environmental effects or evidence of impending irreversible environmental damage occurs during the operating life of the plant, the applicant shall provide the staff with an analysis of the problem and a proposed course of corrective action.

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FOREWORD

This Draft Environmental Statement was prepared by the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation (the staff), in accordance with the Commission's regulations set forth in Title 10 of the Code of Federal Regulations, Part 51 (10 CFR 51), which implements the requirements of the National Environmental Policy Act of 1969 (NEPA).

This environmental review deals with the impacts of operation of the Hope Creek Generating Station. Assessments relating to operation that are presented in this statement augment and update those described in the Final Environmental Statement-Construction Phase (FES-CP) that was issued in February 1974 in support of issuance of a construction permit for Hope Creek Unit 1 by

(1) evaluating changes in facility design and operation that will result in environmental effects of operation (including those that would enhance as well as degrade the environment) different from those projected during the preconstruction review

(2) reporting the results of relevant new information that has become available since the issuance of the FES-CP

(3) factoring into the statement new environmental policies and statutes that have a bearing on the licensing action

(4) identifying unresolved environmental issues or surveillance needs that are to be resolved by license conditions

Introductions (résumés) in appropriate sections of this statement summarize both the extent of updating and the degree to which the staff considers the subject to be adequately reviewed.

Copies of this statement and the FES-CP (1974) are available for inspection at the Commission's Public Document Room, 1717 H Street NW, Washington, D.C., and at the Pennsville Public Library, Pennsville, New Jersey. The documents may be reproduced for a fee at either location. Copies of this statement may be obtained by writing to the sources indicated on the inside front cover.

David H. Wagner is the NRC Project Manager for the environmental review of this project. Should there be any questions regarding the content of this statement, Mr. Wagner may be contacted by telephone at (301)492-8525 or by writing to the following address:

Mr. David H. Wagner
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

1 INTRODUCTION

1.1 Résumé

The proposed action is the issuance of an operating license to Public Service Electric & Gas Company (the applicant) for operation of the Hope Creek Generating Station (NRC Docket No. 50-354), in Salem County, New Jersey.

The generating system consists of a boiling water reactor, steam turbine generator, heat dissipation system, and associated auxiliary facilities and engineered safeguards. Waste heat will be dissipated to the atmosphere by a natural draft cooling tower.

The rated thermal capacity of the unit is 3,293 MWt (ER-OL,* Section 3.2), and the net electrical output is approximately 1,067 MWe (ER-OL, Section 3.2).

1.2 Administrative History

On February 27, 1970, Public Service Electric & Gas Company, on behalf of itself and the Atlantic City Electric Company, filed an application for a construction permit (CP) to construct the Newbold Island Nuclear Generating Station, Units 1 and 2. These units were to be built on Newbold Island in the Delaware River in Burlington County, New Jersey. The proposed site was approximately 9.6 km (6 mi) south of Trenton, New Jersey. By letter dated October 5, 1973, the Director of Regulation of the Atomic Energy Commission (now the Nuclear Regulatory Commission) advised the applicant that from an environmental standpoint, a more desirable location for the Newbold Island units would be one on Artificial Island, New Jersey, adjacent to the Salem Nuclear Generating Station, which was then under construction. On November 1, 1973, the applicant amended the application to relocate the facility to Artificial Island and renamed the facility the Hope Creek Generating Station. The design of the units remained unchanged except for modifications to adapt the facility to the new site. On November 4, 1974, Construction Permits CPPR-120 and CPPR-121 were issued for Hope Creek Units 1 and 2, respectively.

In December 1981, Hope Creek Unit 2, which was approximately 18% completed, was cancelled.

By letter dated March 1, 1983, the applicant filed an application for an operating license for Hope Creek Generating Station. Following a predocketing acceptance review, the application was docketed on June 29, 1983. The staff's Draft Safety Evaluation Report was issued on March 5, 1983. The Safety Evaluation Report and the Final Environmental Statement are scheduled for issuance on October 12, 1984, and November 16, 1984, respectively.

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*"Hope Creek Generating Station, Applicant's Environmental Report - Operating License Stage," issued by Public Service Electric & Gas Company in March 1983. Hereinafter this document is cited in the body of the text as ER-OL, usually followed by a specific reference.

The applicant's projected fuel loading date is January 1986.

1.3 Permits and Licenses

The applicant has provided, in Section 12 of the ER-OL, a status listing of environmentally related permits, licenses, and approvals required from Federal and state agencies in connection with the proposed project. The NRC staff has reviewed the listing and the current status of those approvals listed as "not received." The NRC staff notes that the non-NRC approvals discussed below must be received by the applicant before station operation begins.

The issuance of a water quality certification, or waiver therefrom, by the State of New Jersey, pursuant to Section 401 of the Clean Water Act of 1977, is a necessary prerequisite for issuance of an operating license by NRC.

Application for a New Jersey Pollutant Discharge Elimination System (NJPDES) Permit, pursuant to requirements of the Clean Water Act of 1977, must be submitted to the State no later than 180 days before the date on which the discharge is to begin, unless permission for a later application date has been granted by the permitting agency. The applicant submitted an NJPDES Permit application to the State in May 1984, but it has not been acted on by the State. The estimated fuel loading date of January 1986 should leave ample time for the applicant to comply with State certification and permit application requirements, and for the State to take appropriate actions on issuances.

2 PURPOSE OF AND NEED FOR ACTION

The Commission has amended 10 CFR 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection," effective April 26, 1982, to provide that need-for-power issues will not be considered in ongoing and future operating license proceedings for nuclear power plants unless a showing of "special circumstances" is made under 10 CFR 2.758 or the Commission otherwise so requires. (47 FR 12940, March 26, 1982). Need-for-power issues need not be addressed by operating license applicants in environmental reports to the NRC, nor by the staff in environmental impact statements prepared in connection with operating license applications (10 CFR 51.21, 51.23(e), and 51.53(c)).

This policy has been determined by the Commission to be justified whether or not the additional capacity to be provided by the nuclear facility may be needed to meet the applicant's load responsibility. The Commission has determined that the need for power is fully considered at the construction permit (CP) stage of the regulatory review where a finding of insufficient need could factor into denial of issuance of a CP. At the operating license review stage, the proposed plant is substantially constructed and a finding of insufficient need would not, in itself, result in denial of the operating license. The Commission was further influenced by the substantial information that supports the conclusion that nuclear plants are lower in operating costs than conventional fossil plants. If conservation, or other factors, lowers anticipated demand, utilities remove generating facilities from service according to their costs of operation, and the most expensive facilities are removed first. Thus, a completed nuclear plant would serve to substitute for less economical generating capacity (47 FR 12940, March 26, 1982; see also 46 FR 39440, August 3, 1981).

Accordingly, this environmental statement does not consider need for power. Section 6 does, however, consider the savings associated with operation of the nuclear plant.

2.1 References

U.S. Nuclear Regulatory Commission, "Need for Power and Alternative Energy Issues in Operating License Proceedings," proposed rule, Federal Register, 46 FR 39440, August 3, 1981.

---, "Need for Power and Alternative Energy Issues in Operating License Proceedings," final rule, Federal Register, 47 FR 12940, March 26, 1982.

3 ALTERNATIVES

The Commission has amended its regulations in 10 CFR 51, effective April 26, 1982, to provide that issues related to alternative energy sources will not be considered in ongoing and future operating license proceedings for nuclear power plants unless a showing of special circumstances is made under 10 CFR 2.758 or the Commission otherwise so requires (47 FR 12940, March 26, 1982). In addition, these issues need not be addressed by operating license applicants in environmental reports to the NRC nor by the staff in environmental impact statements prepared in connection with operating license applications (10 CFR 51.21, 51.23(e), and 51.53(c)).

In promulgating this amendment, the Commission noted that alternative energy source issues are resolved at the construction permit (CP) stage and the CP is granted only after a finding that, on balance, no obviously superior alternative to the proposed nuclear facility exists. The Commission concluded that this determination is unlikely to change even if an alternative is shown to be marginally environmentally superior in comparison with operation of the nuclear facility because of the economic advantage that operation of the nuclear plant would have over available alternative sources (47 FR 12940, March 26, 1982; see also 46 FR 39440, August 3, 1981).

3.1 References

U.S. Nuclear Regulatory Commission, "Need for Power and Alternative Energy Issues in Operating License Proceedings," proposed rule, Federal Register, 46 FR 39440, August 3, 1981.

---, "Need for Power and Alternative Energy Issues in Operating License Proceedings," final rule, Federal Register, 47 FR 12940, March 26, 1982.

4 PROJECT DESCRIPTION AND AFFECTED ENVIRONMENT

4.1 Résumé

This résumé highlights changes to the plant design and operating characteristics since the FES-CP was issued in 1974.

A number of changes in design and operating characteristics have occurred since the issuance of the FES-CP. Most notable of these is the cancellation of Unit 2 in 1981. Cancellation of Unit 2 resulted in the elimination of the Unit 2 cooling tower and reactor building (Section 4.2.1) and a drastic reduction in the quantity of water withdrawn from the Delaware River (Section 4.2.3.2).

Other major changes to the design and operation of the facility include relocation of the discharge pipe closer to the shoreline (Section 4.2.4.2), revised transmission facilities (Section 4.2.7), creation of an Emergency Operations Facility in the Nuclear Training Center approximately 12.1 km (7.5 mi) northeast of the site (Section 4.2.1), and elimination of the cold water bypass system (Section 4.2.4).

A new development since the FES-CP was issued requires that the applicant develop plans for a supplemental water storage reservoir to compensate for consumptive water use from the Delaware River when the freshwater flow falls below a prescribed limit (Section 4.2.3.2).

4.2 Facility Description

4.2.1 External Appearance and Station Layout

A general description of the external appearance and plant layout is provided in Section 3 of the FES-CP. Since the issuance of the FES-CP, the major change has been the cancellation of the second unit, which resulted in the elimination of the Unit 2 reactor building and cooling tower. A sketch of the facility when completed is included as Figure 4.1. Figure 4.2 presents the station layout. The major buildings and components on the site include the reactor building, turbine building, administration facility, diesel generator area, radwaste service area, cooling tower, warehouses, auxiliary boilers, and shops. Other changes that have occurred since the FES-CP was issued include the addition of the Emergency Operations Facility, which is located in the Nuclear Training Center about 12.1 km (7.5 mi) northeast of the site.

4.2.2 Land Use

Land Use on the Site

The layout of the station facilities is shown in Figure 4.2, and the general site area is shown in Figure 4.3. The only significant change in the plant layout since the FES-CP was issued has been the elimination of the Unit 2 cooling tower and reactor building. Of the 300 ha (741 acres) on the site, 62 ha

(153 acres) will be devoted to permanent plant facilities. The site is located on Artificial Island, an extension of the New Jersey mainland created from the disposal of hydraulic dredging spoils by the U.S. Army Corps of Engineers. The spoil-derived soils are of low agricultural quality, and there is no agriculture on the island (ER-OL, Section 2.1.1.2).

Other property at the site consists of 89 ha (220 acres) occupied by the Salem Generating Station and 149 ha (368 acres) of unoccupied property not currently committed to any future facilities. Approximately 83 ha (205 acres) of this uncommitted land was used for the disposal of excavated waste material from the construction area and of dredging spoils from the channel for the circulating and service water intake system. The land within the site area is zoned as industrial by the Lower Alloways Creek Township Planning Board (ER-OL, Section 2.1.1.2). Access to the site is by an 8.5-km (5.3-mi) road from Alloway Creek Neck Road 4.8 km (3.0 mi) east of the site (ER-OL, Section 3.1.1) and by a barge slip located northwest of the Hope Creek reactor building (Figure 4.2, Item 27).

4.2.3 Water Use and Treatment

4.2.3.1 General

The water-use scheme for Hope Creek has changed somewhat since the FES-CP was issued. The present water-use design for Hope Creek now includes a one-unit boiling water reactor at the Artificial Island site instead of two units as originally planned. A closed-cycle cooling system that uses a natural draft cooling tower will still be used. Principal changes in water use from the scheme presented in the FES-CP include reductions in the amounts of water withdrawn and returned to the Delaware River and elimination of the cold water bypass system.

4.2.3.2 Surface Water Use

Service water is withdrawn from the Delaware River to supply the safety auxiliary cooling system and the reactor auxiliary cooling system. At the construction permit stage, a cold water bypass system was planned, but this design has been eliminated and the entire service water flow now goes to the cooling tower. Because of this change, single-unit withdrawal has been reduced from approximately 160,000 l/min (42,270 gal/min) to 124,400 l/min (32,870 gal/min), and single-unit return to the river has been reduced from approximately 114,000 l/min (30,120 gal/min) to 80,900 l/min (21,370 gal/min). Cancellation of Unit 2 (and elimination of the cold water bypass system) has reduced the total withdrawal of cooling water from the Delaware River from approximately 321,700 l/min (85,000 gal/min) for both units to 124,400 l/min (32,870 gal/min) for a single unit, which represents a 61% reduction in total surface water use.

A comparison of water system flow rates as proposed in the FES-CP and as currently proposed is shown in Table 4.1.

A new development related to surface water use since the FES-CP was issued requires that the applicant develop plans for a supplemental water storage reservoir to compensate for consumptive water use from the Delaware River when the freshwater flow as measured at Trenton, New Jersey, is less than 85 m^3/s

($3,000\ ft^3/s$). An application has been filed with the Delaware River Basin Commission for such a reservoir and is currently under review by that agency.

4.2.3.3 Groundwater Use

Two groundwater production wells approximately 244 m (800 ft) deep supply fresh water for domestic use and for the makeup water demineralizers. Capacity of each well is 2,500 l/min (660 gal/min). Potable water use at the station is expected to amount to as much as 192,000 l/day (50,700 gal/day).

4.2.3.4 Water Treatment

Sodium hypochlorite is used for the chlorination of the service and circulating water systems to control biological growth and organic fouling. According to the ER-OL, service water will be treated periodically with a 1% sodium hypochlorite solution three times each day. Chlorination frequency and deviation will vary according to seasonal demands. Free chlorine concentrations of 0.5 mg/l as an instantaneous maximum and a 0.2 mg/l average will be attained in the cooling tower blowdown line. Calcium carbonate scale is controlled with a 1% sulfuric acid addition to the circulating water system. Auxiliary boiler feedwater is treated daily with 0.5 kg (1.1 lb) of ammonia to maintain a pH in the range of 8.5-9.0. Well water for domestic use is chlorinated to meet water quality standards of the New Jersey Department of Environmental Pollution. Potable and sanitary waste discharges to the onsite sewage treatment facility are chlorinated before being released into the Delaware River to meet the National Pollution Discharge Elimination System (NPDES) effluent discharge standards.

4.2.4 Cooling System

Cancellation of Unit 2 has resulted in several changes in the design and operation of the cooling system since the FES-CP was issued (Tables 4.2 through 4.5). The primary changes are (1) elimination of the cold water bypass system, (2) four instead of eight service water pumps, (3) a reduction in the amount of water withdrawn and discharged into the Delaware River, and (4) a change in the location and size of the discharge pipe.

4.2.4.1 Intake System

The service water intake structure occupies 34 m (112 ft) of shoreline with the west face of the intake being parallel to and flush with the shoreline. Water flowing into the structure has a maximum velocity of 10.7 cm/s (0.35 ft/s) with 11.9 cm/s (0.39 ft/s) approaching the traveling screens. Four instead of eight vertical traveling screens will be modified to provide a system for returning live fish to the Delaware River (Table 4.2).

Each section of the traveling screens has a trough on the lower lip that prevents fish from being reimpinged and allows them to remain in water as they are being lifted to the return troughs. Organisms and debris removed from the screens by a series of low- and high-pressure screen washes are returned to the Delaware River at a sufficient distance from the intake structure to reduce the potential for reimpingement. In addition to modification of the traveling screens for a fish return system, the velocity of the traveling screens has been reduced from 0.15 m/s (0.50 ft/s) to 0.12 m/s (0.40 ft/s) and the wire mesh size of the screens has been changed from 0.95 cm x 0.95 cm (3/8 in. x 3/8 in.) to 1.27 cm x 0.32 cm (1/2 in. x 1/8 in.) (Table 4.2).

With one unit and elimination of the cold water bypass system, the total water withdrawn from the Delaware River will be 124,400 l/min (32,870 gal/min) instead of 321,700 l/min (85,000 gal/min), a reduction of 61%.

4.2.4.2 Discharge System

Comparison of discharge system characteristics between the CP and OL stage is presented in Table 4.5. The two major changes in the discharge system are due to the cancellation of Unit 2 and the placement of the discharge pipe 3 m (10 ft) off shore instead of 60 m (200 ft) off shore as presented in the FES-CP. In addition, the discharge pipe originally was to be placed 300 m (980 ft) upstream of the intake structure, but now it is 160 m (530 ft) upstream of this structure. Discharge flow per unit has changed from 114,000 l/min (30,120 gal/min) (includes bypass flow) to 80,900 l/min (21,370 gal/min).

4.2.5 Radioactive Waste Management System

Under requirements set by 10 CFR 50.34a, an application for a permit to construct a nuclear power reactor must include a preliminary design for equipment to keep levels of radioactive materials in effluents to unrestricted areas as low as is reasonably achievable (ALARA). The term "ALARA" takes into account the state of technology and the economics of improvements in relation to benefits to the public health and safety and other societal and socioeconomic considerations and in relation to the utilization of atomic energy in the public interest. Appendix I to 10 CFR 50 provides numerical guidance on radiation dose design objectives for light-water-cooled nuclear power reactors (LWRs) to meet the requirement that radioactive materials in effluents released to unrestricted areas be kept ALARA.

To comply with the requirements of 10 CFR 50.34a, the applicant provided final designs of radwaste systems and effluent control measures for keeping levels of radioactive materials in effluents ALARA within the requirements of Appendix I to 10 CFR 50. In addition, the applicant provided an estimate of the quantity of each principal radionuclide expected to be released annually to unrestricted areas in liquid and gaseous effluents produced during normal reactor operations, including anticipated operational occurrences.

The NRC staff's detailed evaluation of the radwaste systems and the capability of these systems to meet the requirements of Appendix I will be presented in Chapter 11 of the Safety Evaluation Report (SER), which is to be issued in October 1984. The quantities of radioactive material that the staff calculates will be released from the plant during normal operations, including anticipated operational occurrences, are presented in Appendix D of this statement, along with examples of the calculated doses to individual members of the public and to the general population resulting from these effluent quantities.

The staff's detailed evaluation of the solid radwaste system and its capability to accommodate the solid wastes expected during normal operations, including anticipated operational occurrences, will be presented in Chapter 11 of the SER.

As part of the operating license for this facility, the NRC will require Technical Specifications that limit release rates for radioactive material in liquid and gaseous effluents and that require routine monitoring and measurement of all principal release points to ensure that the facility operates in conformance with the radiation-dose-design objectives of Appendix I to 10 CFR 50.

4.2.6 Nonradioactive Waste Management Systems

4.2.6.1 General

Nonradioactive effluents will result from the operation of the Hope Creek cooling water system, the chemical wastes treatment system, and the sanitary wastewater treatment system. Some changes have occurred in these systems since the FES-CP was issued.

4.2.6.2 Cooling Water System

Plant makeup water will be taken from the Delaware River and a 1% sodium hypochlorite solution will be injected into the circulating and service water systems to control biological growth. According to the ER-OL, sodium hypochlorite will be injected at approximately 30-min intervals three times a day. Dosage rate is controlled to maintain a measurable free available chlorine concentration at the outlet of the main condensers and in the cooling tower basin. The applicant has indicated (ER-OL, Response to Question E291.22) in his application for renewal of the State discharge permit for the station that the cooling tower blowdown will be dechlorinated, as necessary, with sulfur dioxide so that the maximum daily total residual chlorine concentration does not exceed 0.5 mg/l.

To control calcium carbonate scale in the circulating water system supplying the natural draft cooling tower, sulfuric acid will be added to the circulating water pump pits, with the circulating water pH being maintained at the pH of saturation. Sodium hypochlorite will also be added to control biological growth. Makeup water is supplied to the service water system at approximately 121,000 l/min (32,000 gal/min) and the cooling tower blowdown is approximately 71,800 l/min (19,000 gal/min). In addition to residual chlorine and chlorides from cooling water treatment with chlorine and sulfates from acid addition, the blowdown will contain the same constituents as the makeup waters, but concentrated about 1.5 times because of the evaporation of water in the cooling tower.

4.2.6.3 Chemical Wastes Systems

The chemical wastes system consists of the oily water and low volume wastewater system and receives effluent wastes from the makeup demineralizers, auxiliary boilers, turbine building emergency sumps, switchyard and transfer drains, circulating water chemical storage and water treatment system, and diesel generator and control room drains.

Two trains of ion exchange demineralizers are used to supply 570 l/min (150 gal/min) of demineralized water. Cancellation of Unit 2 has resulted in a revision of the regeneration scheme for the demineralizers. The cation/anion train now requires 332 kg (732 lb) and 139 kg (306 lb) of H_2SO_4 and NaOH, respectively, per regeneration instead of 118 kg (260 lb) of sulfuric acid and 68 kg (150 lb) of sodium hydroxide for each unit as presented in the FES-CP. The mixed bed requires 65 kg (143 lb) each of H_2SO_4 and NaOH per regeneration. A mixed-bed demineralizer was not anticipated for use at the time of the FES-CP.

The auxiliary boiler feedwater is treated with 0.5 kg (1.1 lb) of ammonia per day to maintain a pH range of 8.5 to 9.0. Primary impurities in the boiler blowdown will be suspended solids, oil and grease, copper, and iron.

4.2.6.5 Sanitary Waste Treatment System

Potable and sanitary wastes will be discharged to the sewage treatment facility. This facility consists of an air-injector-type lift station, a 75,700-l (20,000-gal) capacity surge tank, three activated-sludge plants, a waste sludge holding tank, and a chlorination system. The activated-sludge plants are package plants operated in the extended aeration mode. Three complete plants are used (that is, two 30,200 l/day (8,000 gal/day) and one 132,500 l/day (35,000 gal/day) because of the large range in organic and hydraulic loading of the system. The facility will normally accommodate the station staff plus the PSE&G Nuclear Department facility staff on Artificial Island. Additional personnel on site during refueling may raise the total to 2,320 persons. Liquids discharged from the settling tanks are discharged to a chlorine contact chamber. Approximately 272 kg (600 lb) of chlorine per 3,780,000 l (1,000,000 gal) of effluent is used before discharges are released to the Delaware River via the storm drainage system.

4.2.7 Power Transmission System

At the construction permit stage, two 500-kV single-circuit power lines were proposed for the Hope Creek station (ER-OL, Section 3.9). One was a 121-km (75-mi) line to Tuckerton, New Jersey, and the other was a 69-km (43-mi) line to the New Freedom switching station. The Tuckerton line, however, was cancelled and no construction was initiated. Only the New Freedom line and a short, onsite tie line connecting the Hope Creek station with the adjacent Salem Generating Station (SGS) resulted from the addition of the Hope Creek station to the power system. In addition to these two new lines, several changes in power-line connections with SGS were made on the site. First, an existing 500-kV SGS to New Freedom line, which is now paralleled by the partially constructed New Freedom line on the same right-of-way (ROW), was disconnected from SGS and reconnected to the Hope Creek station. The New Freedom line being constructed as a result of the addition of the Hope Creek station will be connected to SGS. This rearrangement of connections precluded the necessity for the two adjacent 500-kV lines to cross each other where they join the common ROW. Second, the existing 500-kV SGS - Keeney line was also disconnected from SGS and reconnected to the Hope Creek station. SGS is now connected to the Keeney switching station only indirectly through the 500-kV tie line between SGS and the Hope Creek station. This arrangement of connections also avoided the crossing of lines on the site. Thus, the new arrangement differs from the construction permit stage primarily in that the Hope Creek station is connected to the power system by the preexisting line to Keeney and the new tie line to SGS. A Hope Creek station line to New Freedom was already planned at the construction permit stage. These changes in the power system resulted primarily from the cancellation of Hope Creek Unit 2.

The SGS to New Freedom line is the only new line that lies off the site. It is partially constructed and lies on one-half of a preexisting 107-m (350-ft) ROW. The other half is occupied by the preexisting Hope Creek station to New Freedom line (formerly the SGS to New Freedom line). The new line will have the same design as the preexisting one. The conductors will be supported by single-circuit lattice-type metal towers with a typical base dimension of 6 to 12 m (20 to 39 ft) and heights of 30 to 57 m (98 to 187 ft). Each of the three phases (conductor bundles) will consist of two subconductors attached to the towers by V-string insulators. Spans between towers will range from 305 to 427 m (1,000 to 1,400 ft). Minimum conductor clearance above ground will be 10 m (33 ft). Two shielding wires (ground wires) will be suspended at the tops of the towers, and each tower will be connected to the earth by a ground wire. The line will be constructed in accordance with the National Electric Safety Code. The onsite tie line is already in operation and is about 0.76 km (0.47 mi) long. It consists of a 0.45-km (0.28-mi) section of the preexisting SGS to Keeney line and a 0.31-km (0.19-mi) section of new line. The Hope Creek transmission facilities are depicted in Figure 4.4.

4.3 Project-Related Environmental Description

4.3.1 Hydrology

The hydrologic description presented in Section 2.5 ("Hydrology") of the FES-CP is still valid. The present hydrologic description has been updated to reflect new information gathered since the FES-CP was issued. It also includes a more detailed description of the estuarine water, surface drainage, and groundwater at and adjacent to the plant site.

4.3.1.1 Estuarine Water

The Hope Creek station shares Artificial Island with the two Salem nuclear units. Artificial Island is located on the eastern shore of the Delaware River Estuary. The Delaware River Estuary extends from Liston Point (river mile 48.2) to the head of tide above Trenton, New Jersey, at river mile 133.4.

Hope Creek is located in the estuarine zone approximately 3.9 km (2.4 mi) upstream of Liston Point. The Delaware River extends upstream from river mile 133.4, and the Delaware Bay lies between Liston Point and the Atlantic Ocean.

The largest tributaries of the Delaware Basin in the estuarine zone are the Schuylkill and the Lehigh Rivers in Pennsylvania; the Christina River in Delaware; and the Assunpink, Crosswicke, Rancocas, and Salem Rivers and Big Timber, Hope, and Alloway Creeks in New Jersey. The Chesapeake and Delaware (C & D) Canal, which connects the Delaware Estuary with the Chesapeake Bay, is located about 11 km (7 mi) upstream from the Hope Creek site.

The Delaware River Estuary System drains a basin of 36,000 km^2 (13,900 mi^2), which includes parts of Delaware, New Jersey, Pennsylvania, Maryland, and New York (Figure 4.5). The contributory flows from these tributaries discharging into the Delaware River are shown in Table 4.6. Approximately 25,100 km^2 (9,700 mi^2), or 70% of the drainage area, consists of consolidated rock aquifers of low capacity; as a result, the basins tend to drain quickly to the river system.

The mean annual precipitation in the Delaware Basin is 112 cm (44 in.), which is equivalent to an annual precipitation volume of 4×10^{10} m^3 (1.42×10^{12} ft^3). As shown in Table 4.6, the average freshwater flow for the total drainage area is 644.7 m^3/s (22,765 ft^3/s). In comparison, the average tidal flow (measured at Wilmington, Delaware, approximately 32 km (20 mi) upstream of the site) is 11,328 m^3/s (400,000 ft^3/s). Thus, the tidal flow dominates the freshwater flow by a factor of nearly 18 to 1 and, therefore, controls the velocities at the site. Even the maximum recorded discharge of 9,317 m^3/s (329,000 ft^3/s) that occurred at Trenton, New Jersey, on August 20, 1955, is less than the normal tidal flow at Wilmington, Delaware.

The tide in the Delaware Estuary is semidiurnal in character. There are two high waters and two low waters in a tidal day (approximately 24 hours 50 min). Reedy Point is the primary tidal gaging station nearest the site. Table 4.7 shows the characteristics of the tide at the gage.

Extreme variations in the water levels are storm induced and have resulted from tropical windstorms (hurricanes) and extra tropical windstorms. During the past 40 years three windstorms have given rise to abnormally high still water levels adjacent to the site that ranged up to 2.6 m (8.5 ft) above mean sea level (MSL) (exclusive of waves). The extreme water levels observed in the vicinity of the plant site during these events are shown in Table 4 8. The extreme low water levels observed in the vicinity of the plant site during windstorms are also shown in Table 4.8.

The probable maximum hurricane surge stillwater level is 7.56 m (24.8 ft) MSL. The design-basis flooding event (maximum combination of storm surge, river flood, and wave runup associated with the probable maximum hurricane) established a minimum design flood protection level for the Hope Creek power block area at el 9.45 m (31.0 ft) MSL. This is 5.94 m (19.5 ft) above plant grade at 3.81 m (12.5 ft) MSL. Cooling water for the plant is provided by a natural draft cooling tower system with service water/cooling tower makeup withdrawn from the Delaware Estuary through an intake structure located on the shoreline. The discharge (cooling tower blowdown) is returned into the estuary through a submerged pipeline at the shoreline. Maximum tidal currents in the Delaware Estuary 4,710 m (15,450 ft) upstream from the plant site are 1.23 m/s (4.05 ft/s) at flood tide and 1.34 m/s (4.39 ft/s) at ebb tide; 1,510 m (4,950 ft) downstream the flood tide current is 0.77 m/s (2.53 ft/s) and the ebb tide current is 0.98 m/s (3.21 ft/s).

The average salinity in the Delaware Estuary adjacent to the site ranges from 5 to 18 parts per thousand (ppt) during periods of low freshwater flows to 0.5 to 5.0 ppt at all other times.

Surface water temperatures have been monitored from 1977 through 1982. The water temperature varies from a low of -0.9°C (30.4°F), which occurred in February 1982, to a maximum monthly temperature of 30.5°C (86.9°F), which occurred in August 1980. Average monthly temperatures vary seasonally from a low of 1.4°C (34.5°F) in February to a high of 27.1°C (80.8°F) in August.

Occasionally, the surface and bottom measurements of salinity and temperature varied. The salinity varied as much as 2.0 ppt per meter of depth, and the temperature varied from 1° to 2°C (2° to 4°F) from surface to bottom.

4.3.1.2 Surface Drainage

Artificial Island is generally quite flat and low in elevation except for the area around the plant, which was raised by fill to el. 3.8 m (12.5 ft) MSL. The control of the surface drainage of the undeveloped part of the island is limited to a few drainage ditches constructed for insect control by the local insect control district. The surface drainage system of the plant site is controlled by both drainage channels and catchbasin/underground drain pipes. All drainage from the site flows into the Delaware Estuary.

4.3.1.3 Groundwater

The site is on the Atlantic Coastal Plain about 29 km (18 mi) south of the fall zone. The applicant has identified several aquifers that underlie the site and reports that the aquifers of the Coastal Plain are almost all unconsolidated sand and gravel. The most productive aquifers under the site are the Raritan and Magothy Formations. Other aquifers are the Mount Laurel and the Wenonah Formations, the Englishtown Formation, and the Vincentown Formation. Sands and gravels of the Pleistocene and Recent Age are irregularly distributed throughout the Coastal Plain, but are used as aquifers only in a few areas adjacent to the Delaware River. The Mount Laurel and Wenonah Formations function hydrologically as a single unit, and together they are probably the most used aquifers in the region of the site. The aquifer is recharged from precipitation on its upper outcroppings and discharges water in low areas along its outcrop area, particularly beneath the Delaware River.

The aquifers beneath the site are separated from the surficial soils by one or more relatively impermeable silty clay beds. The Pleistocene Sand, which extends to about 9.1 m (30 ft) in depth, is probably of limited areal extent, although it extends over most of the site. It is underlain with the Kirkwood silty clay aquitard. The Vincentown Formation is encountered at a depth of about 21 m (70 ft) and is an aquifer. The Vincentown Formation is underlain with the Hornerstown Sand, which is an aquitard composed of clayey sand. Below is the Navesink Sand, and at about 55 m (180 ft) is the Mount Laurel Sand aquifer. Since the hydraulic gradient of the aquifers at the site is too small to measure, it is likely that any groundwater movement at the site is strongly influenced by the tide.

4.3.1.4 Water Use

The water of the Delaware River at the site and for some 40 km (25 mi) upstream is brackish and, consequently, is not used in this region for domestic supplies; its industrial use is limited to cooling applications. The station operation will use about 1.30 m^3/s (46 ft^3/s) of brackish estuary water in the service water and cooling tower system. This water is available in an inexhaustible supply. The impacts of the station's water use are discussed in Section 5.3.1.

On the New Jersey side of the Delaware River there are six towns within a 40-km (25-mi) radius of the site that have public water supplies. Salem is the only one of these towns that obtains a part of its water supply from surface sources (Alloway Creek about 13 km (8 mi) northeast of the site). Water for the other towns (and about one-third of the supply for Salem) is pumped from wells. Nearly all of the water supplies for private use are also obtained from wells - most of which are 5 cm (2 in.) in diameter and more than 23 m (75 ft) deep.

Other than the five PSG&E production wells at the Salem and Hope Creek stations, there are no known productive wells closer than 3.2 km (2 mi) to the site. The nearest residences (summer cottages) are about 4.8 km (3 mi) away. The two Hope Creek production wells, HC-1 and HC-2, are approximately 244 m (800 ft) deep and have a maximum capacity of 2,500 l/min (660 gal/min).

4.3.2 Water Quality

4.3.2.1 Surface Water

Water quality in the Delaware River deteriorates as the river flows south into the industrial areas of Chester, Pennsylvania, and Wilmington, Delaware (Table 4.9). Waste discharges from sewage and industrial treatment facilities are largely responsible for the depressed dissolved oxygen levels, decreases in pH, and increases in conductance and temperature downstream (ER-OL), and these industrial discharges probably affect surface water quality near the site. Water quality, as it affects aquatic communities near the site, is a function of total oxygen demand created by municipal and industrial pollutants discharged into the estuary. Dissolved oxygen levels also vary in response to water temperature, primary production levels, and water column mixing. During the winter, dissolved oxygen levels of 9 parts per million (ppm) or greater occur, and levels of 6 ppm generally occur during warmer periods (ER-OL).

The estuarine area near the Hope Creek station is well mixed, and little vertical stratification occurs in physicochemical parameters such as temperature and dissolved oxygen. Salinities in the oligohaline-mesohaline zone near the plant range from 5-18 ppt during periods of low flow to 0.0-3.0 ppt during periods of high flow.

Trace pollutants have been measured in the Delaware River near the Hope Creek station. Cadmium and lead concentrations of 0.016 mg/l and 0.132 mg/l, respectively, were found to exceed Delaware River Basin Commission (DRBC) water quality objectives for zone 5 of the Delaware River Basin. These levels could possibly be due, however, to high natural background, incremental loading by industry along the river, and encroachment of non-point pollution.

4.3.2.2 Groundwater

Groundwater from the shallow aquifer has a high specific conductance (6,000 to 11,000 μmhos per centimeter) and therefore an elevated salt content. The principal ions contributing to this high conductance are chloride and sodium with smaller amounts of calcium and magnesium. The pH of groundwater normally varies from about 5.5 to 7.0, total hardness from about 1,300 to 3,000 mg/l, and turbidity from about 25 to 700 ntu.

4.3.3 Meteorology

The discussion of the general climatology of the site and vicinity contained in the FES-CP remains unchanged. Additional information on extreme meteorological conditions and severe weather phenomena has been collected.

Extreme temperatures of 41.7°C (107°F) and -26.1°C (-15°F) have been reported at Wilmington, Delaware. About 39 thunderstorms can be expected on about

31 days each year. Hail often accompanies severe thunderstorms. During the period 1955-1967, only one occurrence of hail with diameters 19 mm (3/4 in.) or greater was reported in the latitude-longitude box containing the site. However, in the same period, occurrences of large-sized hail in adjacent 1° boxes ranged from 5 to 16. Tornadoes also occur in the area. The applicant has reported that 44 tornadoes have occurred within the 1° latitude-longitude box containing the site in the period 1950-1981, resulting in an annual tornado frequency of 1.4. The staff has performed an independent assessment of tornado occurrences in the Hope Creek region and computed a recurrence interval for a tornado at the plant site of about 9,000 years. Hurricanes or remnants of hurricanes pass through the region occasionally. During the period 1871-1982, 32 tropical cyclones (tropical depressions, tropical storms, and hurricanes) passed within 185 km (100 nautical miles) of the site.

Since issuance of the FES-CP, the applicant has collected 5 additional years (January 1977-December 1981) of onsite meteorological data. For this period of record, prevailing winds at the 10-m (33-ft) level are from the northwest (about 11.6% of the time). A somewhat bimodal airflow pattern is evident, with winds from the west, west-northwest, and northwest occurring about 29% of the time, and winds from the southeast, south-southeast, and south occurring about 22% of the time. The average wind speed at the 10-m (33-ft) level is about 4 m/s (13 ft/s). Calm conditions (defined as wind speeds less than the starting threshold of the anemometer) occur infrequently, less than 0.5% of the time. Neutral (Pasquill type "D") and slightly stable (Pasquill type "E") conditions predominate at the Hope Creek site, occurring about 33% and 29% of the time, respectively, as defined by the vertical temperature gradient between the 45.7-m (150-ft) and 10-m (33-ft) levels for the 5-year period described above. Moderately stable (Pasquill type "F") and extremely stable (Pasquill type "G") conditions occur about 12% and 6% of the time, respectively, for the same stability indicator. Moderately stable and extremely stable conditions were observed with relatively the same frequency during the preoperational program (1969-1971) for the Salem plant, also located on Artificial Island. However, the frequency of unstable conditions was much lower during the earlier measurements program than during the present measurements program (about 12% compared with about 19%).

4.3.4 Terrestrial and Aquatic Resources

4.3.4.1 Terrestrial Resources

Terrestrial biota of the Hope Creek site and the surrounding region were described in the FES-CP, Section 2.7.1, and in the ER-OL, Section 2.2.1. Descriptions in the FES-CP remain valid except for those areas where vegetation was altered during construction. Artificial Island, which consists of dredge spoils, has only low quality habitats for wildlife and thus is not an important natural resource area. Power-line towers, however, have provided important nesting sites for ospreys on the Island (Section 4.3.5.1). Other land and water areas in the surrounding region are much more valuable to wildlife in general. The vegetation of the island is dominated by the giant reed, *Phragmites communis*, which is common to disturbed areas and has recently invaded the island, replacing may stands of more desirable species (ER-OL, Section 2.2.1.2).

Land area types traversed by the new Salem-New Freedom power line, which is the only offsite power line constructed for the Hope Creek station (Section 4.2.7), are given in Table 4.10. To leave the site, the line had to traverse about 8 km (5 mi) of nonforested brackish wetlands (ER-OL, Figure 3.9). These types of wetlands are described in the FES-CP, Section 2.7.1.2, and the ER-OL, Section 2.2.1.2. After leaving these wetlands, the line passes through a mix of habitat types until it approaches the New Freedom Switching Station and runs parallel to the Great Egg Harbor River. Here the line runs for about 4.4 km (2.7 mi) through forested wetlands (ER-OL, Figure 3.9). The forests to be impacted by the line are about 50% upland oak-pine forest and 50% hardwood swamp forest. No large (for example, >8 ha or 20 acres) Atlantic white cedar bogs, an uncommon community type in the area (McCormick, 1970), occur on the power-line ROW (as shown by maps in the reports by McCormick and Jones (1973)), although smaller stands may be present. Except at tower sites, forests had not yet been cleared for the new line at the time of the NRC site visit on February 15, 1984. Of the 368 ha (909 acres) of this line's ROW, 72 ha (178 acres) or less are prime farmlands (ER-OL, Section 3.9.2).

4.3.4.2 Aquatic Resources

The aquatic resources of the Delaware River and Bay in the vicinity of the Hope Creek site were described in the FES-CP, Section 2.7.2. Surveys of the site vicinity conducted since the FES-CP was issued are summarized in the ER-OL, Sections 2.7.2 and 6.1.1.2. The aquatic resources described in this section are based on a much expanded data base since the FES-CP was issued in February 1974. Information presented here was obtained from the applicant (PSE&G, 1980), and this section highlights the aquatic resources that potentially could be affected by operation of the Hope Creek station. A biological sampling program characterizing the aquatic resources in the Delaware River Estuary and nearby tidal creeks was initiated by the applicant in 1968 and completed in 1978. A few special studies on fish and blue crab populations in the site vicinity, however, are still being continued. Sampling through 1969 focused on fish and macroinvertebrates in the Delaware River and local tidal tributaries. In 1971, studies on larval blue crab, benthos, ichthyoplankton, microzooplankton, and distribution of fish were added; in 1973, phytoplankton studies were initiated.

4.3.4.2.1 Ecological Communities

The Delaware River Estuary in the vicinity of the site is typical of many Atlantic Coastal Plain estuaries, having relatively large daily and seasonal fluctuations in physicochemical factors such as salinity and temperature. This section of the Delaware River is classified as oligo-mesohaline with annual salinity fluctuations ranging from 0.5 to 18 ppt. Seasonal and annual regimes of salinity and temperature are the primary factors that determine the composition, abundance, and distribution of organisms in the vicinity of the site. The salinity regime near the site is primarily a function of the seasonal variation in freshwater discharge, with saltwater intrusion being greater during low flow periods (June-October) and least during high runoff in the late fall and spring. During seasons of high flow, freshwater communities dominate and are typical of those biological communities found in the lower reaches of freshwater rivers. When flow is low and salinity relatively high, communities are generally brackish-water types with a few marine species.

Biological productivity of the Delaware River near the site is relatively high because of its estuarine nature. Tidal action influences primary productivity by supplying food, nutrients, and dissolved oxygen to producers. Vertical mixing set up by strong density gradients in this tidal estuary increases productivity by recycling and trapping nutrients, detritus, and planktonic organisms within the estuary. In addition, extensive tidal marshes bordering the river near the site supply nutrients and food in the form of vascular plant detritus to estuarine organisms. As a consequence of this high primary productivity and supply of organic detritus, consumer populations flourish in this section of the Delaware River Estuary. This area of the estuary is important to fishery resources, functioning as feeding and nursery areas for the young of many fish species and also as a passageway for migratory fish.

Biological communities of the Delaware River Estuary are characterized by large populations of a few productive species. The dominant species are generally those that are most tolerant of changes in physicochemical conditions and account for most of the biomass and energy flow within the estuarine food webs near the site. Species that are well adapted to the rigors of the estuarine environment, especially to abrupt changes in salinity, and are also representative of the major trophic levels include the diatom, Skeletonema costatum; the copepods, Eurytemora affinis and Acartia tonsa; the polychaeta, Scolecolepides viridis; and the fish, Anchoa mitchilli (bay anchovy), Morone americana (white perch), and Cynoscion regalis (weakfish).

4.3.4.2.2 Plankton Populations

Phytoplankton production in the section of the Delaware River near the site is limited because of high water turbidity and wind mixing, which circulates many phytoplankton cells out of the euphotic zone. The phytoplankton community is composed of local estuarine populations augmented by input of other groups from upper and lower regions of the estuary. Production of organic matter by phytoplankton probably supplies only a fraction of the total energy required by consumers near the site because high turbidity limits production to the upper 2 m (6.6 ft) of the water column. Therefore, organic matter from other sources such as vascular plant detritus or organic material inputs from other areas of the estuary probably supplements the energy needs of local consumers.

The phytoplankton community is dominated by the diatoms Skeletonema costatum, Melosira spp., and Chaetoceros spp., which usually constitute numerically 75% or more of the phytoplankton community on an annual basis. As is typical of most temperate aquatic ecosystems, production and abundance of phytoplankton peak in the spring following a winter minimum. Production and biomass of phytoplankton appear to be similar on both sides of the Delaware River near the site; water temperature, salinity, and nutrients are the prime determinants of primary production rate.

The zooplankton community in the reach of the Delaware River consists of larval stages of benthic invertebrates (meroplankton), permanent residents of the plankton (holoplankton), and forms that utilize both the bottom substrate and the water column (macroplankton). The types of communities and abundance of species present reflect the physicochemical and environmental conditions of the river such as freshwater flow, salinity, temperature, tidal currents, and light intensity. During high freshwater runoff periods, freshwater zooplankton such as rotifers, the cyclopoid copepod Halicyclops, and the amphipod Gammarus spp. dominate. With low freshwater flow and higher salinities, more saline groups such as ctenophores and the copepods, Acartia tonsa and Pseudodiaptomus, are common. The zooplankton community also varies seasonally in response to reproductive activity and changes in larval density. Horizontal distribution of local zooplankton populations is primarily determined by direction and velocity flow of the river. Some species are passively transported into the area from other regions of the estuary; others use vertical migration mechanisms to maintain their position within the estuary. Common macrozooplankton species such as Neomysis americana and Gammarus spp. display this behavior.

The meroplankton or larval stages of species whose adults live in or on substrates include the fiddler crab (Uca minax), the grass shrimp (Palaemonetes pugio), and the mud crab (Rhithropanopeus hairissi). These species are more abundant in inshore areas near intertidal mud banks or near tidal creeks.

4.3.4.2.3 Benthic Populations

The invertebrates that constitute the benthic community near the site are primarily euryhaline species that are physiologically conditioned to the wide range of physicochemical conditions occurring in this area of the estuary. Both pelagic larvae and adults of benthic organisms are important components of the estuarine food web because they serve as food for many fish species. Attached benthic groups such as hydroids, oysters, and barnacles are important as habitat formers providing shelter and attachment surfaces for other organisms.

The principal factors regulating benthic community composition, distribution, and abundance are salinity, temperature, and substrate. Benthic diversity and abundance are highest when salinity is highest, primarily because of the movement of species from downbay areas into the site vicinity.

Biofouling, principally by oysters and ribbed mussels, has been an ongoing concern at the nearby Salem power plant site since Unit 1 became operational in 1976 (ER-OL, Amendment 1, Response to Question E291.2). Salem Unit 2 has had minor, but continuing, problems since May 1982. It is believed that the drought in 1981 resulted in a salinity increase in the Delaware River, thus making it possible for oyster larvae to be carried upriver (from the oyster beds located about 3.3 km (2 mi) downstream), survive, and enter the plant's intakes. Biofouling by oysters has occurred in the Salem intake structure and intake equipment and within the service water system and diesel generator jacket water coolers (Unit 1) and the containment fan cooling units of Unit 2 (Imbro, 1983). It is anticipated that the Hope Creek circulating and service water intakes will experience some biofouling similar to that observed at Salem. At Salem, chlorine is introduced into the intake structures behind the traveling screens. A similar introduction point will be used at Hope Creek. Since the reproductive season of the major biofoulers extends over a period of several weeks during the late spring through the summer, chlorination might be necessary throughout much of that period to prevent problems from biofouling.

4.3.4.2.4 Fish Populations

Fish populations of the Delaware River near the site are a diverse assemblage of 92 species that can be divided into two groups of 41 resident and 51 migratory species. Residents prefer either tidal-freshwater or estuarine conditions.

Migratory fish can be separated according to type of movement (diadromous, estuarine-dependent, or marine visitors). The dominant resident or estuarine species are the bay anchovy, hogchoker, Atlantic and tidewater silversides, naked goby, and mummichog. Common migratory species are the estuarine-dependent weakfish, spot, Atlantic croaker, and Atlantic menhaden and the diadromous american eel, white perch, blueback herring, and alewife.

Seasonal variations in fish community composition and abundances are influenced by temporal changes in water temperature and salinity along with availability of food resources. Seasonal stability in fish community structure reflects stability of temperature and salinity with many species utilizing the warmer highly productive summer period for spawning, development of larvae and young, and growth. Only a few species, such as white perch and the hogchoker, remain in the site vicinity over the winter.

The fish community near the site is seasonally dynamic with various species migrating through the area most of the year. In the spring, estuarine-dependent species move from downbay overwintering areas and anadromous species migrating from offshore waters move through the area to freshwater spawning grounds. Some species such as bay anchovy, silversides, hogchoker, and white perch arrive near the site area from downbay in March and April to feed before spawning. The anadromous striped bass, American shad, blueback herring, and alewife pass through the area enroute to upstream spawning grounds. Larvae and young of ocean spawning, but estuarine-dependent, menhaden and spot appear at the site in early spring. By mid-June young of estuarine-spawners move into the low salinity nursery areas near the site taking advantage of the warm temperatures and high food production in this area to maximize growth. Abundance and diversity of fish species are highest in the summer as spawning activity slows and young of some species arrive in the area. Abundance declines during fall as decreasing water temperatures and production initiate emigration to overwintering areas downbay and/or off shore. Species such as menhaden, spot, and river herring move through the area from upriver nursery grounds to offshore overwintering areas.

4.3.4.2.5 Fisheries

Fisheries of the Hope Creek site vicinity were discussed in the FES-CP, Sections 2.7.2.2 and 2.7.2.3. The ER-OL provides updated discussions of fisheries resources (Section 2.1.3.5). The discussion below summarizes recent information on recreational and commercial fishery harvests within the 0- to 80-km (0- to 50-mi) area downstream of the site, which essentially includes all of Delaware Bay between Hope Creek and the mouth of the bay.

Commercial fishery harvests from Delaware Bay have ranged from 1.3 million kg (2.9×10^6 lb) to 9.3 million kg (20.5×10^6 lb) annually in recent years (Table 4.11), with about 55% to 86% of the total consisting of shellfish (blue crabs and oysters). Port Norris and Bivalve, New Jersey, are the center of the shellfish industry of Delaware Bay. Commercial finfish landings have ranged between about 0.2 million kg (0.4×10^6 lb) and 3.3 million kg (7.3×10^6 lb) annually (Table 4.11). The harvests have been dominated by weakfish, American eel, American shad, bluefish, menhaden, and carp (see ER-OL, Table 2.1-13, for detailed breakdown of harvest by species).

Recreational fishery harvests from the Delaware waters of the bay have ranged between about 3 million kg (7×10^6 lb) and 4 million kg (9×10^6 lb) annually (Table 4.12), with an additional harvest of about 454,000 kg (1×10^6 lb) from the New Jersey waters during 1980 (telephone conversation between C. Hickey, NRC, and J. McLain, New Jersey Division of Fish, Game, and Shellfisheries, April 4, 1983). The predominant finfish species harvested have been weakfish and bluefish. Shellfish harvested recreationally include hard clam and soft clam (Table 4.12) and blue crab. The major recreational fishing port downstream of Hope Creek on the New Jersey shore is the State Marina at Fortescue, with its fleet of private, charter, and head boats.

The combined commercial and recreational harvests for Delaware Bay downstream of Hope Creek, therefore, have ranged between about 5.2 million kg (11.5×10^6 lb) and 13.7 million kg (30.2×10^6 lb) annually.

4.3.5 Endangered and Threatened Species

4.3.5.1 Terrestrial

The geographic ranges of several species listed as endangered by the Federal Government (50 CFR 17.11 and 17.12) include the State of New Jersey. An endangered plant species, the small whorled pogonia, occurs in hardwood forests in the eastern United States. Although a small population of pogonias exists in northern New Jersey (47 FR 39827-39831, September 10, 1982), this species is not known to occur on or near Artificial Island or the power-line routes.

Both the endangered bald eagle and the peregrine falcon occur as nonbreeding visitors in the site area, although Artificial Island is not particularly important to either species. There is no natural nesting habitat for these species (trees for eagles and cliffs for peregrines) at the site and no individuals are known to nest here or in the vicinity (ER-OL, Section 2.2.3). Although reintroductions of these species are being attempted with the use of artificial nesting platforms in many areas in the eastern United States, including peregrines in southern New Jersey (*Peregrine Fund Newsletter*, 1983), the Hope Creek station should not affect the success of this activity.

The osprey, a fish-eating bird of prey, is listed as endangered by the State of New Jersey, but not by the Federal Government. Ospreys commonly nest on transmission towers in the vicinity of the site, and in some years as many as 12 nests are active (ER-OL, Section 2.2.3), representing the highest concentration of nesting ospreys in the state. Continued construction of the Hope Creek station should not adversely affect the osprey's active nesting or its over-water hunting. It is unlikely that any habitats along the more inland portions of the power-line routes are important to peregrine falcons, bald eagles, or ospreys.

4.3.5.2 Aquatic

The shortnose sturgeon (*Acipenser brevirostrum*) is listed as endangered by both the U.S. Fish and Wildlife Service and the State of New Jersey. A total of 49 incidental captures of shortnose sturgeon in the Delaware River drainage have occurred during the period 1950 through June 1982 (ER-OL, Section 2.2.3). Thirty-six captures have been documented from the Delaware River during the period 1954 through 1980 (NUREG-0671). Five specimens have been captured in the vicinity of Hope Creek: two were taken in the river by gill net and trawl, and three have been collected on the Salem intake structure, all dead before arrival.

Sea turtles also have been observed and captured within the Hope Creek site vicinity, including two threatened species, the Atlantic loggerhead turtle (*Caretta caretta*) and green sea turtle (*Chelonia mydas*), and one endangered species, Kemp's Atlantic ridley turtle (*Lepidochelys kempii*) (ER-OL, Section 2.2.3). Captures have been by bottom trawl in the river (eight turtles) and on the Salem intake structure (three turtles). Two loggerheads were found on the intake, one on July 11, 1980, and one on July 18, 1983. Both had been dead before entrapment. One ridley was removed alive from the Salem Unit 1 intake trash bars on August 11, 1980, and returned alive to the river.

4.3.6 Historic and Archaeological Sites

Section 2.3 of the FES-CP discusses historic sites and landmarks. This section provides a listing of the sites in the surrounding region that are listed in the *National Register of Historic Places*. At present, in the 16.7-km (10-mi) area around the plant, there are six listed properties in Salem County, New Jersey, and one in Cumberland County, New Jersey. Across the river in New Castle County, Delaware, 48 properties are listed and an additional 4 are defined as eligible. One property is listed in Kent County, Delaware. The operation and maintenance of the plant and associated facilities are not expected to affect any of the properties.

4.3.7 Socioeconomic Characteristics

The general socioeconomic characteristics of the region, including demography and land use, are presented in Section 2 of the FES-CP. As indicated in the FES-CP, the plant is located on Artificial Island, which is on the east bank of the Delaware River in Lower Alloways Creek Township, Salem County, New Jersey.

The 16.7-km (10-mi) area surrounding the plant site includes portions of Salem and Cumberland Counties, New Jersey, and New Castle and Kent Counties, Delaware. The area is predominantly tidal marsh, meadowlands, and agricultural land. Industry and business are located largely in the nearby towns of Salem, New Jersey (1980 population 6,959), which is about 13.3 km (8 mi) northeast of the site; Middletown, Delaware (1980 population 2,946), which is about 15.8 km (9.5 mi) west of the site; and Delaware City, Delaware (1980 population 1,858), which is about 13.3 km (8 mi) northwest of the site.

According to U.S. Bureau of Census data, Lower Alloways Creek Township grew from 1,400 persons in 1970 to 1,547 persons in 1980, and Salem's population declined from 7,648 persons to 6,959 persons in the same decade. The population of Middletown increased from 2,644 persons in 1970 to 2,946 persons in 1980, and Delaware City declined from 2,024 persons to 1,858 persons over the same time period. According to the applicant, the 1980 residential population within 10 mi of the site is estimated to be 22,162 persons. About 21,000 of these persons are located in the 8- to 16-km (5- to 10-mi) area around the plant; about three-fourths of these are located in the NNE, NE, NNW, and W sectors (ER-OL, Figure 2.1-5). The residential population within 16.7 km (10 mi) is estimated to be 27,380 in the year 2010 (ER-OL, Figure 2.1-9).

The staff has reviewed the applicant's demography data by comparing his estimates with independent data sources and maps and found that the applicant's estimates are reasonable.

4.4 References

Code of Federal Regulations, Title 50, "Wildlife and Fisheries," U.S. Government Printing Office, Washington, D.C.

Imbro, E. V., "Biofouling at Salem Units 1 and 2," Engineering Evaluation Report, No. AEOD/E319, U.S. Nuclear Regulatory Commission, Washington, D.C., 1983.

McCormick, J., "The Pine Barrens: A Preliminary Ecological Inventory," Research Report 2, New Jersey State Museum, 1970, p. 103.

---, and L. Jones, "The Pine Barrens: Vegetation Geography," Research Report 3, New Jersey State Museum, 1973, pp. i-vii plus 71 maps.

Peregrine Fund Newsletter, No. 11, 1983, p. 2.

Public Service Electric Gas Company, "An Ecological Study of the Delaware River Near Artificial Island, 1968-1975: A Summary," NRC Docket No. 50-272, Newark, New Jersey, 1980.

U.S. Department of the Interior, Fish and Wildlife Service, "50 CFR Part 17 - Endangered and Threatened Wildlife and Plants; Determination of *Isotria medeoloides* (Small Whorled Pogonia) To Be an Endangered Species," Final Rule, 47 FR 39827-39831, September 10, 1982.

---, National Park Service, National Register of Historic Places, Vols. 1 and 2 (and subsequent listings as they appear in the Federal Register), 1976.

U.S. Nuclear Regulatory Commission, NUREG-0671, "Assessment of the Impacts of the Salem and Hope Creek Stations on the Shortnose Sturgeon, *Acipenser brevirostrum* LeSueur," M. T. Masnik and J. H. Wilson, 1980.

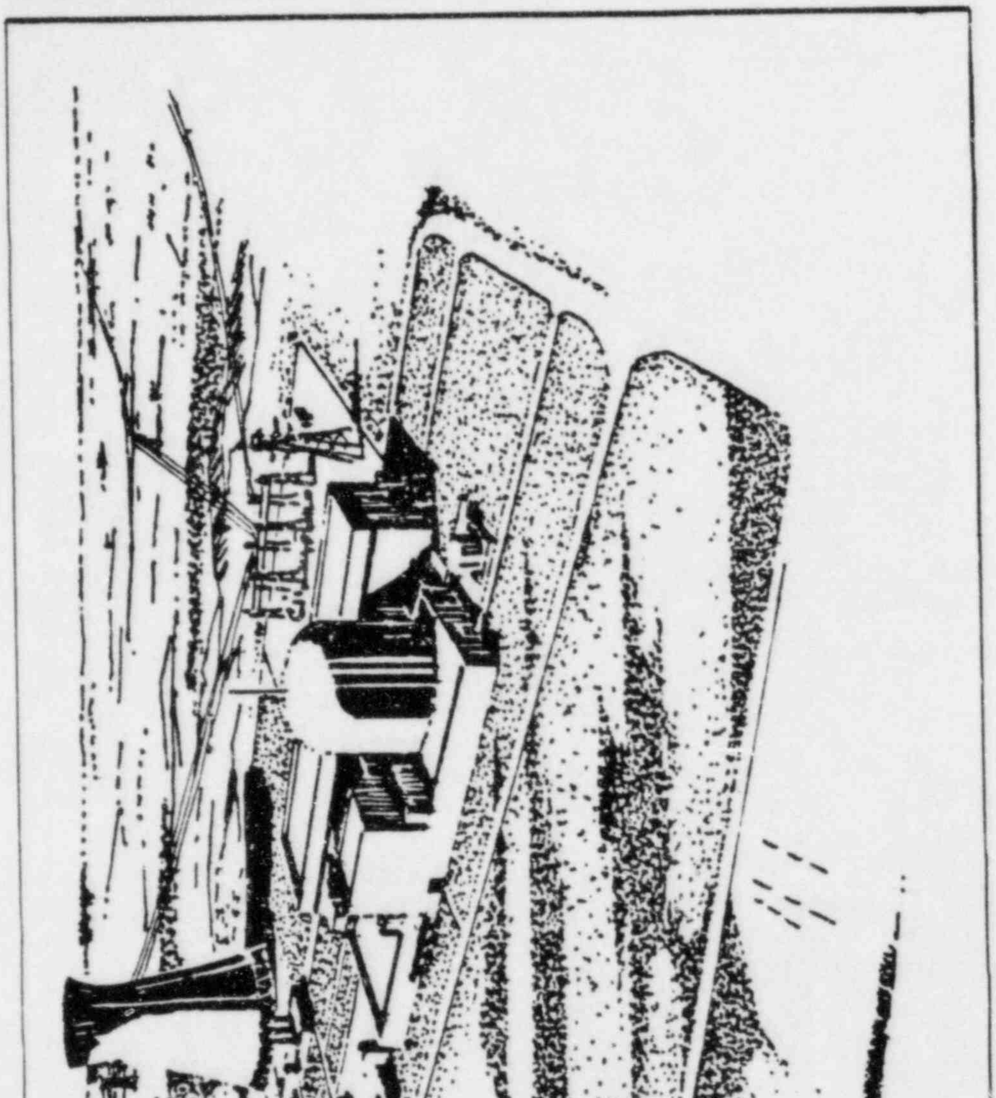

Figure 4.1 Hope Creek Generating Station
Source: ER-OL, Figure 3.1-1

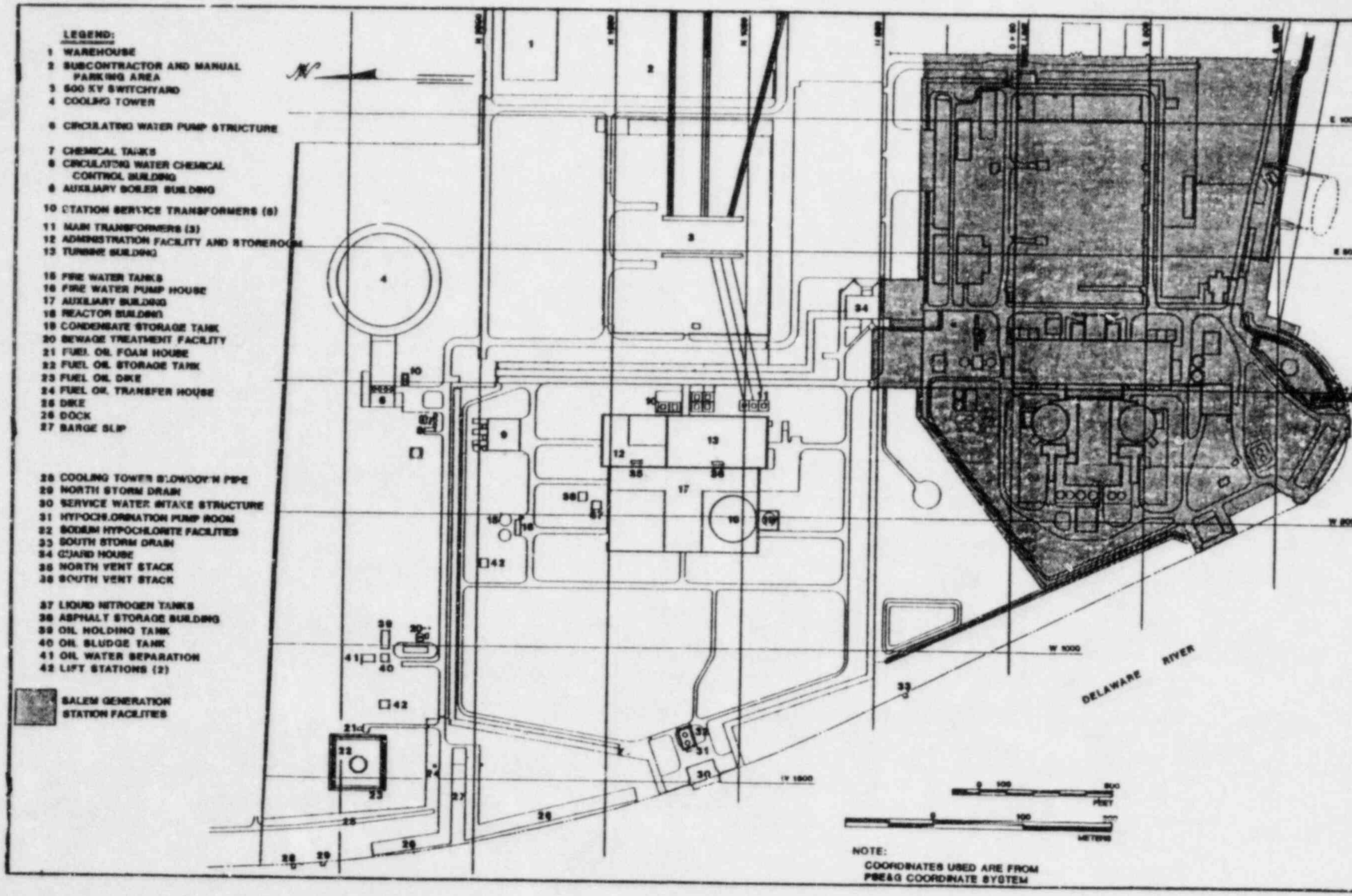

Figure 4.2 Station layout
Source: ER-OL, Figure 2.1-3 (Amendment 3)

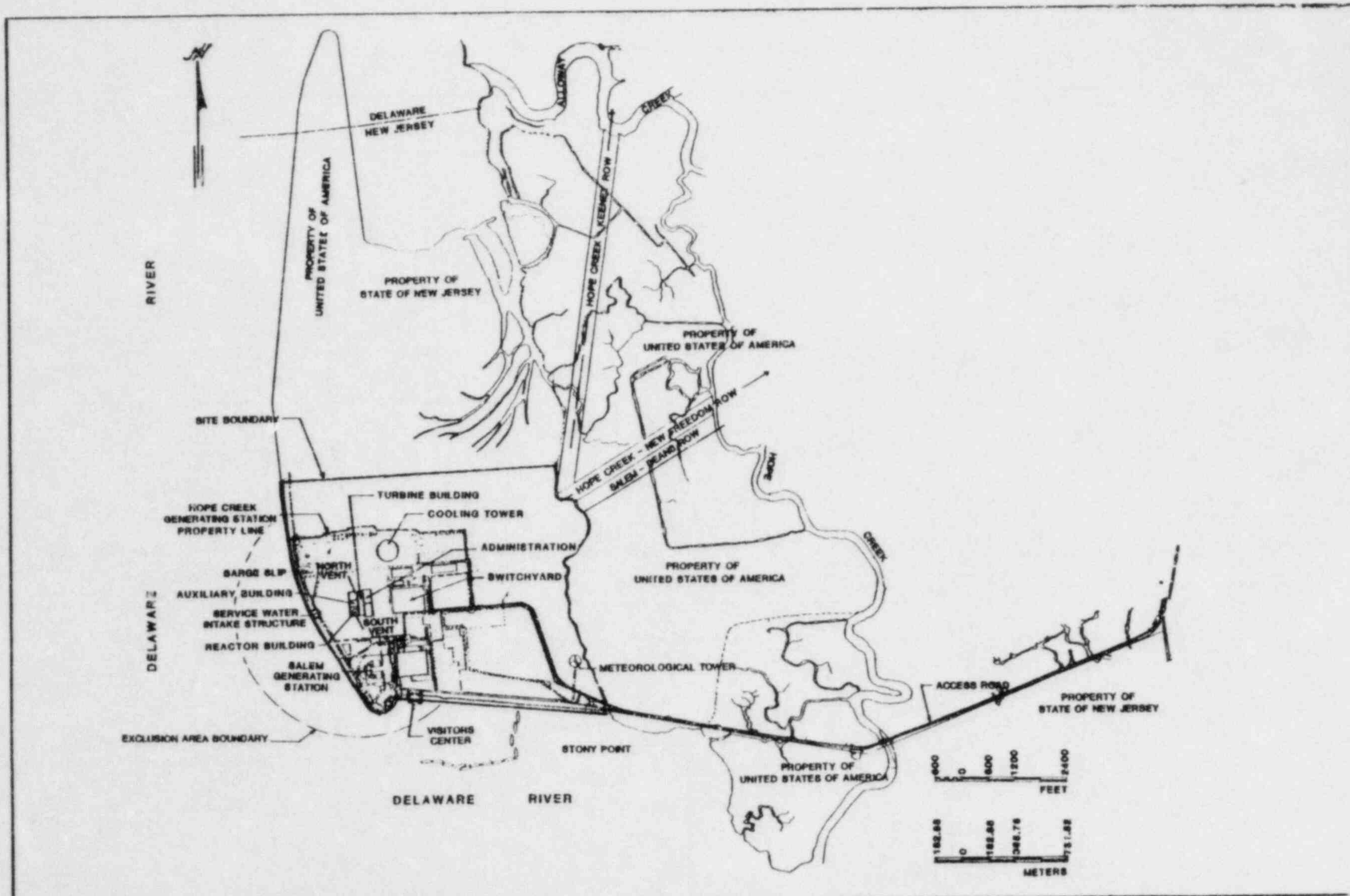

Figure 4.3 Site area
Source: ER-OL, Figure 2.1-4 (Amendment 3)

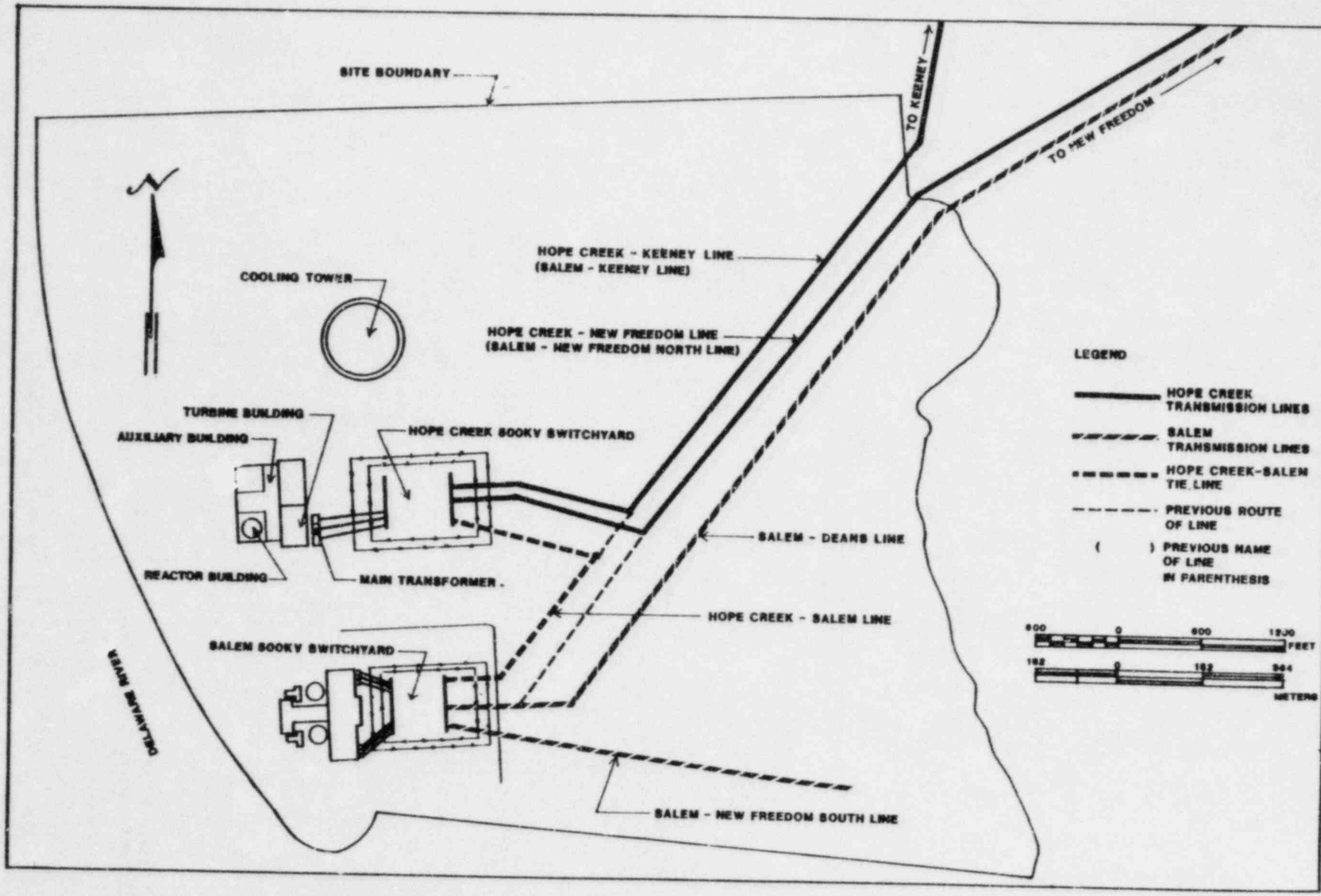

Figure 4.4 Hope Creek-Salem transmission facilities
Source: ER-OL, Figure 3.9-5

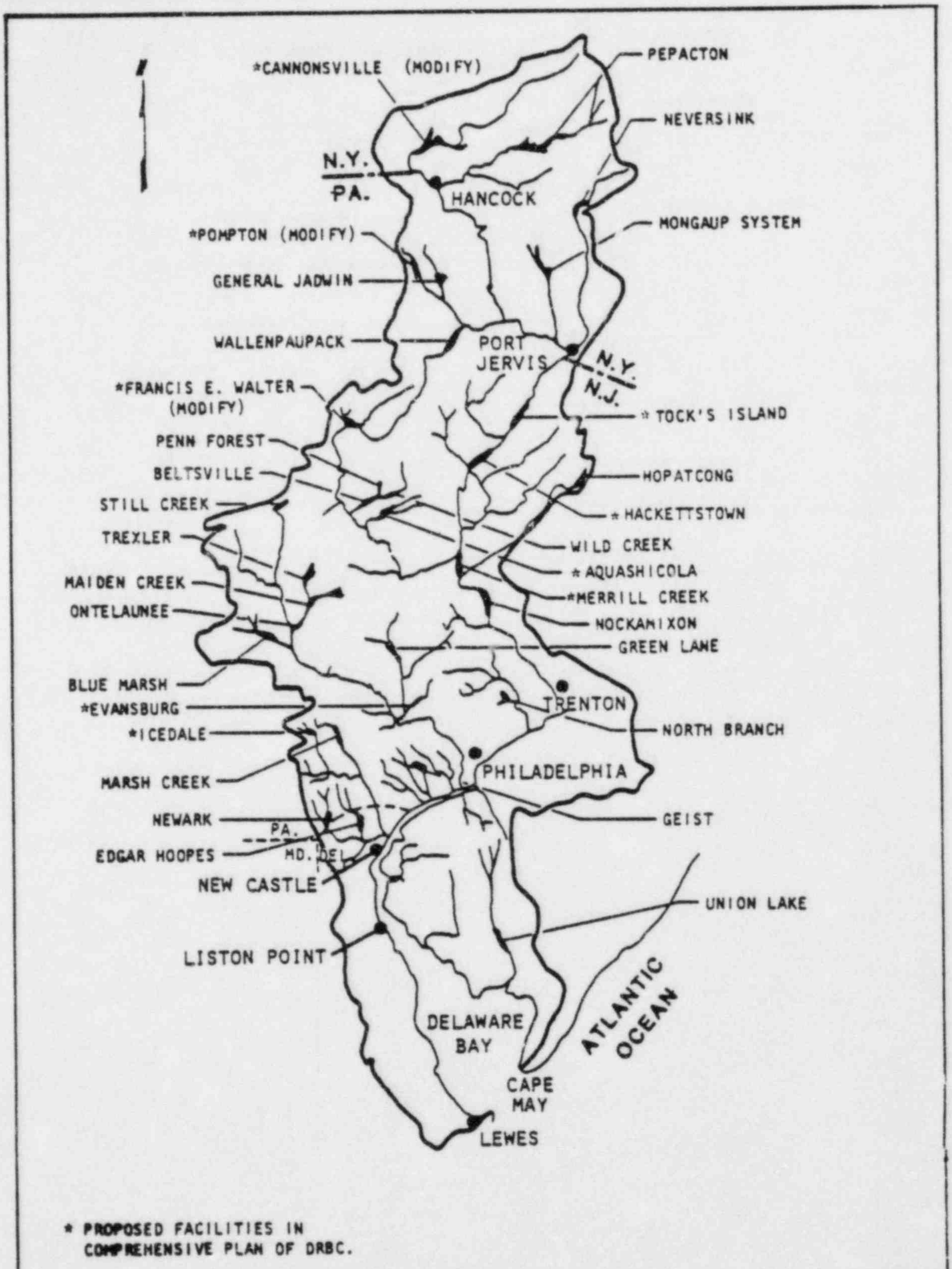

Figure 4.5 Delaware River Estuary system
Source: ER-OL, Figure 2.4-1

Table 4.1 Comparison of water system flow rates and water temperature rises for one unit during normal average operating conditions at the construction permit (CP) stage and the operating license (OL) stage (approximate values)

Function	CP stage (summer averages)		OL stage	
	Flow rate*	Temperature rise**	Flow rate*	Temperature rise**
Withdrawn from river	43,900		34,640***	
Cold water bypass	21,900		0	
Service water system	22,000	18.9	32,870	13
Cooling tower makeup	27,300		32,870	
Evaporation and drift†	13,900		13,300	
Cooling tower blowdown	13,400	14.7-29.8	21,370	3.6-29.5
Discharge to river	30,000	6.6-11.2	21,370	3.6-29.5
Concentration factor	2.0		1.5	

*Flow rates are shown in gal/min. To convert to l/min, multiply values shown by 3.785.

**Temperature rise is shown in °F. To convert to °C, multiply values shown by 0.556.

***Includes screen and strainer wash water of 1,780 gal/min (maximum).

†Daily maximum value shown.

Table 4.2 Comparison of the features of the service water intake structure at the construction permit (CP) stage and the operating license (OL) stage

Parameter	CP stage	OL stage
Number of structures	One	No change
Location	Shoreline - west of reactor building	No change
Service	To cool reactor auxiliaries cooling system and safety auxiliaries cooling system heat exchangers and provide cooling tower makeup	No change
Dilution bypass	Portion of service water (approximately 40%) that bypasses service water system and is used for dilution of discharge	Eliminated
Service water pumps	8 pumps (4 per unit) rated at 56,800 l/min (15,000 gal/min)	4 pumps rated at 62,450 l/min (16,500 gal/min)
River water withdrawal per unit	160,900 l/min* (42,500 gal/min)	124,400 l/min* (32,870 gal/min)
Traveling screens	8 vertical screens	4 vertical screens
Velocity	0.15 m/s (0.50 ft/s)	0.12 m/s (0.40 ft/s)
Wire mesh size	0.95 cm x 0.95 cm (3/8 in. x 3/8 in.)	1.27 cm x 0.32 cm (1/2 in. x 1/8 in.)
Deicing capability	None	As described in Section 3.4-1 of ER-OL
Fish return system	None	As described in Section 3.4-1 of ER-OL

*Approximate values

Table 4.3 Comparison of the features of the service water system at the construction permit (CP) stage and the operating license (OL) stage

Parameter	CP stage	OL stage
Dilution before discharge	Yes	No dilution
Chlorination treatment to control growth	Use of sodium hypochlorite continuous and shock treatment	No change**
Normal number of service water pumps operating (per unit)	3	2
Heat removal rate (per unit)	52.4×10^6 kcal/hr (208×10^6 Btu/hr*	54.3×10^6 kcal/hr (216×10^6 Btu/hr)
Temperature rise	10.5°C (18.9°F)*	7.2°C (13°F)

* Estimate.
** See response to Question E291.16.

Table 4.4 Comparison of the features of the circulating water system at the construction permit (CP) stage and the operating license (OL) stage

Parameter	CP stage	OL stage
Treatment to control organic material	Sodium hypochlorite	No change
Circulating pumps (per unit)	4 vertical wet-pit pumps	No change
Diameter of discharge pipe (tunnel)	3.6 m (12.0 ft) one for each unit	No change
Circulating water flow rate	2,020,000 l/min (552,000 gal/min)	No change
Heat rejection rate	1.96×10^9 kcal/hr (7.76×10^9 Btu/hr)	No change
Circulating water temperature rise	15.5°C (28°F)	No change
Description of main condenser	Double-pass, three-shell, horizontal, deaerating-type surface	No change
Tube length	12 m (40 ft)	No change
Nominal diameter	2.2 cm (0.875 in.)	No change
Total cooling surface area (per unit)	76,300 m^2 (821,430 ft^2)	No change

Source: ER-OL, Table 3.4-8 (Amendment 1).

Table 4.5 Comparison of the features of the discharge water systems at the construction permit (CP) stage and the operating license (OL) stage

Parameter	CP stage	OL stage
Discharge point upstream of intake structure	300 m (1,000 ft)	160 m (460 ft)
Discharge point off shoreline	60 m (200 ft)	3 m (10 ft)
Discharge pipe diameter	1.4 m (4.5 ft)	1.2 m (4.0 ft)
Blowdown rate (per unit)	45,200 l/min* † (12,000 gal/min)	80,900 l/min† (21,360 gal/min)
Average discharge water temperature rise		
Winter	6.2°C (11.2°F)	16.4°C (29.5°F)
Summer	3° to 7°C (6.6°F)	2.0°C (3.6°F)
Discharge velocity	260 cm/s (8.5 ft/s) (two units)	110 cm/s (3.5 ft/s)
Discharge water temperature rise (coldest month)	7.8°C (14°F)	19.6°C (35.5°F)

*Plus bypass flow of approximately 70,022 l/min (18,500 gal/min) per unit.
†Approximate values.

Table 4.6 Drainage areas and gaged river flow of streams tributary to Delaware River and Bay*

River or stream	Drainage area km^2	Drainage area mi^2	Average discharge m^3/s	Average discharge ft^3/s	Average discharge m^3/min/km^2	Average discharge ft^3/s/mi^2
Delaware at Trenton	17,560	6,780	331.6	11,710	1.14	1.73
Crosswicks Creek	218	84	4.3	152	1.20	1.82
Neshaminy	544	210	7.5	265	0.83	1.26
Rancocas, North Branch	287	111	4.6	162	0.96	1.46
Schuylkill at Philadelphia	4,903	1,893	76.9	2,715	0.95	1.44
Chester Creek	158	61	2.2	78	0.83	1.27
Brandywine Creek	743	287	10.7	378	0.87	1.32
White Clay Creek	228	88	3.4	119	0.91	1.36
Maurice River	293	113	5.0	176	1.02	1.56
Total gaged (69.25%)	24,934	9,627	446.2	15,755	1.08	1.64
Ungaged area (30.75%)	11,067	4,273	198.5	7,010**	1.08	1.64
Total drainage area	36,001	13,900	644.7	22,765	1.08	1.64

*Drainage areas greater than 130 km^2 (50 mi^2).

**Ungaged area multiplied by 1.08 equals average m^3/min/km^2 (1.64 average ft^3/s/mi^2).

Table 4.7 Characteristics of tides at the Hope Creek site

Parameter	Meter	Foot
Tide range		
Mean	1.68	5.5
Diurnal	1.80	5.9
Spring	1.83	6.0
Mean low water	0.84	2.8
Mean sea level	0	0
Mean tide level	0	0
National geodetic vertical datum	-0.09	-0.30
Monthly 10% exceedance high tide	1.19	3.9
Monthly 10% exceedance low tide	-1.19	-3.9

Table 4.8 Water levels observed near Hope Creek site during windstorms

Date of event	Maximum water level above mean sea level	Minimum water level below mean sea level
August 23, 1933	2.4 m (8.0 ft)	-
November 25, 1950	2.6 m (8.5 ft)	-
March 6, 1962	2.3 m (7.5 ft)	-
January 26, 1939	-	-2.26 m (-7.4 ft)
December 31, 1962	-	-2.77 m (-9.1 ft)*

*Obtained from Environmental Science Services Administration, Reedy Point Tidal Benchmark Data Sheet, April 3, 1963.

Table 4.9 Delaware River water quality upstream of the Hope Creek site

Parameter	Delaware River at Trenton, New Jersey			Delaware River at Chester, Pennsylvania			Delaware River at Delaware Memorial Bridge, near Wilmington, Delaware		
	Maximum	Minimum	Mean	Maximum	Minimum	Mean	Maximum	Minimum	Mean
Specific conductance (μmhos/cm)	400	50	190	5,900	111	300	12,700	100	2,090
pH	10.2	5.3	8.1	8.7	5.5	6.7	9.3	4.2	6.6
Temperature (°C)*	34.0	0.0	12.27	33.0	Freezing	12.85	31.0	Freezing	15.0
Dissolved oxygen (mg/l)	18.4	4.0	11.24	13.5	0.0	6.4	13.7	0.0	7.5

*To convert to °F, multiply values shown by 1.8 and add 32.

Source: ER-OL, Table 2.4-5

Table 4.10 Habitat types on the power-line right-of-way to the New Freedom Switching Station*

Habitat	Hectare	Acre	Percent
Agricultural land	142	350	38
Brackish marsh	36	90	9.9
Oak-pine forest	93	229	25
Hardwood swamp forest	90	223	24
Pine-oak forest	6.6	16.2	1.8
Pitch pine lowland forest	1.4	3.4	0.4
Total	369	912	100

*Estimated from habitat maps in the report by McCormick and Jones (1973).

Table 4.11 Commercial fishery harvest for Delaware Bay from about Artificial Island and south to the ocean (kg)*

Year	Finfish	Shellfish	Total
1976	236,985	1,424,297	1,661,282
1977	360,248	941,960	1,302,208
1978	376,548	1,222,486	1,599,034
1979	479,970	1,202,373	1,682,343
1980	1,070,895	1,310,924	2,381,819
1981	3,299,775	5,960,352	9,260,127

*To convert to lb, multiply values shown by 2.20.

Source: ER-OL, Table 2.1-13.

Table 4.12 Recreational fishery harvest for Delaware Bay from about Artificial Island and south to the ocean (State of Delaware waters only)

Harvest	Kilogram	Pound
1974		
Weakfish	1,600,000	3,500,000
Bluefish	227,000	500,000
Shark	84,000	190,000
Flounder (summer)	69,500	153,000
Other fish	227,000	500,000
Total fish	2,207,500	4,843,000
Hard clam	1,000,000	2,200,000
Soft clam	820,000	1,800,000
Total clam	1,820,000	4,000,000
1974 total	4,027,500	8,843,000
1977		
Total	3,400,000	7,500,000

Source: ER-OL, Section 2.1.3.5.

5 ENVIRONMENTAL CONSEQUENCES AND MITIGATING ACTIONS

5.1 Résumé

This section evaluates changes in environmental impacts that have developed since the FES-CP was issued. Section 5.3.1.1 discusses the reduction of water withdrawn from the Delaware River as a result of the cancellation of Hope Creek Unit 2 and elimination of the cold water bypass system. Additionally, Section 5.3.1.1 includes a discussion of the potential upstream movement of saline water and the status of a supplemental water storage plan to compensate for consumptive water use during periods of low freshwater flow. Floodplain management is addressed in Section 5.3.3. Cancellation of Unit 2 has resulted in reduced aquatic resource impacts as noted in Section 5.5.2.

Information in Section 5.9 on radiological impacts has been revised to reflect knowledge gained since the FES-CP was issued. The material on plant accidents contains information that has been revised and updated, including actual experience with nuclear power plant accidents beyond design-basis accidents and the lessons learned from the accident at Three Mile Island Unit 2. Information on the environmental effects of the uranium fuel cycle, decommissioning, and operational monitoring programs is also provided.

5.2 Land Use

5.2.1 Plant Site

Impacts on land use at the plant site were evaluated in FES-CP, Sections 4.2.1 (construction) and 5.1.2 (operation). Current land use at the site is described in Section 4.2.2 of this document. The only land use on Artificial Island is for power generation. Of the 300 ha (741 acres) on the island committed to the HCGS site, approximately 62 ha (153 acres) will be devoted to permanent facilities. Remaining lands, some of which are being temporarily disturbed by construction activities, may return to or remain in their previous condition of unused, dredge-spoil marshland covered by the giant reed, *Phragmites*. Because of the cancellation of Unit 2, onsite land requirements will be less than those previously planned.

The only aspect of normal plant operation that has potential for land use impacts at the site is the emission of drift from the cooling tower and the deposition of this drift on agricultural, residential, industrial, and recreational lands near the site. Because drift deposition will be low, vegetation will not likely be affected (Section 5.5.1), and no land uses should be adversely impacted.

5.2.2 Transmission Lines

Effects of transmission lines on land use were evaluated in FES-CP, Sections 4.2.2 (construction) and 5.4 (operation). Clearing of forests for the HCGS transmission lines will remove about 191 ha (472 acres) of land from forestry uses. Cultivation and grazing can continue beneath the lines as they did before construction, except in the small areas occupied by tower bases.

Various aspects of power-line operating (for example, ozone production) have the potential for impact on land use through the effects on biota, as evaluated in Section 5.5.1.2. None of these possible impacts is expected to be of consequence to agricultural or other land uses in the area.

5.3 Water

5.3.1 Water Use Impacts

5.3.1.1 Surface Water

The Hope Creek Generating Station cooling and service water supply will be brackish water obtained from the Delaware Estuary. The station's closed-cycle circulating water system (cooling tower) and service water system withdraw water from the estuary at the rate of about 2.1 m^3/s (72 ft^3/s). The return of flow is about 1.3 m^3/s (47 ft^3/s) depending on cooling tower makeup water requirements. The regional use of the adjacent estuary water is mainly recreational, such as for water sports, fishing, and boating.

The public water supplies within a 24- to 40-km (15- to 25-mi) radius of the site are identified on Table 5.1 and Figure 5.1. The nearest surface public water supply is located near the town of Salem, New Jersey, 14.5 km (9 mi) northeast of the site. No surface drainage from the plant site could possibly affect this reservoir because of the distance involved, the intervening surface elevations (topography), and the fact that the plant is located on an island in the estuary.

The primary concern related to surface water use as expressed in the FES-CP (Section 5.1.4) was that consumptive water use by the plant could contribute to the upstream movement of saline water during drought periods. The Delaware River Basin Commission (DRBC) has determined that a minimum flow of 85 m^3/s (3,000 ft^3/s) must be maintained at Trenton, New Jersey, to maintain the saline front downstream of the Philadephia water intake area. The consumptive use of brackish water by the Hope Creek station is expected to contribute only slightly to the upstream migration of saltwater during periods of drought. In addition, cancellation of Unit 2 and elimination of the cold water bypass system have decreased by 61% (from approximately 321,000 l/min (84,800 gal/min) to approximately 125,000 l/min (33,000 gal/min)) the amount of water withdrawn from the Delaware River which would further reduce the station's already small effect on upstream movement of saline water during low-flow conditions. Water use during normal operation has been reduced from 1.2% to 0.6% of the freshwater river flow at Trenton, New Jersey, and from 0.04% to 0.02% of the average tidal flow past the plant.

Since issuance of the FES-CP, the DRBC has required the applicant to develop a supplemental water storage plan to compensate for consumptive water use when the freshwater flow at Trenton, New Jersey, is less than 85 m^3/s (3,000 ft^3/s). The applicant, in a joint effort with other Delaware River Basin electric utilities, filed an application with the DRBC for a storage reservoir on December 30, 1977. This application is currently under review.

5.3.1.2 Groundwater

All of the potable water used within a 8-km (5-mi) radius from the containment structure is from groundwater sources. The freshwater supply for the operation of the station comes from the production well supply system.

The Hope Creek station has two onsite wells that are 244 m (800 ft) deep and use groundwater from the Raritan and Magothy Formations. Production well HC-1 is located about 230 m (750 ft) north-northwest of the containment building, HC-2 is located about 365 m (1,200 ft) northeast of the containment building. The onsite wells will supply up to 192,000 l/day (50,700 gal/day) of freshwater for plant operation.

The DRBC, which regulates major water diversions within the Delaware Basin, has taken into account the groundwater conditions in the well areas and has approved operation of the wells. Once decommissioning of the dewatering systems takes place, the shallow aquifer flow regimes and water quality are expected to return to prepumping conditions. Depending on well and screen depth, all local wells should be recharged by the Delaware River, and wells further north of the site should be recharged by freshwater inflows.

The nearest publicly used (non-nuclear generating station) wells are about 5 km (3 mi) from the containment structure and generally use the shallower Mount Laurel-Wenonah Formation. There are no known public water supplies using groundwater from the Raritan-Magothy Formation within a distance of 14.5 km (9 mi). Only two public water supplies utilize this formation within a radius of 14.5 to 35 km (9 to 21.7 mi). Both are at such a distance that plant groundwater withdrawal will not impact them.

5.3.2 Water Quality

5.3.2.1 Surface Water

Delaware River water in the vicinity of the site is not used for domestic supplies, and its industrial use is limited to cooling applications. All nonradioactive liquid waste streams from the station will be treated before being discharged to the Delaware River, and these effluents must meet state and Federal effluent limitations. Discharges must be controlled to meet effluent limitations set by the Environmental Protection Agency (EPA) under the National Pollution Discharge Elimination System (NPDES) Permit, and the DRBC water quality requirements dictate that station discharges will not have an adverse impact on Delaware River water quality or on the water quality of other states.

The major effluents of concern are total residual chlorine, other chemicals in the cooling tower blowdown, discharges from the sanitary waste treatment system, and thermal additions. Recent EPA effluent limitation guidelines for chlorine in cooling tower blowdown allow discharge of free available chlorine with a maximum of 0.5 mg/l and an average of 0.2 mg/l. Discharge of free available and total residual chlorine is limited to a maximum of 2 hours per day. The applicant's original plan was to chlorinate imtermittently to meet these limitations; however, recent biofouling experienced at Salem Generating Station indicates that continuous chlorination may be necessary at Hope Creek. In response to the requirements of the Atomic Safety and Licensing Board (October 1974) in the Hope Creek construction permit, and the perceived need to provide continuous chlorination regimes at Hope Creek, the applicant has initiated a chlorination study to determine if continuous chlorination is required and the levels of chlorine in the station discharge that will result from biofouling control measures. The applicant indicates that a draft report on this study will not be available until mid-1985. Additionally, as indicated in Section 4.2.6.2, the applicant plans to dechlorinate the cooling tower blowdown as necessary to achieve a level of 0.5 mg/l maximum total residual chlorine.

Dilution by river and tidal flow of any free available chlorine released to the river along with the chlorine demand of river water should reduce the free chlorine levels below detectable limits. Chemicals in the cooling tower blowdown such as sulfuric acid, chromium, and iron are also regulated by the NPDES. Maximum and average concentrations of chromium and iron are limited to 0.2 mg/l and 1.0 mg/l, respectively. Dilution by river and tidal flow should further reduce these levels to maintain river water quality. Detectable amounts of other chemicals in the cooling tower blowdown are not permitted. Compliance with water quality standards for potentially toxic components of the station discharge, such as residual chlorine, are discussed in Section 5.5.2.4.

Thermal effluent limitations imposed by the DRBC require that the net temperature increase of the Delaware River should not be greater than 2.2°C from September to May and not greater than 0.8°C from June to August. These limitations only apply within a heat dissipation area no larger than 1,070 m (3,500 ft) from the point where the effluent enters the river. Also, suspended solids, biochemical oxygen demand, and dissolved oxygen requirements of the DRBC as applied to waste water treatment effluents are necessarily restrictive enough to maintain the water quality integrity of the Delaware River near the site.

5.3.2.2 Groundwater

No impact on groundwater should occur because no wastes will be disposed of through underground injection. Most private wells are upstream and inland of the site, and because net river flow carries facility discharges away from well sites, offsite private wells should not be affected by discharges. In addition, dilution of plant effluents by river water should eliminate the possibility of impact on wells south of the site.

5.3.3 Other Hydrologic Impacts

Floodplain Aspects

Construction at the Hope Creek site had already begun at the time Executive Order 11988, Floodplain Management, was signed in May 1977. It is, therefore, the staff's conclusion that consideration of alternative locations for any structures identified as being in the floodplain is neither required nor practicable.

The floodplain is defined as the lowland and relatively flat areas adjoining inland and coastal waters, subject to a 1% or greater chance of flooding in any given year. For the Hope Creek/Salem site, the floodplain (shown in Figure 5.2) is the low lying area adjacent to the surrounding tidal shoreline to the east, south, and west of the plant. Flooding at the site would be caused by either intense precipitation and/or a storm surge caused by northeasters or hurricanes.

The 100-year flood was conservatively estimated to be 2.7 m (8.9 ft) mean sea level (MSL) using the Federal Insurance Administration's (FIA's) Flood Insurance Study for the Township of Lower Alloways Creek, Salem County, New Jersey, dated October 18, 1982. Table 5.2 shows a comparison between the 100-year flood level at the site and other floods either estimated or measured for the site and other nearby estuarine areas.

Areas inundated by the 100-year flood are shown on Figure 5.2, which also shows areas where site construction has encroached on the preconstruction 100-year floodplain. Only 30% of the preconstruction plant area was above the 100-year floodplain level. The remaining 70% of the plant areas was slightly below the 100-year floodplain level. This area was raised by fill to levels above the 100-year floodplain. The removal of this floodplain will have no measurable hydrologic effects on the flood level elsewhere. Furthermore, the plant has been designed for floods far more severe than the 100-year flood, up to and including probable maximum floods from storm surge, river basin flood, and precipitation runoff. The only postconstruction facilities in the floodplain at the Hope Creek site are the service water intake structure and the barge slip. The effect of these structures on the floodplain off the site will be insignificant. Additionally, the intake structure is flood protected for events well in excess of the 100-year flood level.

The staff considers that the effect of the presence or operation of the plant on the 100-year floodplain is negligible.

5.4 Air Quality

5.4.1 Fog and Ice

As stated in the FES-CP, atmospheric emissions from the natural draft cooling tower will consist primarily of waste heat and water vapor, resulting in persistent cloudlike plumes. The general conclusions of the FES-CP, with respect to atmospheric impacts resulting from cooling tower operation, remain unchanged. Visible plumes from the cooling tower will likely be longest during the winter and at night, and plume shadowing (decreasing the amount of solar radiation received at a point on the ground) is not expected to be significant. Some rime icing may occur on elevated structures above approximately 61 m (200 ft), although the combination of meteorological conditions required for significant icing (persistent wind direction, stable atmospheric conditions, and temperatures below freezing) are not very frequent. For example, the applicant estimates that meteorological conditions conducive to icing on the Delaware Memorial Bridge occur only 0.7% of the time during the winter. Cooling tower plume interaction with other airborne releases is not likely because of the disparities in release heights and plume rise. Overall, the impact of the cooling tower on climatic conditions will be negligible.

5.4.2 Other Emissions

As stated in the FES-CP, nonradioactive pollutants (for example, SO_2 and NO_x) produced by operation of emergency diesel generators and auxiliary boilers should not significantly degrade air quality in the vicinity of the plant. EPA Region II has determined that Hope Creek does not need a Prevention of Significant Deterioration (PSD) permit, which eliminates the need to perform quantitative atmospheric dispersion modeling for releases from the emergency diesel generators and auxiliary boilers. The applicant also has committed to New Jersey Department of Environmental Protection to operate no more than two of the three auxiliary boilers at one time.

5.5 Terrestrial and Aquatic Resources

5.5.1 Terrestrial Resources

5.5.1.1 Cooling Tower Operation

Cooling towers have the potential to cause the following impacts on terrestrial resources:

(1) Increased ground-level fogging and icing resulting from water droplets in the cooling tower drift may interfere with highway traffic.

(2) Plumes and enhanced cloud formation may cause increased precipitation and ground-level shading.

(3) Vegetation may be adversely affected by increased icing or by the salts contained in the drift deposited on soils or directly on foliage.

(4) Wildlife may be affected by the impact of drift on vegetation and, in the case of birds, collision with towers.

Impacts of cooling towers have been addressed in many published studies (Carson, 1976; Talbot, 1979; and Wilber and Webb, 1983). These studies and experience with hundreds of natural draft cooling towers (the majority are located in Great Britain (Carson, 1976)) that have operated for many years without significant impact suggest that operation of the Hope Creek cooling tower will have no significant impact on terrestrial resources. A survey of literature on cooling towers conducted by the NRC staff for the purposes of the Hope Creek review surfaced no studies that detected significant impacts from the operation of natural draft cooling towers. Increases in ground-level fogging, precipitation, icing, cloud formation and associated shading, and effects on productivity of vegetation and crops at Hope Creek will, therefore, be inconsequential. The fact that the nearest agricultural and residential land is located several kilometers from the site further minimizes the potential for impact.

The primary potential impact on terrestrial resources is reduced productivity of native, exotic, and agricultural plants because of the deposition of cooling tower drift on foliar surfaces and soils. Studies indicate that the drift deposition rate must be above 100 kg/ha/year (90 lb/acre/year) before agricultural plant productivity will be reduced (Mulchi and Armbruster, 1981; NUREG-0555). Death of plants would require much higher deposition rates. The natural draft cooling tower is predicted to result in an annual average deposition rate of less than 7.2 kg/ha/year (6.4 lb/acre/year) at a distance of 0.4 km (0.3 mi) from the tower. At the nearest farm, 5.6 km (3.5 mi) east of Hope Creek the annual average deposition rate is predicted to be 0.22 kg/ha/year

(0.20 lb/acre/year) (New Jersey Department of Environmental Protection, 1980). Therefore, it is unlikely that any adverse impact will occur. The staff of the New Jersey Department of Environmental Protection also came to this conlusion after studying natural salt deposition rates, the predicted deposition rates associated with the Hope Creek cooling tower, and agricultural practices in the vicinity of the site (New Jersey Department of Environmental Protection, 1980).

Although some birds will collide with the cooling tower, unpublished surveys at existing cooling towers indicate that the number will be relatively small. Although scientific publications report that birds often collide with radio and TV towers, such reports for cooling towers are scarce.

5.5.1.2 Transmission System

The 500-kV transmission lines will produce small amounts of ozone and nitrogen oxides, electromagnetic fields, and corona noise, and will cause some bird mortality as a result of collision with structures and conductors. In addition, periodic cutting of vegetation for right-of-way maintenance will affect terrestrial biota.

The electromagnetic fields associated with the lines can cause an induced current in nearby grounded objects and the buildup of voltage on nearby ungrounded objects such as automobiles, electric or nonelectric fences, and rain gutters on buildings. A person or animal that contacts such an object could receive a shock and experience a painful sensation at the point of contact. The strength of the shock depends on the electric field strength, the size of the object, and how well both the object and the person or animal are insulated from the ground.

With constant contact, a person could experience a current level of up to 5 mA (milliamps) under worst-case conditions (that is, a large well-insulated vehicle parked under power lines and a well-grounded person for a 500-kV line). In normal situations, however, conditions that would result in the worst case are rare, and induced currents should be much less than 5 mA. The average let-go level has been estimated as 9 mA for men, 6 mA for women, and 5 mA for children. A current of 4.5 mA has been estimated as a safe let-go level for children (Lee et al., 1982).

A spark discharge may also occur just before contact is made with the object. This discharge is similar to the static discharge shock a person can experience after walking across a carpet and then touching a metal door knob, although in the case of transmission lines the shock can occur repeatedly at a high frequency (60 times per second) as long as there is a slight space between the person and the object. The energy in a spark discharge can be harmful at levels above 25 J (joules). For 500-kV transmission lines, in the worst case (that is, for a large vehicle parked under a power line), the energy in a discharge would usually be less than 30 mJ (millijoules) (Lee et al., 1982). To mitigate potential problems with shocks involving induced currents or spark discharges, the National Electric Safety Code (NESC) suggests that adequate grounding for objects near the transmission lines be provided (such that, induced currents should not exceed 5 mA). The applicant expects that electric field strength will typically be a maximum of 5 kV/m (1.5 kV/ft) beneath the lines and <2 kV/m (0.6 kV/ft) at the edge of the transmission line right-of-way (ROW) (ER-OL, Section 5.5.4). These values are below the NESC guidelines, that is, less than 7.5 kV/m (2.3 kV/ft) maximum within the ROW and less than 2.6 kV/m (0.79 kV/ft) maximum at the edge of the ROW.

Extensive experience with high voltage lines up to 765 kV and the overall results of numerous studies provide little evidence that transmission lines pose a long-term biological hazard (Lee et al., 1982). With few exceptions, 30 reviews of the literature on biological effects of electromagnetic fields concluded that power-line electromagnetic fields have not been shown to cause harmful effects in plants, animals, or people (Lee et al., 1982). The applicant has encountered no significant environmental problems associated with electromagnetic fields from the 500-kV lines (ER-OL, Section 5.5.4), and should be able to operate the Hope Creek power lines without significant effect. If problems do arise, it is likely that they can be easily eliminated by modifications of the lines or rights-of-way.

Noise, radio and TV interference, and production of ozone and nitrogen oxides result from corona phenomena (electrical discharges in the air around the conductors) associated with the operation of power lines. Corona increases with voltage, adverse weather conditions (for example, high humidity or fog), and the amount of surface irregularities (for example, scratches or dirt particles) on the conductors. Modern-day power lines are designed to limit the occurrence of corona to relatively low levels. Corona noise and possibly some radio and TV interference will be noticeable near the lines. Under adverse weather conditions, a 500-kV line (double circuit) increases the ambient ozone concentration at ground level under the lines by no more than 0.0022 parts per million (ppm), compared with an average ambient ozone concentration of 0.01 to 0.03 ppm in rural areas (Lee et al., 1982) and a national primary air quality standard of 0.12 ppm. Therefore, ozone production by the power lines is expected to be inconsequential. Production of nitrogen oxides is even less significant (Lee et al., 1982).

Bird mortality will result from collisions with towers and conductors. This mortality cannot be accurately quantified, although Stout and Cornwell (1976) estimated that only 0.07% of the mortality of waterfowl from causes other than hunting resulted from collision. Bird collisions with lines are most evident where the lines pass through areas of bird concentration, such as river crossings and wetland areas frequented by large numbers of waterfowl. No great concentrations of waterfowl are known to occur along the Hope Creek lines, although Salem County supports a large waterfowl population and, of New Jersey's 20 counties, ranked fourth in the fall waterfowl harvests from 1971 through 1980 (Carney et al., 1983). Because the new line will be adjacent to an existing line, the additional impacts should be minimal.

The power line ROW will be managed by periodic removal or trimming of tall-growing trees and shrubs within and at the edge of the ROW. The practice of trimming and removal is in widespread use among the utilities and should have no unexpected or serious impacts. Population numbers of some wildlife species occurring on the ROW may fluctuate with the cutting cycle, with the low numbers occurring during the first year after each cutting. Pesticides or herbicides will not be used (ER-OL, Section 5.5.4), which minimizes the potential for significant impact.

5.5.2 Aquatic Resource Impacts

The impacts of operation of Hope Creek on aquatic resources of the Delaware River were considered and assessed in the FES-CP (Sections 5.1.3 and 14.0) and in the Atomic Safety and Licensing Board's (ASLB's) Initial Decision of October 25, 1974 (LBP-74-79, 8 AEC 745 (1974)). Both of those assessments examined the impact potential to the Delaware River from operation of two units. Construction of Hope Creek Unit 2 has been cancelled since those CP-stage reviews, thus reducing the makeup and blowdown volumes. Other design changes have been made and are described in Section 4.2.4 of this report. The sections that follow update the assessment of operational impacts resulting from the recent design changes and new resource information collected since the CP-stage reviews.

The ASLB initial decision (LBP-74-79) authorizing CP issuance found the aquatic resource impacts that could result from the operation of the Hope Creek station acceptable. However, since the FES-CP was issued, several years of ecological monitoring data from the Delaware River and operational experience at the adjacent Salem Generating Station warrant a re-examination of potential ecological impacts that could occur as a result of operation of Hope Creek.

5.5.2.1 Entrainment Impacts

Cancellation of Unit 2 since issuance of the FES-CP will result in a reduction of normal makeup water use from 1.2% to 0.6% of the freshwater flow at Trenton, New Jersey, and a reduction from 0.04% to 0.02% of the average tidal flow past the plant. The FES-CP concluded that a 0.04% removal of water from the Delaware River would not have a "noticeable effect on the well-being of the regional aquatic ecosystem." A reduction from 0.04% to 0.02% in removal of Delaware River water for makeup water further supports this conclusion.

Even though removal of only 0.02% of the tidal flow seems insignificant in relation to total river volume, the magnitude of entrainment impacts on the fishery resources of the Delaware River Estuary, however, will be addressed. In this section the impacts of phytoplankton, zooplankton, and ichthyoplankton entrainment on potential fishery production is estimated by comparing projected fishery production losses with commercial finfish landings.

5.5.2.1.1 Phytoplankton Entrainment

To estimate potential fishery production that could be lost because of entrainment of phytoplankton, the following calculations were performed and assumptions made

(1) The average annual chlorophyll a density in the Delaware River near the site is 9.1 mg/m^3 (Public Service Electric and Gas Company (PSE&G), 1980a). This value represents the annual average chlorophyll a concentration that was measured monthly at 10 stations near the site for each year from 1974 to 1976.

(2) The makeup water withdrawal rate from the Delaware River is 6.52×10^7 m^3/year (2.30×10^9 ft^3/year).

(3) The carbon to chlorophyll a ratio is 40:1 (Vollenwieder, 1971); the wet weight of photoplankton biomass to carbon ratio is 10:1 (Lind, 1979); therefore, the phytoplankton biomass to chlorophyll a ratio is 400:1.

(4) The average food web conversion of phytoplankton biomass to fishery production is 0.0058 (Harvey, 1950).

Potential fishery production lost annually because of entrainment of phytoplankton is, therefore, calculated as:

(9.1 mg chlorophyll a/m^3) (6.52 x 10^7 m^3/year)(400)(0.0058) = 1,377 kg (3,030 lb).

This calculation conservatively assumes that all of the phytoplankton killed would have been consumed by organisms in the estuarine food chain leading to fish. Entrained organisms do not constitute an irretrievable loss to the aquatic ecosystem, however, because they provide nutrition for decomposer organisms and these decomposers recycle nutrients back into the water for utilization by primary producers. Entrained biota can also serve as a source of detrital food material for many estuarine organisms that depend on detritus as their primary source of nutrition. This analysis also assumes that no compensatory mechanisms are occurring in the food chain.

5.5.2.1.2 Zooplankton Entrainment

To estimate potential fishery production that could be lost as a result of entrainment of zooplankton, the following calculations were performed and assumptions made:

(1) The average annual microzooplankton density is 37,400 organisms/m^3 and the average annual macrozooplankton density is 48.0 animals/m^3. These values represent the annual average zooplankton densities measured monthly at several stations near the site from 1973 to 1977. Values were obtained from the applicant's annual reports for these years (PSE&G, 1973-1977) and from the ecological summary report (PSE&G, 1980a).

(2) Makeup water withdrawal rate from the Delaware River is 6.52 x 10^7 m^3/year. (2.30 x 10^9 ft^3/year).

(3) The average dry weight of an individual microzooplankter is 0.006 mg, and for a macrozooplankter it is 0.25 mg (Heinle, 1966; Lindsay and Morrison, 1974). The dry weight to wet weight ratio is 0.20.

(4) The average food chain conversion efficiency of zooplankton biomass to fishery production is 0.017 (Clark, 1946; Harvey, 1950).

Potential fishery production lost annually as a result of entrainment of zooplankton is calculated as:

(1) Microzooplankton = 37,400/m^3 (6.52 x 10^7 m^3/year)(0.030 mg wet wt.) (0.017) = 1,245 kg/year (2,739 lb/year).

(2) Macroplankton = 48/m^3 (6.52 x 10^7 m^3/year)(1.25 mg wet wt.)(0.017) = 67 kg/year (146 lb/year).

(3) Total zooplankton = 1,245 kg + 67 kg = 1,312 kg/year (2,892 lb/year) of potential fishery production lost.

5.5.2.1.3 Ichthyoplankton Entrainment

Calculation of the potential fishery production that could be lost as a result of entrainment of fish eggs and larvae required the following:

(1) Determination of the annual average density of fish eggs and larvae for the major fish species entrained from 1974 to 1977. These values were obtained from the applicant's annual reports (PSE&G, 1973-1977) and the ecological summary report (PSE&G, 1980a).

(2) Calculation of the number of eggs and larvae entrained annually by the plant was determined by multiplying the annual makeup water use (6.52 x 10^7 m^3 (2.30 x 10^9 ft^3)) by the average annual densities each year for each major species.

(3) Assumption that the natural mortality rate of fish eggs is 90%, fish larvae is 99.79%, and young-of-the year fish (to age 1) is 30% (NRC, 1976).

(4) Determination of the average weight of the major species during their first year of life (PSE&G, 1980b, Table 5.19).

Average annual densities of eggs for the major species of anchovy, weakfish, silversides, and other species during 1974 to 1977 are shown in Table 5.3. The average number entrained that could have survived if not entrained and the potential annual fish production lost are also presented in this table.

Anchovy is the species most impacted by entrainment of eggs with 130 kg (287 lb) of potential production lost compared with only 1.0 kg (2 lb) or less for the other species. Entrainment of larvae, however, represents a greater impact on potential fishery production with potential anchovy losses of 873 kg (1,925 lb) to only 5 kg (9-10 lb) for silversides and Atlantic Croaker (Table 5.4). When entrainment of fish eggs and larvae is considered together, total potential loss for anchovy is about 1,003 ± 1,379 kg (2,212 ± 3,040 lb), weakfish 28 ± 51 kg (62 ± 113 lb), goby 61 ± 35 kg (134 ± 78 lb), and the remaining species less than 10 ± 9 kg (21 ± 20 lb) (Table 5.5).

5.5.2.1.4 Total Entrainment Impacts

The total potential fishery production lost as a result of entrainment of phytoplankton, zooplankton, and ichthyoplankton is conservatively estimated at 3,801 kg (8,380 lb). Of this total, ichthyoplankton contributed 1,112 kg (2,452 lb), zooplankton 1,312 kg (2,892 lb), and phytoplankton 1,377 kg (3,036 lb) (Table 5.6). This total potential production loss can be placed into perspective in relation to commercial fishery landings within 0-80 km (0-50 mi) of the site. The annual average commercial finfish catch from 1976 to 1981 within 0-80 km of the site was 974,000 kg (2,142,000 lb) (ER-OL, Table 2.1-13). A potential fishery production loss of 3,801 kg (8,380 lb) represents only 0.39% of this commercial catch. Weakfish eggs and larvae were the only species entrained of commercial or economic importance. Average commercial catch of weakfish within 0-80 km of the site was 536,364 kg (1.18 x 10^6 lb) (ER-OL, Table 2.1-13); therefore, the potential entrainment loss of 51 kg (113 lb) is only 0.009% of the commerical weakfish catch. Because the potential fish production losses attributable to entrainment appear to be such a small percentage of the commercial catch, and thus an even smaller percentage of the total standing crop available in the area (Gulland, 1970; Adams et al., 1983), the staff concludes that entrainment of aquatic organisms as a result of the operation of the Hope Creek station would not appear to adversely impact fishery populations.

5.5.2.2 Impingement Impacts

The FES-CP and the ASLB (LBP-74-79) concluded that impingement impacts resulting from operation of the Hope Creek station should not have a detectable influence on the biota of the Delaware River. However, because the service water intakes at Hope Creek and the adjacent operational Salem Generating Station are similar with respect to location and design, this presents a unique opportunity to assess potential impingement impacts at Hope Creek.

The applicant conducted an intensive impingement monitoring study at the Salem station from April 1977 through December 1978, and the results and conclusions from this study are presented in a comprehensive report (PSE&G, 1980b). During 1977, approximately 13×10^6 fish weighing a total of 52,600 kg (116,000 lb) were estimated to have been impinged; during 1978, about the same number and approximately 40,000 kg (88,200 lb) of fish were impinged (Table 5.7). During 1977, 52% of all fish collected in the impingement samples were alive; during 1978, 61% survived impingement. The survival rate of impinged fish varied considerably, however, depending on the species. Of the commercially or recreationally important species, about 3,000 (6,600 lb) and 8,000 kg (17,600 lb) of white perch were estimated to have been impinged in 1977 and 1978, respectively. Estimated weight of weakfish impinged was projected to be about 7,000 kg (15,400 lb) and 11,000 kg (24,200 lb) in 1977 and 1978, respectively (Table 5.7). During 1977, about 19,000 kg (42,000 lb) of blue crabs were estimated to have been impinged; during 1978, approximately 9,000 kg (19,800 lb) were estimated to have been impinged (Table 5.7).

The total number and weight of fish impinged at Hope Creek are expected to be considerably lower than the amounts impinged at Salem because:

(1) At Salem the intake velocity at the trash racks is about 30.5 cm/s (1 ft/s); velocity through the traveling screens at Hope Creek is about 12 cm/s (0.40 ft/s). The lower velocity at Hope Creek should allow more and smaller fish to escape the currents in front of the screens.

(2) The total water volume withdrawn from the Delaware River by Salem is 140 m^3/s (4,950 ft^3/s); Hope Creek will only require 2.1 m^3/s (75 ft^3/s). If the magnitude of fish impingement can be related in a linear manner to volume of water withdrawn (this should be a reasonable assumption when the flow velocities of the respective water masses are similar), then the Hope Creek plant should impinge less than 10% of the fish impinged at Salem on the basis of consideration of water volumes alone.

Modifications in the intake design of Hope Creek since the FES-CP was published include a fish rescue system with fish buckets containing water as a part of the screen construction, a low-pressure (20-psi) fish removal spray, and troughs to return impinged organisms to the river. Impingement studies at Salem have shown that at least 50% of the impinged fish survive and are returned to the river alive. What is unknown, however, is the percentage of alive fish returned that eventually die because of increased susceptibility to predation, disease, and so forth, as a consequence of the impingement experience.

In addition to the fish rescue system, changes in the design of the intake structure since the CP stage should also aid in minimizing potential impingement problems. The intake structure will be parallel to and flush with the shoreline. This design should reduce impingement relative to impingement rates experienced at power stations that have intake canals leading to the intake structures. Intake canals tend to entrap fish and induce fatigue from constant swimming against the currents in these canals. An intake structure flush with the shoreline should increase the possibility of fish escaping impingement once they encounter the zone of influence of the intake currents.

Impingement at Hope Creek should be considerably less than the impingement experience encountered at Salem because of (1) the lower intake velocity at Hope Creek and (2) the much reduced water withdrawal from the Delaware River. Salem removes about 66 times more cooling water per unit time from the Delaware River than does the Hope Creek plant. Impingement, therefore, should be significantly less at Hope Creek.

On the basis of the intake-related comparisons between Hope Creek and Salem and the impingement experience documented at Salem, the staff estimates that impingement impacts at Hope Creek could be 10% or less than the impacts currently experienced at Salem. Given this percentage, Hope Creek could be expected to impinge less than 1,500 kg (3,300 lb) of weakfish per year and less than 2,000 kg (4,400 lb) of blue crabs per year. In terms of the commercial fisheries within 0-80 km of the site, these impingement rates represent less than 0.5% of the weakfish and blue crab fishery. Impingement of organisms due to operation of Hope Creek should not significantly impact important fishery populations. This supports the findings of the FES-CP and the initial decision of the ASLB regarding aquatic resource impacts resulting from operation of Hope Creek.

5.5.2.3 Thermal Impacts

In conjunction with cancellation of Unit 2, several changes have occurred in the discharge water system since the FES-CP was published. The more notable of these changes include (1) reduction in the volume of water discharged to the Delaware River, (2) a change in the location of the discharge pipe from 300 m (985 ft) upstream of the intake structure to 160 m (525 ft) upstream, and (3) placement of the discharge point 3 m (10 ft) off shore rather than 60 m (197 ft) off shore.

The major change that relates directly to thermal impacts on aquatic biota is a reduction in the amount of discharge water (for one unit) from 114,000 l/min (30,120 gal/min) to 80,900 l/min (21,360 gal/min), a 29% decrease. More importantly, in terms of potential ecological effects, the heat rejection rate has decreased from 416×10^6 Btu/hr to 216×10^6 Btu/hr, which is a reduction of 48% in the total amount of heat discharged to the Delaware River. On the basis of a heat rejection rate of 416×10^6 Btu/hr, the FES-CP concluded that "there will be no discernible far-field temperature rise of the river as a result of this discharge." Therefore, a 48% reduction in the amount of heat released to the river as a result of the cancellation of Unit 2 further supports the conclusion that aquatic organisms in the Delaware River should not experience adverse impacts because of thermal discharges from Hope Creek.

Since the FES-CP was published, the applicant has changed the point of thermal discharge from 60 m (197 ft) to 3 m (10 ft) off shore. With elimination of the cold-water bypass, the applicant was concerned that discharge flow might not be high enough to prevent siltation and eventual blocking of the discharge outlet would occur. The applicant was concerned (ER-OL) that raising the pipe off the river bottom to prevent siltation could cause a hazard to navigation in this shallow river. Movement of the point of discharge near shore and off the bottom eliminated the potential siltation and river traffic problems. The shoreline discharge should not adversely affect shore zone biota because of the tidal influence near the site. The width of the Delaware River narrows from about 8.1 km (4.9 mi) to approximately 3.1 km (1.9 mi) near the site, and as a result, in part, of this constriction, a large tidal amplitude of 2-2.6 m (6.6-8.5 ft) occurs in this area. This large semidiurnal tidal amplitude combined with the high tidal flow of 11,328 m^3/s (400,000 ft^3/s) should dilute, mix, and rapidly dissipate thermal discharges near the shore. This should minimize any potential thermal-related impacts on shore zone organisms such as small fish and benthos. Any thermal discharge along the shoreline and in shallow water near the site should have little effect on benthic organisms, or mobile epifauna such as blue crabs, in these areas. Substrate types in these zones are dominated by clay, mud, and shifting sands. The availability of suitable substrate limits distribution and abundance of benthic organisms within these areas where burrowing forms such as the polychaete worms dominate. Burrowing forms are generally less vulnerable than are attached species to adverse conditions because they can isolate themselves from the overlaying water.

In addition, any resident or migratory fish that come in contact with any portion of the thermal plume that has temperatures higher than their preference temperature should be able to readily avoid the plume. The applicant estimates that during winter at high slack tide a distance of 680 m (2,230 ft) is required for mixing in order to meet the 2.2°C (4.2°F) maximum temperature difference limitation of the DRBC. During the warmer months, the applicant estimates that the 2.2°C (4.2°F) standard within the 1,070-m (3,500-ft) limit will also be met.

Results of these thermal modeling simulations may underestimate, however, the extent of the plume during all seasons because these results represent the expected conditions for only one tidal excursion past the point of discharge. A larger thermal mixing zone than that presented by the applicant is expected because a given mass of water will be exposed to the discharge over several tidal cycles. This increased mixing-zone size should not create adverse impacts to the aquatic resources of the site vicinity or the river system beyond those predicted in the FES-CP or as a result of the design changes discussed above.

5.5.2.4 Chemical Discharges

Makeup water will be treated with chlorine before it is used in both the service and circulating water systems to control biofouling. The new EPA guidelines for chlorine residuals in cooling tower blowdown allow discharge of free available chlorine (FAC) at a maximum daily concentration of 0.5 mg/l FAC and a daily average concentration of 0.2 mg/l FAC. At the FES-CP stage the applicant planned to chlorinate intermittently to meet these limitations.

Compliance with EPA guidelines does not identify and quantify, however, the amount of toxic chlorine compounds that will be discharged. The kinds, number, and concentration of chemical species produced from chlorination vary with the amount of chlorine added, temperature, time of contact, and chemical characteristics of the water (Merkens, 1958).

Even if the concentration of free residual chlorine is maintained at 0.2 mg/l, the concentration of combined residual chlorine will vary with demand and the amount of chlorine added. This makes prediction of effects attributable to chlorine difficult in that the indirect chlorine impact may be caused to a large extent by the combined residual chlorine (chlorinated hydrocarbons and chloramines).

Toxicity to estuarine and marine biota influenced by the cooling water discharge will, in general, not be attributable to the products resulting from the chlorine demand, but rather to the residual chlorine (free and combined). Brungs (1973) has concluded that in most cases the concentration of total residual chlorine (without regard to type) is a satisfactory criterion to define acute toxicity. A measure of only free available chlorine does not take into account the presence of combined residual chlorine (for example, as chloramines), which is also toxic; a criterion based exclusively on concentration of free available chlorine is not, therefore, a satisfactory safeguard with regard to the toxicity to marine biota. The water quality criterion for total residual chlorine for marine organisms has been established at 0.001 mg/l (EPA, 1976).

The applicant has calculated that if the station discharge contains 0.5 mg/l of residual chlorine under conditions of least dilution (August during ebb tide), the concentration at the end of the 1,070-m (3,500-ft) mixing zone would be less than 0.01 mg/l. The staff concludes that, given the dilution rate presented by the applicant at the end of the mixing zone, the mortality of aquatic biota in the vicinity of the Hope Creek discharge zone would be confined to an acceptable level if the concentration of total residual chlorine in the cooling water discharge were limited to concentrations not exceeding 0.5 mg/l.

Releases of chemicals other than chlorine (see Sections 4.2.6 and 5.3.2) are not expected to measurably affect the aquatic biota in the vicinity of the site. Releases of chemicals such as sulfuric acid, chromium, and iron are regulated by the NPDES.

A study of copper concentrations in the intake and discharge zones of the adjacent Salem station was conducted by Harrison et al. (NUREG/CR-2965). In water samples collected near the plant total copper ranged from 6.7 μg/l to 10.6 μg/l and labile copper from 0.9 μg/l to 3.8 μg/l. Effluent limitations set by the DRBC and the New Jersey Department of Environmental Protection for copper in effluents are 0.20 mg/l (200 μg/l). The levels of copper found in the water near Salem are from 20 to 200 times lower than these limitations; therefore, no ecological impact is expected to result from release of copper from Hope Creek, assuming discharge levels will be similar to those at Salem.

5.6 Threatened and Endangered Species

5.6.1 Terrestrial

As described in Section 4.3.5.1, species listed as threatened or endangered by the Federal Government do not occur regularly or breed at the site. Therefore, operation of Hope Creek should have no significant impact on these species. The osprey, listed as endangered by the State of New Jersey, nests on transmission towers near the site. These birds began nesting on the towers during the presence of the Salem Generating Station and construction of Hope Creek. Because the birds are apparently accustomed to these facilities, they should not be adversely affected by Hope Creek operation. In spite of the electromagnetic fields emanating from the 500-kV lines, osprey nesting on the towers was apparently successful, because the nesting population increased from 3 to 12 active nests between 1974 and 1981 (ER-OL, Section 2.2.3).

5.6.2 Aquatic

The potential impact of the withdrawal of water by the Salem and Hope Creek stations on the endangered shortnose sturgeon, *Acipenser brevirostrum*, has been addressed in detail by Masnik and Wilson, members of the NRC staff (NUREG-0671). On the basis of the known life history of this species, the operating characteristics of the stations, and the status of the shortnose sturgeon in the Delaware River, Masnik and Wilson concluded that the operation of Hope Creek and Salem would not jeopardize the continued existence of the shortnose sturgeon.

The National Marine Fisheries Service (NMFS) reviewed the draft of that NRC biological assessment, along with other relevant data, and prepared its independent Biological Opinion under Section 7 of the Endangered Species Act of 1973, as amended (Leitzell, 1980). The NMFS concluded that the combined impact of the continued operation of Salem Unit 1, the future operation of Salem Unit 2 (which began operation in May 1982), and the completion and subsequent operation of Hope Creek Units 1 and 2 (Unit 2 subsequently was cancelled), is not likely to jeopardize the continued existence of shortnose sturgeon or to destroy or adversely modify the habitat that may be critical to it.

The potential impact to sea turtle populations should be minimal, since their occurrence in the site vicinity is relatively infrequent. Two of the three turtles entrapped at the Salem intake were dead before the entrapment, and a third was able to be removed and released alive (see Section 4.3.5.2). The low intake and discharge volume at Hope Creek should preclude significant involvement of sea turtles with either the plant's water withdrawal or effluent discharges.

5.7 Historic and Archeological Sites

As discussed in Section 4.3.6, the operation and maintenance of the plant and associated facilities are not expected to have any effect on any sites or properties eligible for or listed in the *National Register of Historic Places* (also see letter from R. W. Myers, Deputy State Historic Preservation Officer, to J. A. Shissias, PSE&G, dated March 6, 1984, in Appendix E).

5.8 Socioeconomic Impacts

The socioeconomic impacts of Hope Creek operation are discussed in Section 5.5 of the FES-CP. It is estimated currently that 397 operating workers will be required for the operation of the station. In addition, 140 contractor security employees will be required. Over 250 operating workers are already on site (ER-OL, Table 8.1-5). The remaining operating workers, who will be hired until 1987, are likely to reside in locations similar to those where existing plant employees live. Therefore, about 30% of the workers are expected to reside in Gloucester County, 20% in Salem County, 20% in Burlington County, 10% in Camden County, 10% in the eastern Delaware area, 5% in Cumberland County, and the remainder in the surrounding counties (letter dated May 29, 1984, Mittl (PSE&G) to Schwencer (NRC)). Because of the relatively small number of workers required to operate the station, the impact on the communities in which they will reside and on the traffic is expected to be minimal.

The annual payroll for the operating workers is projected to be $18.56 million (1983 dollars) with the annual payroll for the contractor security workers projected to be $7 million (1984 dollars). Table 5.8 presents the estimated gross receipts and franchise state taxes and the local real estate taxes that will result from operation of the station. The projected dollar amounts are provided for the first 5 years of operation.

5.9 Radiological Impacts

5.9.1 Regulatory Requirements

Nuclear power reactors in the United States must comply with certain regulatory requirements in order to operate. The permissible levels of radiation in unrestricted areas and of radioactivity in effluents to unrestricted areas are recorded in 10 CFR 20, "Standards for Protection Against Radiation." These regulations specify limits on levels of radiation and limits on concentrations of radionuclides in the facility's effluent releases to the air and water (above natural background). The radiation protection standards of 10 CFR 20 specify limitations on whole-body radiation doses to members of the general public in unrestricted areas at three levels: 500 mrems in any calendar year, 100 mrems in any 7 consecutive days, and 2 mrems in any 1 hour. These limits are consistent with national and international standards in terms of protecting public health and safety.

In addition to the Radiation Protection Standards of 10 CFR 20, there are recorded in 10 CFR 50.36a license requirements that are to be imposed on licensees in the form of Technical Specifications on Effluents from Nuclear Power Reactors to keep releases of radioactive materials to unrestricted areas during normal operations, including expected operational occurrences, as low as is reasonably achievable (ALARA). Appendix I to 10 CFR 50 provides numerical guidance on dose-design objectives for light-water reactors (LWRs) to meet this ALARA requirement. Applicants for permits to construct and for licenses to operate an LWR shall provide reasonable assurance that the following calculated dose-design objectives will be met for all unrestricted areas: 3 mrems/year to the total body or 10 mrems/year to any organ from all pathways of exposure from liquid effluents; 10 mrads/year gamma radiation or 20 mrads/year beta radiation air dose from gaseous effluents near ground level - and/or 5 mrems/year to the total body or 15 mrems/year to the skin from gaseous effluents; and 15 mrems/year to any organ from all pathways of exposure from airborne effluents that include the radioiodines, carbon-14, tritium, and the particulates.

Experience with the design, construction, and operation of nuclear power reactors indicates that compliance with these design objectives will keep average annual releases of radioactive material in effluents at small percentages of the limits specified in 10 CFR 20 and, in fact, will result in doses generally below the dose-design objective values of Appendix I to 10 CFR 50. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to ensure that the public is provided a dependable source of power, even under unusual operating conditions that may temporarily result in releases higher than such small percentages but still well within the limits specified in 10 CFR 20.

In addition to the impact created by facility radioactive effluents as discussed above, within the NRC policy and procedures for environmental protection described in 10 CFR 51, there are generic treatments of environmental effects of all aspects of the uranium fuel cycle. These environmental data have been summarized in Table S-3 (Table 5.20) and are discussed later in this report in Section 5.10. In the same manner the environmental impact of transportation of fuel and waste to and from an LWR is summarized in Table S-4 (Table 5.10) and presented in Section 5.9.3.1.2 of this report.

Recently an additional operational requirement for uranium fuel cycle facilities including nuclear power plants was established by the Environmental Protection Agency in 40 CFR 190. This regulation limits annual doses (excluding radon and daughters) for members of the public to 25 mrems total body, 75 mrems thyroid, and 25 mrems other organs from <u>all</u> fuel-cycle facility contributions that may impact a specific individual in the public.

5.9.2 Operational Overview

During normal operations of Hope Creek, small quantities of radioactivity (fission, corrosion, and activation products) will be released to the environment. As required by NEPA, the staff has determined the estimated dose to members of the public outside of the plant boundaries as a result of the radiation from these radioisotope releases and relative to natural-background-radiation dose levels.

These facility-generated environmental dose levels are estimated to be very small because of both the plant design and the development of a program that will be implemented at the facility to contain and control all radioactive emissions and effluents. Radioactive-waste management systems are incorporated into the plant and are designed to remove most of the fission-product radioactivity that is assumed to leak from the fuel, as well as most of the activation and corrosion-product radioactivity produced by neutrons in the reactor-core vicinity. The effectiveness of these systems will be measured by process and effluent radiological monitoring systems that permanently record the amounts of radioactive constituents remaining in the various airborne and waterborne process and effluent streams. The amounts of radioactivity released through vents and discharge points to areas outside the plant boundaries are to be recorded and published semiannually in the Radioactive Effluent Release Reports for the facility.

Airborne effluents will diffuse in the atmosphere in a fashion determined by the meteorological conditions existing at the time of release and are generally dispersed and diluted by the time they reach unrestricted areas that are open to the public. Similarly, waterborne effluents will be diluted with plant waste water and then further diluted as they mix with the Delaware River beyond the plant boundaries.

Radioisotopes in the facility's effluents that enter unrestricted areas will produce doses through their radiations to members of the general public in a manner similar to the way doses are produced from background radiations (that is, cosmic, terrestrial, and internal radiations), which also include radiation from nuclear-weapons fallout. These radiation doses can be calculated for the many potential radiological-exposure pathways specific to the environment around the facility, such as direct-radiation doses from the gaseous plume or liquid effluent stream outside of the plant boundaries, or internal-radiation-dose commitments from radioactive contaminants that might have been deposited on vegetation, or in meat and fish products eaten by people, or that might be present in drinking water outside the plant or incorporated into milk from cows at nearby farms.

These doses, calculated for the "maximally exposed" individual (that is, the hypothetical individual potentially subject to maximum exposure), form the basis of the staff's evaluation of impacts. Actually, these estimates are for a fictitious person because assumptions are made that tend to overestimate the dose that would accrue to members of the public outside the plant boundaries. For example, if this "maximally exposed" individual were to receive the total body dose calculated at the plant boundary as a result of external exposure to the gaseous plume, he/she is assumed to be physically exposed to gamma radiation at that boundary for 70% of the year, an unlikely occurrence.

Site-specific values for various parameters involved in each dose pathway are used in the calculations. These include calculated or observed values for the amounts of radioisotopes released in the gaseous and liquid effluents, meteorological information (for example, wind speed and direction) specific to the site topography and effluent release points, and hydrological information pertaining to dilution of the liquid effluents as they are discharged.

An annual land census will identify changes in the use of unrestricted areas to permit modifications in the programs for evaluating doses to individuals from principal pathways of exposure. This census specification will be incorporated into the Radiological Technical Specifications and satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR 50. As use of the land surrounding the site boundary changes, revised calculations will be made to ensure that the dose estimate for gaseous effluents always represents the highest dose that might possibly occur for any individual member of the public for each applicable foodchain pathway. The estimate considers, for example, where people live, where vegetable gardens are located, and where cows are pastured.

An extensive radiological environmental monitoring program, designed specifically for the environs of Hope Creek, provides measurements of radiation and radioactive contamination levels that exist outside of the facility boundaries both before and after operations begin. In this program, offsite radiation levels are continuously monitored with thermoluminescent detectors (TLDs). In addition, measurements are made on a number of types of samples from the surrounding area to determine the possible presence of radioactive contaminants that, for example, might be deposited on vegetation, be present in drinking water outside the plant, or be incorporated into cow's milk from nearby farms. The results for all radiological environmental samples measured during a calendar year of operation are recorded and published in the Annual Radiological Environmental Operating Report for the facility. The specifics of the final operational-monitoring program and the requirement for annual publication of the monitoring results will be incorporated into the operating license Radiological Technical Specifications for the Hope Creek facility.

5.9.3 Radiological Impacts From Routine Operations

5.9.3.1 Radiation Exposure Pathways: Dose Commitments

The potential environmental pathways through which persons may be exposed to radiation originating in a nuclear power reactor are shown schematically in Figure 5.3. When an individual is exposed through one of these pathways, the dose is determined in part by the amount of time he/she is in the vicinity of the source, or the amount of time the radioactivity inhaled or ingested is retained in his/her body. The actual effect of the radiation or radioactivity is determined by calculating the dose commitment. The annual dose commitment is calculated to be the total dose that would be received over a 50-year period, following the intake of radioactivity for 1 year under the conditions existing 20 years after the station begins operation. (Calculation for the 20th year, or midpoint of station operation, represents an average exposure over the life of the plant.) However, with few exceptions, most of the internal dose commitment for each nuclide is given during the first few years after exposure because of the turnover of the nuclide by physiological processes and radioactive decay.

There are a number of possible exposure pathways to humans that are studied to determine the impact of routine releases from the Hope Creek facility on members of the general public living and working outside of the site boundaries, and whether the releases projected at this point in the licensing process will in fact meet regulatory requirements. A detailed listing of these exposure pathways would include external radiation exposure from the gaseous effluents, inhalation of iodines and particulate contaminants in the air, drinking milk from a cow or eating meat from an animal that feeds on open pasture near the site on which iodines or particulates may have deposited, eating vegetables from a garden near the site that may be contaminated by similar deposits, and drinking water or eating fish caught near the point of discharge of liquid effluents.

Other less important pathways include: external irradiation from radionuclides deposited on the ground surface; eating animals and food crops raised near the site using irrigation water that may contain liquid effluents; shoreline, boating and swimming activities near lakes or streams that may be contaminated by effluents; drinking potentially contaminated water; and direct radiation from within the plant itself. Note that for the Hope Creek site there is no drinking water pathway of concern because the nearest private well is approximately 5 km (3 mi) from the plant and because the site is in the estuary portion of the Delaware River where the water is brackish.

Calculations of the effects for most pathways are limited to a radius of 80 km (50 mi). This limitation is based on several facts. Experience, as demonstrated by calculations, has shown that all individual dose commitments (>0.1 mrem/year) for radioactive effluents are accounted for within a radius of 80 km from the plant. Beyond 80 km the doses to individuals are smaller than 0.1 mrem/year, which is far below natural-background doses, and the doses are subject to substantial uncertainty because of limitations of predictive mathematical models.

The staff has made a detailed study of all of the above important pathways and has evaluated the radiation-dose commitments both to the plant workers and the general public for these pathways resulting from routine operation of the facility. A discussion of these evaluations follows.

5.9.3.1.1 Occupational Radiation Exposure for Boiling Water Reactors (BWRs)

Most of the dose to nuclear plant workers results from external exposure to radiation coming from radioactive materials outside of the body rather than from internal exposure from inhaled or ingested radioactive materials. Experience shows that the dose to nuclear plant workers varies from reactor to reactor and from year to year. For environmental-impact purposes, it can be projected by using the experience to date with modern BWRs. Recently licensed 1,000-MWe BWRs are operated in accordance with the post-1975 regulatory requirements and guidance that place increased emphasis on maintaining occupational exposure at nuclear power plants ALARA. These requirements and guidance are outlined primarily in 10 CFR 20, Standard Review Plan Chapter 12 (NUREG-0800), and RG 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

The applicant's proposed implementation of these requirements and guidelines is reviewed by the staff during the licensing process, and the results of that review are reported in the staff's Safety Evaluation Report. The license is granted only after the review indicates that an ALARA program can be implemented. In addition, regular reviews of operating plants are performed to determine whether the ALARA requirements are being met.

Average collective occupational dose information for 177 BWR reactor years of operation is available for those plants operating between 1974 and 1981. (The year 1974 was chosen as a starting date because the dose data for years prior to 1974 are primarily from reactors with average rated capacities below 500 MWe.) These data indicate that the average reactor annual collective dose at BWRs has been about 790 person-rems, although some plants have experienced annual collective doses averaging as high as 1,660 person-rems/year over their operating lifetime (NUREG-0713, Vol. 3). These dose averages are based on widely varying yearly doses at BWRs. For example, for the period mentioned above, annual collective doses for BWRs have ranged from 44 to 3,626 person-rems per reactor. However, the average annual dose per nuclear plant worker of about 0.8 rem (ibid) has not varied significantly during this period. The worker dose limit, established by 10 CFR 20, is 3 rems/quarter if the average dose over the worker lifetime is being controlled to 5 rems/year or 1.25 rems/quarter if it is not.

The wide range of annual collective doses experienced at BWRs in the United States results from a number of factors such as the amount of required maintenance and the amount of reactor operations and in-plant surveillance. Because these factors can vary widely and unpredictably, it is impossible to determine in advance a specific year-to-year annual occupational radiation dose for a particular plant over its operating lifetime. There may on occasion be a need for relatively high collective occupational doses, even at plants with radiation protection programs designed to ensure that occupational radiation doses will be kept ALARA.

In recognition of the factors mentioned above, staff occupational dose estimates for environmental impact purposes for Hope Creek are based on the more conservative estimate by the applicant rather than the assumption that the facility will experience the annual average occupational dose for BWRs to date. Thus the staff has projected that the collective occupational doses for Hope Creek will be 920 person-rems, but annual collective doses could average as much as twice this value over the life of the plant.

The average annual dose of about 0.8 rem per nuclear-plant worker at operating BWRs and PWRs has been well within the limits of 10 CFR 20. However, for impact evaluation, the NRC staff has estimated the risk to nuclear-power-plant workers and compared it in Table 5.9 to published risks for other occupations. Based on these comparisons, the staff concludes that the risk to nuclear-plant workers from plant operation is comparable to the risks associated with other occupations.

In estimating the health effects resulting from both offsite (see Section 5.9.3.2) and occupational radiation exposures as a result of normal operation of this facility, the NRC staff used somatic (cancer) and genetic risk estimators that are based on widely accepted scientific information. Specifically, the staff's estimates are based on information compiled by the National Academy of Sciences' Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR I). The estimates of the risks to workers and the general public are based on conservative assumptions (that is, the estimates are probably higher than the actual number). The following risk estimators were used to estimate health effects: 135 potential deaths from cancer per million person-rems and 258 potential cases of all forms of genetic disorders per million person-rems. The cancer-mortality risk estimates are based on the "absolute risk" model described in BEIR I. Higher estimates can be developed by use of the "relative risk" model along with the assumption that risk prevails for the duration of life. Use of the "relative risk" model would produce risk values up to about four times greater than those used in this report. The staff regards the use of the "relative risk" model values as a reasonable upper limit of the range of uncertainty. The lower limit of the range would be zero because there may be biological mechanisms that can repair damage caused by radiation at low doses and/or dose rates. The number of potential cancers would be approximately 1.5 to 2 times the number of potential fatal cancers, according to the 1980 report of the National Academy of Sciences' Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR III).

Values for genetic risk estimators range from 60 to 1,500 potential cases of all forms of genetic disorders per million person-rems (BEIR I). The value of

258 potential cases of all forms of genetic disorders is equal to the sum of the geometric means of the risk of specific genetic defects and the risk of defects with complex etiology.

The preceding values for risk estimators are consistent with the recommendations of a number of recognized radiation-protection organizations, such as the International Commission on Radiological Protection (ICRP, 1977), the National Council on Radiation Protection and Measurement (NCRP, 1975), the National Academy of Sciences (BEIR III), and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR, 1982).

The risk of potential fatal cancers in the exposed work-force population at the Hope Creek facility is estimated as follows: multiplying the annual plant-worker-population dose of about 920 person-rems given in FSAR Table 12.4-5 by the somatic risk estimator, the staff estimates that about 0.12 cancer death may occur in the total exposed population. The value of 0.12 cancer death means that the probability of 1 cancer death over the lifetime of the entire work force as a result of 1 year of facility operation is about 12 chances in 100. The risk of potential genetic disorders attributable to exposure of the work force is a risk borne by the progeny of the entire population and is thus properly considered as part of the risk to the general public.

5.9.3.1.2 Public Radiation Exposure

Transportation of Radioactive Materials

The transportation of "cold" (unirradiated) nuclear fuel to the reactor, of spent irradiated fuel from the reactor to a fuel reprocessing plant, and of solid radioactive wastes from the reactor to waste burial grounds is considered in 10 CFR 51.20. The contribution of the environmental effects of such transportation to the environmental costs of licensing the nuclear power reactor is set forth in Summary Table S-4 from 10 CFR 51.20, reproduced herein as Table 5.10. The cumulative dose to the exposed population as summarized in Table S-4 is very small when compared to the annual collective dose of about 60,000 person-rems to this same population or 26,000,000 person-rems to the U.S. population from background radiation.

Direct Radiation for BWRs

Radiation fields are produced around nuclear plants as a result of radioactivity within the reactor and its associated components, as well as a result of radioactive-effluent releases. Although the components are shielded, dose rates observed around BWR plants from these plant components have varied from undetectable levels to values on the order of 100 mrems/year at onsite locations where members of the general public were allowed. For newer BWR plants with a standardized design, dose rates have been estimated using special calculational modeling techniques. The calculated cumulative dose to the exposed population from such a facility would be much less than 1 person-rem/year per unit, insignificant when compared with the natural background dose.

Low-level radioactivity storage containers outside the plant are estimated to make a dose contribution at the site boundary of less than 0.1% of that due to the direct radiation described above.

Radioactive-Effluent Releases: Air and Water

Limited quantities of radioactive effluents will be released to the atmosphere and to the hydrosphere during normal operations. Plant-specific radioisotope-release rates were developed on the basis of estimates regarding fuel performance and descriptions of the operation of radwaste systems in the applicant's FSAR, and by using the calculative models and parameters described in NUREG-0016. These radioactive effluents are then diluted by the air and water into which they are released before they reach areas accessible to the general public.

Radioactive effluents can be divided into several groups. Among the airborne effluents, the radioisotopes of the fission product noble gases, krypton and xenon, as well as the radioactivated gas argon, do not deposit on the ground nor are they absorbed and accumulated within living organisms; therefore, the noble gas effluents act primarily as a source of direct external radiation emanating from the effluent plume. Dose calculations are performed for the site boundary where the highest external-radiation doses to a member of the general public as a result of gaseous effluents have been estimated to occur; these include the total body and skin doses as well as the annual beta and gamma air doses from the plume at that boundary location.

Another group of airborne radioactive effluents - the fission product radioiodines, as well as carbon-14 and tritium - are also gaseous but these tend to be deposited on the ground and/or inhaled into the body during breathing. For this class of effluents, estimates of direct external-radiation doses from deposits on the ground, and of internal radiation doses to total body, thyroid, bone, and other organs from inhalation and from vegetable, milk, and meat consumption are made. Concentrations of iodine in the thyroid and of carbon-14 in bone are of particular interest.

A third group of airborne effluents, consisting of particulates that remain after filtration of airborne effluents in the plant prior to release, includes fission products such as cesium and strontium and activated corrosion products such as cobalt and chromium. The calculational model determines the direct external radiation dose and the internal radiation doses for these contaminants through the same pathways as described above for the radioiodines, carbon-14, and tritium. Doses from the particulates are combined with those of the radioiodines, carbon-14, and tritium for comparison to one of the design objectives of Appendix I to 10 CFR 50.

The waterborne-radioactive-effluent constituents could include fission products such as nuclides of strontium and iodine; activation and corrosion products, such as nuclides of sodium, iron, and cobalt; and tritium as tritiated water. Calculations estimate the internal doses (if any) from fish consumption, from water ingestion (as drinking water), and from eating of meat or vegetables raised near the site on irrigation water, as well as any direct external radiation from recreational use of the water near the point of discharge.

The release rates for each group of effluents, along with site-specific meteorological and hydrological data, serve as input to computerized radiation-dose models that estimate the maximum radiation dose that would be received outside the facility via a number of pathways for individual members of the public, and for the general public as a whole. These models and the radiation-dose calculations are discussed in Revision 1 of RG 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR Part 50, Appendix I" (October 1977), and in Appendix B of this statement.

Examples of site-specific dose assessment calculations and discussions of parameters involved are given in Appendix D. Doses from all airborne effluents except the noble gases are calculated for individuals at the location (for example, the site boundary, garden, residence, milk cow, and meat animal) where the highest radiation dose to a member of the public has been established from all applicable pathways (such as ground deposition, inhalation, vegetable consumption, cow milk consumption, or meat consumption.) Only those pathways associated with airborne effluents that are known to exist at a single location are combined to calculate the total maximum exposure to an exposed individual. Pathway doses associated with liquid effluents are combined without regard to any single location, but they are assumed to be associated with maximum exposure of an individual through other than gaseous-effluent pathways.

5.9.3.2 Radiological Impact on Humans

Although the doses calculated in Appendix D are based primarily on radioactive-waste treatment system capability and are below the 10 CFR 50, Appendix I design objective values, the actual radiological impact associated with the operation of the facility will depend, in part, on the manner in which the radioactive-waste treatment system is operated. Based on its evaluation of the potential performance of the ventilation and radwaste treatment systems, the staff has concluded that the systems as now proposed are capable of controlling effluent releases to meet the dose-design objectives of Appendix I to 10 CFR 50.

Operation of the Hope Creek facility will be governed by operating license Technical Specifications that will be based on the dose-design objectives of Appendix I to 10 CFR 50. Because these design-objective values were chosen to permit flexibility of operation while still ensuring that plant operations are ALARA, the actual radiological impact of plant operation may result in doses close to the dose-design objectives. Even if this situation exists, the individual doses for the member of the public subject to maximum exposure will still be very small when compared to natural background doses (~100 mrems/year) or the dose limits (500 mrems/year - total body) specified in 10 CFR 20 as consistent with considerations of the health and safety of the public. As a result, the staff concludes that there will be no measurable radiological impact on any member of the public from routine operation of the Hope Creek facility.

Operating standards of 40 CFR 190, the Environmental Protection Agency's Environmental Radiation Protection Standards for Nuclear Power Operations, specify that the annual dose equivalent must not exceed 25 mrems to the whole body, 75 mrems to the thyroid, and 25 mrems to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials (radon and its daughters excepted) to the general environment from all uranium-fuel-cycle operations and radiation from these operations that can be expected to affect a given individual. The staff concludes that under normal operations the Hope Creek facility is capable of operating within these standards.

The radiological doses and dose commitments resulting from a nuclear power plant are well known and documented. Accurate measurements of radiation and radioactive contaminants can be made with very high sensitivity so that much smaller amounts of radioisotopes can be recorded than can be associated with any possible observable ill effects. Furthermore, the effects of radiation on living systems have for decades been subject to intensive investigation and consideration by individual scientists as well as by select committees that have occasionally been constituted to objectively and independently assess radiation dose effects. Although, as in the case of chemical contaminants, there is debate about the exact extent of the effects of very low levels of radiation that result from nuclear-power-plant effluents, upper bound limits of deleterious effects are well established and amenable to standard methods of risk analysis. Thus the risks to the maximally exposed member of the public outside of the site boundaries or to the total population outside of the boundaries can be readily calculated and recorded. These risk estimates for the Hope Creek facility are presented below.

The risk to the maximally exposed individual is estimated by multiplying the risk estimators presented in Section 5.9.3.1.1 by the annual dose-design objectives for total-body radiation in 10 CFR 50, Appendix I. This calculation results in a risk of potential premature death from cancer to that individual from exposure to radioactive effluents (gaseous or liquid) from 1 year of reactor operations of less than one chance in one million.* The risk of potential premature death from cancer to the average individual within 80 km (50 mi) of the reactors from exposure to radioactive effluents from the reactors is much less than the risk to the maximally exposed individual. These risks are very small in comparison to natural cancer incidence from causes unrelated to the operation of the Hope Creek facility.

Multiplying the annual U.S. general public population dose from exposure to radioactive effluents and transportation of fuel and waste from the operation of this facility (that is, 27 person-rems) by the preceding somatic risk estimator, the staff estimates that about 0.0036 cancer death may occur in the exposed population. The significance of this risk can be determined by comparing it to the natural incidence of cancer death in the U.S. population. Multiplying the estimated U.S. population for the year 2010 (~280 million persons) by the current incidence of actual cancer fatalities (~20%), about 56 million cancer deaths are expected (American Cancer Society, 1978).

For purposes of evaluating the potential genetic risks, the progeny of workers are considered members of the general public. Multiplying the sum of the U.S. population dose from exposure to radioactivity attributable to the normal annual operation of the plant (that is, 27 person-rems), and the estimated dose from occupational exposure (that is, 920 person-rems) by the preceding genetic risk estimators, the staff estimates that about 0.25 potential genetic disorder may occur in all future generations of the exposed population. Because BEIR III indicates that the mean persistence of the two major types of genetic disorders is about 5 generations and 10 generations, in the following analysis the risk of potential genetic disorders from the normal annual operation of the plant

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*The risk of potential premature death from cancer to the maximally exposed individual from exposure to radioiodines and particulates would be in the same range as the risk from exposure to the other types of effluents.

is conservatively compared with the risk of actual genetic ill health in the first 5 generations, rather than the first 10 generations. Multiplying the estimated population within 80 km of the plant (~5.4 million persons in the year 2010) by the current incidence of actual genetic ill health in each generation (~11%), about 3 million genetic abnormalities are expected in the first 5 generations of the 80-km population (BEIR III).

The risks to the general public from exposure to radioactive effluents and transportation of fuel and wastes from the annual operation of the facility are very small fractions of the estimated normal incidence of cancer fatalities and genetic abnormalities. On the basis of the preceding comparison, the staff concludes that the risk to the public health and safety from exposure to radioactivity associated with the normal operation of the facility will be very small.

5.9.3.3 Radiological Impacts on Biota Other Than Humans

Depending on the pathway and the radiation source, terrestrial and aquatic biota will receive doses that are approximately the same or somewhat higher than humans receive. Although guidelines have not been established for acceptable limits for radiation exposure to species other than humans, it is generally agreed that the limits established for humans are sufficiently protective for other species.

Although the existence of extremely radiosensitive biota is possible and increased radiosensitivity in organisms may result from environmental interactions with other stresses (for example, heat or biocides), no biota have yet been discovered that show a sensitivity (in terms of increased morbidity or mortality) to radiation exposures as low as those expected in the area surrounding the facility. Furthermore, at all nuclear plants for which radiation exposure to biota other than humans has been analyzed (Blaylock, 1976), there have been no cases of exposure that can be considered significant in terms of harm to the species, or that approach the limits for exposure to members of the public that are permitted by 10 CFR 20. Inasmuch as the 1972 BEIR Report (BEIR I) concluded that evidence to date indicated that no other living organisms are very much more radiosensitive than humans, no measurable radiological impact on populations of biota is expected as a result of the routine operation of this facility.

5.9.3.4 Radiological Monitoring

Radiological environmental monitoring programs are established to provide data where there are measurable levels of radiation and radioactive materials in the site environs and to show that in many cases no detectable levels exist. Such monitoring programs are conducted to verify the effectiveness of inplant systems used to control the release of radioactive materials and to ensure that unanticipated buildups of radioactivity will not occur in the environment. Secondarily, the environmental monitoring programs could identify the highly unlikely existence of releases of radioactivity from unanticipated release points that are not monitored. An annual surveillance (land census) program will be established to identify changes in the use of unrestricted areas to provide a basis for modifications of the monitoring programs or of the Technical Specifications conditions that relate to the control of doses to individuals.

These programs are discussed generically in greater detail in RG 4.1, Revision 1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants," and in the Radiological Assessment Branch Technical Position, Revision 1, November 1979, "An Acceptable Radiological Environmental Monitoring Program."*

5.9.3.4.1 Preoperational

The preoperational phase of the monitoring program should provide for the measurement of background levels of radioactivity and radiation and their variations along the anticipated important pathways in the areas surrounding the facility, the training of personnel, and the evaluation of procedures, equipment, and techniques. The applicant proposed a radiological environmental-monitoring program to meet these objectives in the ER-CP, and it was discussed in the FES-CP. The current program is presented in Section 6.1.5 of the applicant's ER-OL and is summarized here in Table 5.11.

The applicant states in Section 6.1.5 of the ER-OL that because of the proximity of the Hope Creek Generating Station and the Salem Generating Station, a common radiological environmental monitoring program is conducted for both stations. The preoperational program for the Salem Generating Station was conducted from 1968 until December 1976 when it became the operational program. It serves as the preoperational program for the Hope Creek Generating Station. When the Hope Creek Generating Station achieves its initial criticality, the program current then will become the operational program for both generating stations.

The staff has reviewed the preoperational environmental monitoring program of the applicant and finds that it is generally acceptable as presented. The NRC review of this area will continue up to the time of implementation of the operational environmental monitoring program.

5.9.3.4.2 Operational

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

The applicant states that the operational program will in essence be a continuation of the preoperational program described above. The proposed operational program will be reviewed before plant operation. Modification will be based on anomalies and/or exposure pathway variations observed during the preoperational program.

The final operational-monitoring program proposed by the applicant will be reviewed in detail by the staff, and the specifics of the required monitoring

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*Available from the Radiological Assessment Branch, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

program will be incorporated into the operating license Radiological Technical Specifications.

5.9.4 Environmental Impacts of Postulated Accidents

5.9.4.1 Plant Accidents

The staff has considered the potential radiological impacts on the environment of possible accidents at the Hope Creek station in accordance with a Statement of Interim Policy published by the Nuclear Regulatory Commission on June 13, 1980 (45 FR 40101-40104). The following sections discuss the staff's considerations and conclusions.

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

Next, Section 5.9.4.3 describes actual experience with nuclear power plant accidents and their observed health effects and other societal impacts. This is followed by a summary review in Section 5.9.4.4 of safety features of the Hope Creek station and of the site that act to mitigate the consequences of accidents.

The results of calculations of the potential consequences of accidents that have been postulated in the design basis are then given in Section 5.9.4.5. Also described are the results of calculations for the Hope Creek site using contemporary probabilistic methods and their inherent uncertainties to estimate the possible impacts and the risks associated with severe accident sequences of low probability of occurrence.

5.9.4.2 General Characteristics of Accidents

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

There are several features that combine to reduce the risk associated with accidents at nuclear power plants. Safety features provided for in design, construction, and operation constitute the first line of defense and are to a very large extent devoted to the prevention of the release of radioactive materials from their normal places of confinement within the plant. There are also a number of additional lines of defense that are designed to mitigate the consequences of failures in the first line. These safety features are designed taking into consideration the specific locations of radioactive materials within the plant; their amounts; their nuclear, physical, and chemical properties;

and their relative tendency to be transported into and for creating biological hazards in the environment. Descriptions of these features for Hope Creek may be found in the applicant's FSAR and in the staff's Safety Evaluation Report (SER), which is scheduled for publication in October 1984. The most important mitigative features are described in Section 5.9.4.4(1) below.

(1) Fission Product Characteristics

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

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

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

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

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

All of these radioactive materials exhibit the property of radioactive decay with characteristic half-lives ranging from fractions of a second to many days or years (see Table 5.12). Many of them decay through a sequence or chain of decay processes, and all eventually become stable (nonradioactive) materials. The radiation emitted during these decay processes is the reason that they are hazardous materials. As a result of radioactive decay, most fission products transmute into other elements. Iodines transmute into noble gases, for example, while the noble gases transmute into alkali metals. Because of this property, fission products which escape into the environment as one element may later become a contamination hazard as a different element.

(2) Exposure Pathways

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

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

(3) Health Effects

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

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

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

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

(4) Health Effects Avoidance

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

5.9.4.3 Accident Experience and Observed Impacts

As of February 1983, there were 76 commercial nuclear power reactor units licensed for operation in the United States at 52 sites, with power-generating capacities ranging from 50 to 1,180 megawatt electric (MWe). (Hope Creek is designed for 1,067 MWe). The combined experience with all these units represents approximately 700 reactor years of operation over an elapsed time of about 23 years. Accidents have occurred at several of these facilities (Oak Ridge National Laboratory, 1980; NUREG-0651). Some of these have resulted in releases of radioactive material to the environment ranging from very small fractions of a curie to a few million curies. None is known to have caused any radiation injury or fatality to any specific member of the public, nor any significant individual or collective public radiation exposure, nor any significant contamination of the environment. This experience base is not large enough to permit a reliable quantitative statistical inference for predicting accident probabilities. It does, however, suggest that significant environmental impacts caused by accidents are very unlikely to occur over time periods of a few decades.

Melting or severe degradation of reactor fuel has occurred in only one of these units, during the accident at Three Mile Island Unit 2 (TMI-2) on March 28, 1979. In addition to the release to the environment of a few million curies of noble gases, mostly xenon-133, it has been estimated that approximately 15 curies of radioiodine also were released to the environment at TMI-2 (NUREG/CR-1250). This amount represents a minute fraction of the total radioiodine inventory present in the reactor at the time of the accident. No other radioactive fission products were released to the environment in measurable quantity. It has been estimated that the maximum cumulative offsite radiation dose to an individual was less than 100 mrems (NUREG/CR-1250; President's Commission on the Accident at Three Mile Island, 1979). The total population exposure has been estimated to be in the range from about 1,000 to 5,300 person-rems (NUREG-0558). This exposure could produce between zero and one additional fatal cancer over the lifetime of the population. The same population receives each year from natural background radiation about 240,000 person-rems. Approximately a half-million cancers are expected to develop in this group over their lifetimes (NUREG/CR-1250; President's Commission on the Accident at Three Mile Island, 1979), primarily from causes other than radiation. Trace quantities (barely above the limit of detectability) of radioiodine were found in a few samples of milk produced in the area. No other food or water supplies were affected.

Accidents at nuclear power plants also have caused occupational injuries and a few fatalities, but none attributed to radiation exposure. Individual worker exposures have ranged up to about 5 rems as a direct consequence of reactor accidents (although there have been higher exposures to individual workers as a result of other unusual occurrences). However, the collective worker exposure levels (person-rem) from accidents are a small fraction of the exposures experienced during normal routine operations that average about 440 to 1,300 person-rems in a PWR and 790 to 1,660 person-rems in a BWR per reactor-year.

Accidents also have occurred at other nuclear reactor facilities in the United States and in other countries (Oak Ridge National Laboratory, 1980; Thompson and Beckerley, 1964). Because of inherent differences in design, construction, operation, and purpose of most of these other facilities, their accident record has only indirect relevance to current nuclear power plants. Melting of reactor fuel occurred in at least seven of these accidents, including the one in 1966 at the Enrico Fermi Atomic Power Plant, Unit 1. Fermi Unit 1 was a sodium-cooled fast breeder demonstration reactor designed to generate 61 MWe. This accident did not release any radioactivity to the environment. The damages were repaired and the reactor reached full power 4 years following the accident. It operated successfully and completed its mission in 1973.

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

5.9.4.4 Mitigation of Accident Consequences

Pursuant to the Atomic Energy Act of 1954, the NRC has conducted a safety evaluation of the application to operate Hope Creek. Although detailed information on plant design will be published in the Hope Creek Safety Evaluation Report, the principal design features are addressed in the following section.

(1) Design Features

The Hope Creek plant contains features designed to prevent accidental release of fission products from the fuel and to lessen the consequences should such a release occur. These accident-preventive and mitigative features are referred to collectively as engineered safety features (ESFs). To establish design and operating specifications for ESFs, postulated events referred to as design-basis accidents are analyzed.

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

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

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

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

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

Much more extensive discussion of these design features may be found in the applicant's FSAR and the staff's forthcoming SER. In addition, the implementation of the lessons learned from the TMI-2 accident - in the form of improvements in design, procedures, and operator training - will significantly reduce the likelihood of a degraded core accident that could result in large releases of fission products to the containment. The applicant will be required to meet the TMI-related requirements specified in NUREG-0737.

(2) Site Features

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

First, the site has an exclusion area, as required by 10 CFR 100. The total site area is about 299 ha (740 acres). The exclusion area, located within the site boundary, is a circular area with a minimum distance of 901 m (2,955 ft) from the center of the reactor building to the exclusion area boundary. There are no residents within the exclusion area. The applicant owns all surface and mineral rights on Artificial Island within the exclusion area and has the authority, as required by 10 CFR 100, to determine all activities in this area. There are no highways or railroads within the exclusion area. The Delaware River, including that section within the exclusion area, is used for barge and freight traffic as well as for commercial and recreational salt water fishing. In the event of an emergency, the applicant has made arrangements with the U.S. Coast Guard to control access to and activities on the Delaware River traversing the exclusion area.

Second, beyond and surrounding the exclusion area is a low population zone (LPZ), also required by 10 CFR 100. The LPZ for the Hope Creek site is a circular area with an 8.0-km (5.0-mi) radius. Within this zone, the applicant must ensure that there is a reasonable probability that appropriate protective measures could be taken on behalf of the residents in the event of a serious accident. The applicant has indicated that 1,190 persons lived within an 8-km radius in 1980. The major source of transients within the 8-km radius is related to the use of the Delaware River. The applicant indicates that the U.S. Army Corps of Engineers' data for the year 1979 show approximately 1.4 million persons making trips by vessel past the Artificial Island site.

There are two beaches and four wildlife refuge and management areas within the 8-km (5-mi) LPZ of the Hope Creek facility. About 2,100 persons visited the wildlife areas in 1981.

In case of a radiological emergency, the applicant has made arrangements to carry out protective actions, including evacuation of personnel in the vicinity of the Hope Creek station (see also the following section, "Emergency Preparedness").

Third, 10 CFR 100 also requires that the distance from the reactor to the nearest boundary of a densely populated area containing more than about 25,000 residents be at least one and one-third times the distance from the reactor to the outer boundary of the LPZ. Because accidents of greater potential hazards than those commonly postulated as representing an upper limit are conceivable, although highly improbable, it was considered desirable to add the population center distance requirement in 10 CFR 100 to provide for protection against excessive doses to people in large centers. The city of Newark, Delaware, with a 1980 population of 25,245, located 30 km (17.8 mi) northwest of the site, is the nearest population center. This population center distance is at least one and one-third times the LPZ distance. The population density within a 48-km (30-mi) radius of the site was 125 people/km^2 (320 people/mi^2) and is projected to increase to about 418 by the year 2030.

The safety evaluation of the Hope Creek site has also included a review of potential external hazards, that is, activities off site that might adversely affect the operation of the nuclear plant and cause an accident.

The review encompassed nearby industrial and transportation facilities that might create explosive, fire, missile, or toxic gas hazards.

The risk to the Hope Creek station from such hazards has been found to be negligible. A more detailed discussion of the compliance with the Commission's siting criteria and the consideration of external hazards will be presented in the Hope Creek SER.

(3) Emergency Preparedness

Emergency preparedness plans including protective action measures for the Hope Creek station have been developed by Public Service Electric and Gas Company and, for offsite areas, by state and local authorities. The onsite plans are being reviewed by the NRC; the Federal Emergency Management Agency (FEMA) is reviewing the offsite plans. In accordance with the provisions of 10 CFR 50.47, effective November 3, 1980, an operating license will not be issued to the applicant unless a finding is made by the NRC that the state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Among the standards that must be met by these plans are provisions for two emergency planning zones (EPZs). A plume exposure pathway EPZ of about 16 km (10 mi) in radius and an ingestion exposure pathway EPZ of about 80 km (50 mi) in radius are required. Other standards include appropriate ranges of protective actions for each of these zones, provisions for dissemination to the public of basic emergency planning information, provisions for rapid notification of the public during a serious reactor emergency, and methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences in the EPZs of an accidental radiological release.

The Hope Creek station is adjacent to a licensed commercial power reactor, Salem Generating Station, operated by the applicant. The offsite plans and much of the onsite plans for Hope Creek and Salem are common to both units.

NRC and FEMA have agreed that FEMA will make a finding and determination as to the adequacy of state and local government emergency response plans. NRC will determine the adequacy of the applicant's emergency response plans with respect to the standards listed in 10 CFR 50.47(b), the requirements of Appendix E to 10 CFR 50, and the guidance contained in NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," dated November 1980.

After the above determinations by NRC and FEMA, the NRC will make a finding in the licensing process as to the overall and integrated state of preparedness. The NRC staff findings will be reported in the SER.

5.9.4.5 Accident Risk and Impact Assessment

(1) Design-Basis Accidents

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

In the Hope Creek safety analysis and evaluation, three categories of accidents have been considered by the applicant and the staff. These categories are based on probability of occurrence and include (1) incidents of moderate frequency (events that can reasonably be expected to occur during any year of operation), (2) infrequent accidents (events that might occur once during the lifetime of the plant), and (3) limiting faults (accidents not expected to occur, but that have the potential for significant releases of radioactivity). The radiological consequences of incidents in the first category, also called anticipated operational occurrences, are similar to the consequences from normal operation that are discussed in Section 5.9.3. Some of the accidents postulated in the second and third categories for Hope Creek are shown in Table 5.13. These events are designated design-basis accidents in that specific design and operating features such as those described in Section 5.9.4.4(1) are provided to limit their potential radiological consequences. Approximate radiation doses that might be received by a person at the exclusion area boundary are also shown in the table, along with a characterization of the duration of the releases. The results shown in the table reflect an estimate of the potential upper bound of individual radiation exposures from the indicated initiating accidents. For these calculations, pessimistic (conservative) assumptions are made as to the course taken by the accident. These assumptions include conservatively large amounts of radioactive material released by the initiating events, additional single failures in equipment and operation of ESFs in a degraded mode.* The results of these calculations show that radioiodine releases have the potential for offsite exposures ranging up to about 15 rems to the thyroid. For such an exposure to occur, an individual would have to be located at a point on the site boundary where the radioiodine concentration in the plume has its highest value and inhale at a breathing rate characteristic of jogging for a period of 2 hours. The health risk to an individual receiving such a thyroid exposure is the potential appearance of benign or malignant thyroid nodules in about 5 out of 1,000 cases, and the development of a fatal cancer in about 2 out of 10,000 cases.

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

The staff experience has been that realistic dose estimates for a spectrum of accidents up to and including those as severe as design-basis accidents would result in values considerably lower than the above estimates or the staff's dose estimates for design-basis accidents established for the purpose of implementing the provisions of 10 CFR 100. It should be noted that although the staff did not perform any particular calculations of such dose estimates at the operating license stage for Hope Creek, such estimates were made by the staff in the Final Environmental Statement at the construction permit stage, and these estimates were only small fractions of the 10 CFR 20 limit at the site boundary.

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

(2) Probabilistic Assessment of Severe Accidents

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

The assessment methodology employed is essentially that described in the reactor safety study (RSS) (WASH-1400), which was published in 1975 as NUREG-75/014, but includes improvements in the assessment methodology that occurred after publication of the RSS* (such as thermal-hydraulic models, core melt phenomenology, and containment response analysis).

Accident sequences initiated by internal causes that are used in the staff analysis are described in Appendix F to this report, based on review of similar plants and consideration of recent design improvements at Hope Creek to reduce the probability of anticipated transients without scram. External events that might initiate severe accidents were not considered, except for loss of offsite power. For those sites for which externally initiated events were considered, the early fatality risk from externally initiated accidents was from 2 to 30 times that of internally initiated accidents, but other risks were comparable or less. Accident sequences are grouped into release categories based

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

on similarities of the sequences regarding core-melt-accident progression, containment failure characteristics, and the parameters of atmospheric release of radionuclides required for consequence analysis.

Table 5.14 provides information used in the staff's consequence assessment for each specific release category and summarizes the staff analysis described in Appendix F. The information includes time estimates from termination of the fission process during the accident until the beginning of release to the environment (release time), duration of the atmospheric release, warning time for offsite evacuation, estimates of the energy associated with each release, height of the release location above the ground level, and fractions of the core inventory (see Table 5.12) of seven groups of radionuclides in each release. The radionuclide release fractions shown in Table 5.14 were derived using WASH-1400 radiochemistry assumptions of fission product releases from fuel and their attenuation through various elements of the primary system and containment (such as the suppression pool), and the methods of this derivation are outlined in Appendix F. The staff's estimate of the probability associated with each release category used in the staff analysis is also shown in Table 5.14. As in the RSS, there are substantial uncertainties in these probabilities. This is due, in part, to difficulties associated with the quantification of human error and to inadequacies in the data base on failure rates of individual plant components (NUREG/CR-0400). These uncertainties are discussed in Section 5.9.4.5(7).

The magnitudes (curies) of radioactivity released to the atmosphere for each accident sequence or release category are obtained by multiplying the release fractions shown in Table 5.14 by the maximum amounts predicted to be in the Hope Creek core, and by a factor accounting for decay before release. The core inventory of radionuclides is shown in Table 5.12 for Hope Creek at a core thermal power level of 3,458 MWt. This is the power level used in the FSAR for analysis of radiological consequences and is used here instead of the 3,293-MWt expected maximum power to correct for power density variations and instrument error in measurement of power levels normally present in operating reactors. The 54 nuclides shown in the table represent those (of the hundreds actually expected to be present in the operating plant) that are potentially major contributors to the health and economic effects of severe accidents. They were selected on the basis of the half-life of the nuclide, consideration of the health effects of daughter products, and the approximate relative offsite dose contribution.

The potential radiological consequences of these releases have been calculated by the computer code CRAC, based on the consequence model used in the RSS (see NUREG-0340 and NUREG/CR-2300), adapted and modified as described below to apply to a specific site. The essential elements are shown in schematic form in Figure 5.4. Environmental parameters specific to the Hope Creek site have been used and include:

(1) meteorological data for the site representing a full year (1981) of consecutive hourly measurements and seasonal variations with good data recovery characteristics (annual average probabilities of wind blowing in 16 directions of the compass are shown in Table 5.15)

(2) projected population for the year 2010 extending throughout regions of 80-km (50-mi) and 563-km (350-mi) radius from the site

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

(4) land-use statistics on a statewide basis, including farm land values, farm product values including dairy production, and growing season information, for the State of New Jersey and each surrounding state within the 563-km (350-mi) region

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

The calculation was extended out to 3,200 km (2,000 mi) from the site, to account for the residual radionuclides that would remain in the atmosphere at large distances, with rain assumed in the interval between 563 km (350 mi) and 3,200 km (2,000 mi) to deplete the plume of all non-noble-gas inventory. To obtain a probability distribution of consequences, calculations were performed assuming the occurrence of each release category at each of 91 different "start" times distributed throughout a 1-year period. Each calculation used site-specific hourly meteorological data and seasonal information for the period following each start time.

The consequence model was also used to evaluate the consequence reduction benefits of offsite emergency response such as evacuation, relocation, and other protective actions. Early evacuation and relocation of people would considerably reduce the exposure from the radioactive cloud and the contaminated ground in the wake of the cloud passage. The evacuation model used (see Appendix G) has been revised from that used in the RSS for better site-specific application. The quantitative characteristics of the evacuation model used for the Hope Creek site are estimates made by the staff (see Table 5.16 for a summary of emergency response assumptions). There normally would be some facilities near a plant, such as schools or hospitals, where special equipment or personnel may be required to effect evacuation, and some people near a site who may choose not to evacuate. Such facilities (including Lower Alloways Creek School, Salem Nursing and Convalescent Center, and Salem County Jail) have been identified near the Hope Creek site. Therefore, actual evacuation effectiveness could be greater or less than that characterized, but it would not be expected to be very much less, because special consideration has been and will be given in emergency planning to any unique aspects of dealing with special facilities in the area around Hope Creek and the adjacent Salem units.

The other protective actions include (1) either complete denial of use (interdiction) or permitting use only at a sufficiently later time after appropriate decontamination of food stuffs such as crops and milk, (2) decontamination of severely contaminated environment (land and property) when it is considered to be economically feasible to lower the levels of contamination to protective action guide (PAG)* levels, and (3) denial of use (interdiction) of severely

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*The PAG levels used in the CRAC analyses are different from those drafted by the U.S. Environmental Protection Agency (EPA-520/1-75-001, September 1975), or by the U.S. Department of Health and Human Services (47 FR 47073, October 22, 1982), for reactor accidents. The PAG levels used are defined in Table VI 11-6 of WASH-1400 and were based on the recommendations of the former U.S. Federal Radiation Council and the British Medical Research Council. However, for control of long-term external irradiation, the staff used the PAG level for urban areas in WASH-1400 Table VI 11-6 for both urban and rural areas.

contaminated land and property for varying periods of time until the contamination levels are reduced to such values by radioactive decay and weathering that land and property can be economically decontaminated as in (2) above. These actions would reduce the radiological exposure to the people from immediate and/or subsequent use of, or living in, the contaminated environment, but would also result in costs of implementation. Lowering the PAG levels would lower the delayed health effects but would increase costs.

Early evacuation within and early relocation of people from outside the plume exposure pathway zone (see Appendix G) and other protective actions as mentioned above are considered as essential sequels to serious nuclear reactor accidents involving significant release of radioactivity to the atmosphere. Therefore, the results shown for Hope Creek include the benefits of these protective actions.

There are large uncertainties in each facet of the estimates of consequences, as there are for the probabilities (see Section 5.9.4.5(7)).

The results of the calculations using this consequence model are radiological doses to individuals and to populations, health effects that might result from these exposures, costs of implementing protective actions, and costs associated with property damage by radioactive contamination.

(3) Dose and Health Impacts of Atmospheric Releases

The results of the staff calculations of the environmental dispersion of radioactive releases to the atmosphere and the radiological dose to people and health impacts performed for Hope Creek are presented in the form of probability distributions in Figures 5.5 through 5.8 and are included in the impact summary Table 5.17. The graphs in Figures 5.5 through 5.8 display a type of probability distribution called a complementary cumulative distribution function (CCDF). CCDFs show the relationship between the probability of a type of accident consequence being equaled or exceeded and the magnitude of the consequence. These graphs are useful in visualizing the degree to which the probability of occurrence of consequences decreases as the magnitude of the consequence increases. Probability per reactor-year* is the chance that a given event would occur or a given consequence magnitude would be exceeded in 1 year of operation for one reactor. Different accident releases and atmospheric dispersion conditions, source-term magnitudes, and dose effects result in wide ranges of calculated magnitudes of consequences. Similarly, probabilities of equaling or exceeding a given consequence magnitude would also vary over a wide range because of varying probabilities of accidents and dispersion conditions.** Therefore, the CCDFs are presented as logarithmic plots in which numbers varying over a large range can be conveniently shown on a graph scaled in powers of 10. For example, a consequence magnitude of 10^6 means a consequence magnitude of 1 million (1 followed by six zeroes); a probability of 10^{-6} per reactor-year means a chance of 1 in 1 million or one-millionth (0.000001) per reactor-year. All release categories shown in Table 5.14 contribute to the results; the consequences from each are weighted by its associated probability.

*ry in the plots means reactor-year.
**See Section 5.9.4.5 (7) for further discussion of areas of uncertainty.

Figure 5.5 shows the probability distribution for the number of persons who might receive whole-body doses equal to or greater than 25 rems, total bone marrow doses equal to or greater than 200 rems, and thyroid doses equal to or greater than 300 rems from early exposure,* all on a per-reactor-year basis. The 200-rem total bone marrow dose figure corresponds, approximately, to a threshold value for which hospitalization would be indicated for the treatment of radiation injury. The 25-rem whole-body dose (which has been identified earlier as the lower limit for a clinically observable physiological effect in nearly all people) and the 300-rem thyroid dose figures correspond to the Commission's guideline values for reactor siting in 10 CFR 100.

Figure 5.5 shows in the left-hand portion that there are, approximately, 50 chances in 1 million (5×10^{-5}) per reactor-year that one or more persons may receive doses equal to or greater than any of the doses specified. The fact that the three curves run almost parallel in horizontal lines initially shows that if one person were to receive such doses, the chances are about the same that up to 1,000 would be so exposed. The chances of larger numbers of persons being exposed at those levels are seen to be considerably smaller. For example, the chances are less than 1 in 10 million (10^{-7}) that 10,000 or more people might receive doses of 200 rems or greater. Virtually all the doses reflected in this figure would be expected to occur to persons within a 80-km (50-mi) radius of the plant.

Figure 5.6 shows the probability distribution for the total population exposure in person-rems, that is, the probability per reactor-year that the total population exposure will equal or exceed the values given. Most of the population exposure up to 5 million person-rems would occur within 80 km (50 mi) but very severe releases would result in exposure to persons beyond the 80-km (50-mi) range, as shown.

For perspective, population doses shown in Figure 5.6 may be compared with the annual average dose to the population within 80 km (50 mi) of the Hope Creek site resulting from natural background radiation of about 600,000 person-rems, and to the anticipated annual population dose to the general public (total United States) from normal plant operation of about 30 person-rems (Appendix D, Tables D-7 and D-8).

Figure 5.7 represents the statistical relationship between population exposure and the induction of fatal cancers that might appear over a period of many years following exposure. The impacts on the total population and the population within 80 km (50 mi) are shown separately. Further, the fatal latent cancer estimates have been subdivided into those attributable to exposures of the thyroid and all other organs. About 40% of the latent cancer (including thyroid) fatalities would occur within 80 km (50 mi) of the plant.

Figure 5.8 shows the probability distribution of early fatalities. This calculated distribution reflects the assumption of severely exposed people benefiting from supportive medical treatment. The early fatalities would be expected to

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

all be within 24 km (15 mi) of the plant. As discussed in Appendix G, because it is possible that for very severe but low probability accidents some of the people requiring supportive medical treatment may not receive it, the consequences at the low-probability end of the spectrum may be somewhat higher than shown.

An additional potential pathway for doses resulting from atmospheric release is from fallout onto open bodies of water. This pathway has been investigated in the NRC analysis of the Fermi Unit 2 plant, which is located on Lake Erie, and for which appreciable fractions of radionuclides in the plume could be deposited in the Great Lakes (NUREG-0769). It was found that for the Fermi site, the computed individual and societal doses from this pathway were smaller than the interdicted doses from other pathways. Further, the individual and societal liquid pathway doses could be substantially eliminated by the interdiction of the aquatic food pathway in a manner comparable to interdiction of the terrestrial food pathway in the present analysis. Radioactive material accidentally released from Hope Creek would, depending on the wind direction, fall out onto the Delaware River, the Atlantic Ocean, lakes or reservoirs, or on land and eventually run off. The staff has also considered fallout onto and runoff and leaching into water bodies in connection with a study of severe accidents at the Indian Point reactors in southeastern New York (Codell, 1982-1983). In this study empirical models were developed based on considerations of radionuclide data from samples collected in the New York City water supply system after fallout from atmospheric weapons tests. As with the Fermi study, the Indian Point evaluation indicated that the uninterdicted risks from this pathway were fractions of the interdicted risks from other pathways. Further, if interdicted in a manner similar to interdiction assumed for other pathways, the liquid pathway risk from fallout would be a very small fraction of the risks from other pathways. Considering the regional meteorology and hydrology, the staff sees nothing to indicate that the liquid pathway contribution to the total accident risk from Hope Creek is significantly greater than that found for Fermi 2 and Indian Point. Therefore, the staff concludes that the water pathway would be of small importance compared to the results presented here for fallout onto land.

(4) Economic and Societal Impacts

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

The results are shown as the probability distribution for cost of offsite mitigating actions in Figure 5.9 and are included in the impact summary Table 5.17. The factors contributing to these estimated costs include the following:

- evacuation costs
- value of crops contaminated and condemned
- value of milk contaminated and condemned

- costs of decontamination of property where practical
- indirect costs resulting from the loss of use of property and incomes derived therefrom

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

Figure 5.9 shows that at the extreme end of the accident spectrum these costs could exceed tens of billions of dollars, but that the probability that this would occur is exceedingly small (less than 1 chance in 1 million per reactor-year).

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

The geographical extent of the kinds of impacts discussed above, as well as many other types of impacts, is a function of several factors. For example, the dispersion conditions and wind direction following a reactor accident, the type of accident, and the magnitude of the release of radioactive material are all important in determining the geographical extent of such impacts. Because of these large inherent uncertainties, the values presented herein are mean values of the important types of risk based on the methodology employed in the accident consequence model (NUREG-0340; NUREG/CR-2300) and do not indicate specific geographical areas.

(5) Releases to Groundwater

A pathway for public radiation exposure and environmental contamination that could be associated with severe reactor accidents was identified in Section 5.9.4.2(2). Consideration has been given to the potential environmental impacts of this pathway for the Hope Creek station. A penetration of the basemat of the containment building can release molten core debris to the strata beneath the plant. Soluble radionuclides in this debris can be leached and transported with groundwater to downgradient domestic wells used for drinking or to surface water bodies used for drinking water, aquatic food, and recreation. In BWRs, such as Hope Creek, there is an additional opportunity for groundwater contamination as a result of the release of suppression pool water to the ground through a breach in the containment.

An analysis of the potential consequences of a liquid pathway release of radioactivity for generic sites was presented in the "Liquid Pathway Generic Study" (LPGS) (NUREG-0440). The LPGS compared the risk of an accident involving the liquid pathway (drinking water, irrigation, aquatic food, swimming, and shoreline usage) for five conventional, generic, land-based nuclear plants and for a floating nuclear plant (for which the nuclear reactors would be mounted on a barge and moored in a water body). Parameters for the land-based site were chosen to represent averages for a wide range of real sites and are thus typical, although they do not represent any particular real site. The study concluded that the individual and population doses for the liquid pathway through groundwater contamination range from small fractions to very small fractions of those that can arise from airborne pathways.

The discussion in this section is a summary of an analysis performed to compare the liquid pathway consequences of a postulated core-melt accident at Hope Creek with that of the generic estuarine land-based site considered in the LPGS. The method consists of a direct scaling of LPGS population doses based on the relative values of key parameters characterizing the LPGS estuarine land-based site and the Hope Creek site. The parameters that were evaluated include the amounts of radioactive materials entering the ground, groundwater travel time, sorption on geological media, surface water transport, aquatic food consumption, and shoreline usage.

Doses to individuals and populations were calculated in the LPGS without consideration of interdiction methods such as isolating the contaminated groundwater, restricting aquatic food consumption, or denying use of the water. In the event of significant contamination, commercial and sports fishing as well as many other water-related activities could be restricted, if necessary. The consequences would, therefore, be largely economic or social, rather than radiological. In any event, the individual and population doses from the liquid pathway range from fractions to very small fractions of those that can arise from airborne pathways.

All of the reactors considered in the LPGS were PWRs with ice condenser containments. Although there are likely to be differences in the mechanisms and probabilities of release between the LPGS and Hope Creek (BWR) reactors, it is unlikely that an actual core-melt liquid pathway release for the BWRs would exceed that conservatively estimated for the LPGS. The staff is not aware of any studies that indicate the probabilities or magnitudes of liquid releases for BWRs. The source term for Hope Creek is, therefore, assumed to be equivalent to the LPGS source term.

The site occupies part of the southern end of Artificial Island immediately north of the two Salem nuclear units. The Hope Creek reactor is located about 290 m (950 ft) from the shoreline of the Delaware Estuary.

The top 9 to 12 m (30 to 40 ft) of the soil at the site consist of hydraulic fill, alluvium, clay, silt, and sand with some organic material. This layer is highly impermeable, with a permeability of 14.9 to 59.5 m/year (48.9 to 195 ft/year). From about 11 to 12 m (35 to 40 ft) below plant grade there is a 1.5- to 3-m (5- to 10-ft) layer of river bed sand and gravel that appears to be continuous throughout the site. This layer of sand and gravel riverbed deposits is referred to as the shallow aquifer and is hydraulically connected to the Delaware River Estuary. The permeability of the sand in this layer is estimated at 744 to 2,231 m/year (2,440 to 7,320 ft/year) based on grain-size analyses.

The Kirkwood Formation lies from about 12 to 21 m (40 to 70 ft) below the site. It consists of gray silty clay in the site area. Although this formation is used as an aquifer, it has a relatively low permeability in the site area and is considered an aquitard. Permeability values are estimated to be less than 745 m/year (2,440 ft/year).

The Vincentown Formation is encountered at a depth of about 21 to 41 m (70 to 135 ft) below the surface and consists of fine to medium-grained sand and gravel. Grain-size analysis indicates a permeability of about 2,975 m/year (9,760 ft/year) and an effective porosity of 0.28 and a total porosity of 0.35. Water level gradients are undetectable in this formation at the site, although it is an important aquifer regionally. The Vincentown Formation is hydraulically connected to the Delaware River Estuary and demonstrates appreciable tidal fluctuations.

Preconstruction measurements indicate that the water table beneath the site varies seasonally and responds to the tidal fluctuations in the estuary. A conservative estimate of the maximum groundwater elevation of 3.8 m (12.5 ft) MSL (plant grade) adjacent to the containment building was used as a design basis to evaluate the consequences of the postulated accidental release to groundwater. The groundwater gradient near the ground surface is to the southwest toward the adjacent estuary and has been conservatively estimated to fall at the rate of 3.8 m in 280 m (12.5 ft in 950 ft) or a slope of 0.0132.

The containment building is founded in the top of the Vincentown Formation passing through both the shallow aquifer and the Kirkwood Formation. Radioactivity released from the postulated core-melt accident at Hope Creek initially would be deposited into the Vincentown Formation. Radionuclides would then be transported by natural groundwater movement to the Delaware Estuary. No groundwater users would be affected, because there are no water supply wells along the containment flow path.

Using the parameters and the pathway discussed above, the time for the groundwater to migrate the 290 m (950 ft) to the Delaware Estuary has been conservatively estimated to be 2.0 years (742 days). This compares with a travel time of 0.61 year (223 days) for the LPGS site. It was demonstrated in the LPGS that for holdup times in the order of years, virtually all of the liquid pathway population dose results from Sr-90 and Cs-137. Therefore, the remainder of this analysis considers only these two radionuclides.

Movement of much of the radioactivity from an assumed core-melt accident would be slower than the groundwater velocity because of the effects of sorption (ion exchange) on the geologic media. Distribution coefficients (K_d) in sands ranging from 1.7 to 43 for strontium and 22 to 314 for cesium are reported by Isherwood (1977). For this example, retardation factors were calculated for a sandy type of soil using conservatively low distribution coefficient values for Sr-90 and Cs-137 of 2 and 22, respectively. This resulted in retardation factors of 12 and 127 for Sr-90 and Cs-137, respectively. This would result in a travel time for Sr-90 and Cs-137 of 25 years and 257 years, respectively. Because of radioactive decay, only about 56% of the Sr-90 and less than 1% of Cs-137 would eventually enter the Delaware Estuary. This compares to 88% of the Sr-90 and 31% of the Cs-137 escaping the groundwater pathway in the LPGS estuary example. The staff has conservatively assumed that any of the Sr-90 or Cs-137 escaping into the Delaware Estuary would subsequently be carried to the Delaware Bay and then to the Atlantic Ocean by tidal currents and freshwater flow from the Delaware River Basin.

The two major liquid pathways for an estuary site are aquatic food consumption and direct shoreline exposure. The commercial and recreational seafood catch (finfish and shellfish) for the Delaware Bay has been estimated by the applicant to be about 4.7×10^6 kg/year (10.4×10^6 lb/year). On the basis of the values determined for a similar analysis for the adjoining Salem station, the staff has estimated that beach usage would be about 4×10^6 person-hours/year. This is 1.5% of the approximately 2.6×10^8 user-hours/year used in the LPGS site.

In the case of the LPGS, about 26% of the fish dose and virtually all of the beach dose was due primarily to Cs-137 alone. The remainder of the fish dose was due to Sr-90. About 92% of the population dose was due to shoreline exposure and swimming with the remaining 8% being caused by fish ingestion.

Combining the ratios of the source term, groundwater pathway, fish catch, and shoreline usage indicates that the total population dose from a core-melt accident at Hope Creek would be about a factor of 0.007 (or 0.7%) of that for the LPGS estuary land-based site. The staff, therefore, concludes that the liquid pathway at Hope Creek does not pose an unusual contribution to risk when compared with other land-based estuary sites, and is small in comparison to the risk posed by airborne pathways.

Finally, there are measures that could be taken to further minimize the impact of severe accidents involving the liquid pathway. The staff estimates that the minimum groundwater travel time from the reactor to Delaware Bay is about 2.0 years and that the most significant radionuclides would be retarded by sorption. The travel time would allow time for measures to diminish the migration of the contaminated groundwater off the site. Grouting, where cement or chemical slurrys are injected under high pressure to seal aquifers, and slurry walls, where cement or chemical slurrys are mixed with the in situ soil to form an impermeable barrier, could be used to isolate the contamination. Dewatering of the water table could be used to prevent the mixing of contaminated water from the reactor with groundwater or to collect contaminated water for treatment. A comprehensive discussion of these and other mitigation methods potentially applicable to Hope Creek is contained in reports by Harris et al. (1982a and b).

(6) Risk Considerations

The foregoing discussions have dealt with both the probability per year of operation of accidents and their impacts (or consequences). Because the ranges of both factors are quite broad, it also is useful to combine them to obtain average measures of environmental risks. Such averages can permit a useful comparison of the impact on the public from radiological risks from accidental releases, both to the impact from normal operational releases, and to the impact from other forms of risk. Any comparison, however, should be tempered with an appreciation for the uncertainties in estimated values (see Section 5.9.4.5(7)).

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

Table 5.18 shows societal risk estimates associated with population dose, early fatalities with supportive medical treatment and with minimal medical treatment, early injuries, latent cancer fatalities, costs for evacuation and other protective actions, and land area for long-term interdiction. These risk values are obtained by multiplying the probabilities by the consequences, then summing these products over the entire range of consequences. Because the probabilities are on a per-reactor-year basis, the risks shown also are on a per-reactor-year basis.

The population exposures and latent cancer fatality risks for severe accidents may be compared with those from normal operation shown in Appendix D and Section 5.9.3.2 of this statement. The comparison (excluding exposure to station personnel) shows that the accident risks are up to 40 times higher than under normal operation. For a different perspective, the latent cancer (including thyroid) fatality risks of 4×10^{-2} persons per reactor-year within the 80-km (50-mi) region (from Table 5.18) may be compared with such risks from causes other than reactor accidents. Approximately 5 million persons are projected to live within the 80-km (50-mi) region in the year 2010. The average background cancer mortality rate is 1.9×10^{-3} cancer fatality per person per year in the United States (American Cancer Society, 1981). Therefore, at this rate, about 10,000 background cancer fatalities per year are expected in the population within the 80-km (50-mi) region in the year 2010. Thus, the risk of cancer fatality from reactor accidents at Hope Creek is small compared to the risk of normal occurrence of such fatality.

There are no early fatality, early injury, long-term land interdiction, or economic risks associated with protective actions and decontamination for normal releases, but these risks can be associated with large accidental releases. For perspective and understanding of the meaning of the early fatality risk of 9×10^{-6} person per reactor-year with supportive medical treatment and 3×10^{-4} person per reactor-year with minimal medical treatment (from Table 5.18), the staff notes that occurrences of early fatalities with supportive and minimal medical treatments would be contained, approximately, within the 24-km (15-mi) and 64-km (40-mi) regions, respectively. The number of persons projected to live within these regions in the year 2010 are 240,000 and 3.2 million, respectively. The risk from non-nuclear accidents for the average individual in the United States is 5×10^{-4} accidental death per year (NUREG/CR-1916). Therefore, the expected number of non-Hope Creek accidental fatalities per year within the 24-km (15-mi) and 64-km (40-mi) regions are 120 and 1,600, respectively, in the year 2010. Thus, the risk of early fatality with supportive or minimal medical treatment from reactor accidents at Hope Creek is extremely small compared with that from non-Hope Creek accidents.

Figure 5.10 shows the calculated risk expressed as whole-body dose to an individual from early exposure as a function of the downwind distance from the plant within the plume exposure pathway zone. The values are on a per-reactor-year basis, and all accident sequences and release categories contributed to the dose, weighted by their associated probabilities.

Evacuation and other protective actions can reduce the risk to an individual of early fatality or of latent cancer fatality. Figure 5.11 shows lines of constant risk per reactor-year to an individual living within the emergency planning zone of the Hope Creek site, of early fatality (as functions of distance) resulting from potential accidents in the reactor. No one lives in the area enclosed by the outer isopleth in Figure 5.11. Calculations based on everyone within 16 km (10 mi) of the plant evacuating show no risk of early fatality. There is some calculated early fatality risk to those people between 16 km (10 mi) and 24 km (15 mi) of the plant who are assumed not to evacuate but to relocate 12 hours after plume passage. Figure 5.12 shows curves of constant risk of latent cancer fatality. Directional variation of these plots reflects the variation in the average fraction of the year the wind would be blowing in different directions from the plant. For comparison, the following risks of fatality per year to an individual living in the United States may be noted

(National Research Council, 1979, p. 577): automobile accident, 2.2×10^{-4}; falls, 7.7×10^{-5}; drowning, 3.1×10^{-5}; burning, 2.9×10^{-5}; and firearms, 1.2×10^{-5}. For comparison to the estimated latent cancer fatality risk to an individual from Hope Creek reactor accidents, note that the non-nuclear-related risk of cancer fatality in the United States is 0.0019 per year (American Cancer Society, 1981).

A severe accident which requires the interdiction and/or decontamination of land areas will force numerous businesses to temporarily or permanently close. These closures would have additional economic effects beyond the contaminated areas through the disruption of regional markets and sources of supplies. This section provides estimates of these impacts, which were made using (1) the RSS consequence model (Appendix IV, WASH-1400) and (2) the regional input-output modeling system (RIMS II), developed by the Bureau of Economic Analysis (BEA) (NUREG/CR-2591).

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

Assumptions used in the analysis include:

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

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

(3) In the crop interdicted area there would be no loss in nonagricultural output; a 1-year loss in agricultural output, except no loss in greenhouse, nursery, and forestry output; no loss in livestock and poultry output; and a 2-month loss of dairy output.

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

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

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

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

Only the impacts occurring during the first year following an accident are considered. The longer term consequences are not considered because they will vary widely depending on the level and nature of efforts to mitigate the accident consequences and to decontaminate the physically affected areas. The estimates assume no compensating effects such as the use of unused capacity in the physically unaffected area to offset the initial lost production in the physically affected area or income payments to individuals displaced from their jobs that would enable them to maintain their habits. These compensating effects would reduce the industrial impacts. Realistically, these compensating effects would occur over a lengthy period. The estimates using no compensating effects are the best measure of first-year economic impacts.

Table 5.19 presents the regional economic output, employment impacts, and corresponding expected risks associated with the seven different release categories (see Table 5.14 for release category description). The estimated overall risk values using output losses as the measure of accident consequences, expressed on a per-reactor-year basis, is $34,818. This number is composed of direct impacts of $26,031 in the nonagricultural sector and $4,972 in the agricultural sector, and indirect impacts of $3,815 from decreased exports and supply constraints. The corresponding expected employment loss per reactor-year is about 1.4 jobs.

It should be noted that over 32% of the expected losses, or $11,009, result from releases occurring toward the north-northeast. The TQUVγ' etc. sequences contribute $7,033 of that amount. On an absolute basis, a TCγ' category release to the north-northeast is the greatest and would result in a loss of $12.7 billion and 513,000 jobs. Releases to the southeast along the Delaware Bay contribute nothing to the total expected loss. For each release category, for all directions, the minimal expected losses are from $0 to $38 per reactor-year.

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

The total estimated economic risk per year from reactor decontamination and restoration, replacement fuel costs, and the first postaccident year's regional economic impacts is $177,000 (1980 dollars) for Hope Creek. This includes the replacement power and recovery costs discussed above (but expressed in 1980 dollars) and the "Expected Losses per Reactor Year, Total," listed in Table 5.19. Not included in this are the costs of offsite decontamination, evacuation, relocation, and medical treatment. The risk of costs of offsite decontamination, evacuation, and relocation is about $31,000 (note that the cost shown in Table 5.18, $40,000, includes costs already accounted for in the regional industrial impacts). Therefore, the total of the economic risks considered in this study is about $208,000 (1980 dollars). Economic risks from medical treatment were not included in this total.

There are other impacts that can be expressed in monetary terms that are not included in the cost calculations discussed earlier. These impacts, which would result from an accident to the facility, produce added costs to the public (that is, ratepayers, taxpayers, and/or shareholders). These costs would accrue from decontamination and repair or replacement of the facility and from increased use of fossil fuels to provide replacement power during restoration of the facility. Experience with such costs is being accumulated as a result of the accident at the Three Mile Island facility.

If an accident occurs during the first year of operation of Hope Creek (1987), the economic penalty to which the public would be exposed would be approximately \$1,850 million for decontamination and restoration including replacement of the damaged nuclear fuel. This estimate is based on a 10% escalation of the 1980 economic penalty determined for the Three Mile Island facility (Comptroller General, 1981). Although insurance would cover \$300 million or more of the \$1,850 million accident cost, the insurance is not credited against this cost because the arithmetic product of the insurance payment and the risk probability would theoretically balance the insurance premium.

In addition, the staff estimates that system fuel costs would increase by approximately \$115 million for replacement power during each year Hope Creek is forced out of service. This estimate assumes that the unit will operate at an average 55% capacity and that replacement energy will be provided 53% from coal-fired generation, 34% from oil-fired generation, and 13% from gas-fired generation. If the unit does not operate for 8 years, the replacement power cost would amount to \$920 million (1987 dollars).

The probability of a core melt or severe reactor damage at Hope Creek was estimated to be about 10^{-4} per reactor-year (this accident probability is intended to account for all severe core-damage accidents leading to large economic consequences for the owner and not just those leading to significant offsite consequences). Multiplying the previously approximated cost of \$2,770 million for an accident to Hope Creek during the initial year of its operation by the above 10^{-4} probability results in an economic risk of approximately \$277,000 (1987 dollars) applicable to Hope Creek during its first year of operation. This is also the approximate economic risk (1987 dollars) to Hope Creek during the second year and each subsequent year of operation. Although nuclear units depreciate in value and may operate at reduced capacity factors, so that the economic consequences of an accident become less as the unit becomes older, this is considered to be offset by higher costs of decontamination and restoration of the units in the later years. Similarly, inflation is balanced by the present worth discount factor assuming a conservative 0% real discount rate. The \$277,000 annual risk for Hope Creek (1987 dollars) is equivalent to an annual risk of approximately \$208,000 (1984 dollars), assuming a 10% discount rate.

(7) Uncertainties

The probabilistic risk assessment discussed above has been based mostly on the methodology in the RSS, which was published in 1975 (WASH-4400, now designated NUREG-75/014). Although substantial improvements have been made in various facets of the RSS methodology since this publication was issued, there are still large uncertainties in the results of the analysis presented above because of the uncertainties associated with the likelihoods of the accident sequences and containment failure modes leading to the release categories, the source terms for the release categories, and the estimates of environmental consequences.

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

- Probability of Occurrence of Accident

  If the probability of a release category were to be changed by a certain factor, the probabilities of various types of consequences from that release category would also change exactly by the same factor. Thus, an order of magnitude uncertainty in the probability of a release category would result in an order of magnitude uncertainty in both societal and individual risks stemming from the release category. As in the RSS, there are substantial uncertainties in the probabilities of the release categories. This is due, in part, to difficulties associated with the quantification of human error and to inadequacies in (1) the data base on failure rates of individual plant components and (2) the data base on external events and their effects on plant systems and components that are used to calculate the probabilities. For externally initiated events, the uncertainty is only in the degree of underestimation, because external events were not included in the Hope Creek analysis (except loss of off-site power).

- Quantity and Chemical Form of Radioactivity Released

  The models used in these calculations contain approximations to describe the physical behavior of the radionuclides which affects the transport within the reactor vessel and other plant structures and the amounts of release. This relates to the quantity and chemical form of each radionuclide species that would be released from a reactor unit during a particular accident sequence. Such releases would originate in the fuel and would be attenuated by physical and chemical processes in route to being released to the environment. Depending on the accident sequence, immobilization or holdup of radionuclides in the reactor vessel, the primary cooling system, the containment, and adjacent buildings would influence both the magnitude and chemical form of radioactive releases. The releases of radionuclides to the environment, called source terms, used in the staff analysis were determined using the RSS methodology applicable to a BWR of Peach Bottom design. Information available in NUREG-0772 and from the latest research activities sponsored by the Commission and the industry indicates that the most realistic source terms cannot be much greater than the larger source terms used in this analysis (release categories TCγ' and TWγ' of Table 5.14), but they could be substantially lower (except for noble gases) than the release categories used here for the same types of initiating accident sequences. On the other hand some lower source term values could be underestimated, primarily because of the manner in which the source term was evaluated for early releases using the RSS methodology. The impact of lesser values of source terms would be substantially lower estimates of health effects, particularly early fatalities and injuries.

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

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

- Errors of Completeness, Modeling, Arithmetic, and Omission

  This area of lumped uncertainty includes such topics as the omission of a model of sabotage, consideration of externally-initiated accidents (except loss of offsite power), common cause failures, improvements in design or operating criteria undertaken or to be undertaken by the applicant, potential errors in the different models used to assess risks, errors associated with applying analyses from other plants to Hope Creek (see Appendix F), statistical errors, and arithmetic errors. The impact on risk estimates of this class of uncertainty could be large, but is unknown and virtually impossible to quantify accurately (Rowsome, 1982). Uncertainties of this type are expected to be larger than for other reactors for which comprehensive probabilistic risk assessments were performed.

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

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

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

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

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

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

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

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

- Meteorological Sampling Scheme Used

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

- Emergency Response Effectiveness

  The modeling assumptions of the emergency response of the people residing around the Hope Creek site may not correspond to what would happen during an actual severe reactor accident. Included in these considerations are such subjects as evacuation effectiveness under different circumstances, possible sheltering and its effectiveness, and the effectiveness of population relocation. The staff believes that the uncertainties associated with emergency response effectiveness could cause large uncertainties in estimates of early health consequences. The uncertainties in estimates of latent health consequences and costs are considered smaller than those of early health consequences. A limited sensitivity analysis in this area is presented in Appendix G. It indicates that the risk of early fatality with supportive medical treatment would be decreased by a factor of about 12, if the area of early evacuation was extended from 16.7 km (10 mi) to 25.0 km (15 mi).

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

  There are many uncertainties associated with estimates of dose and early health effects on individuals exposed to high levels of radiation. Included are the uncertainties associated with the conversion of contamination levels to doses, relationships of doses to health effects, and considerations of the availability of what was described in the RSS as supportive medical treatment (a specialized medical treatment program of limited availability that would minimize the early health effect consequences of high levels of radiation exposure following a severe reactor accident). The staff analysis shows that the variation in estimates of early fatality risks stemming from considerations of supportive medical treatment alone is about a factor of 30 for the Hope Creek site.

- <u>Dose Conversion Factors and Dose Response Relationships for Latent Health Consequences</u>

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

- <u>Chronic Exposure Pathways, Including Environmental Decontamination and the Fate of Deposited Radionuclides</u>

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

- <u>Economic Data and Modeling</u>

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

- <u>Fission Product Inventory</u>

  The fission product inventory presented in Table 5.12 is an approximation of that which would be present after extended operation at maximum power. The amount of each isotope listed will, in fact, vary with time in a manner dependent on the fuel management scheme and the power history of the core. The actual inventory at the time of an accident could not be much larger for any isotope than the amount in Table 5.12, but, especially for long-lived fission products, could be substantially smaller.

The means for quantitative evaluation of the uncertainties in a probabilistic risk analysis such as the type presented here are not well developed. The staff, however, has attempted to identify all sources of uncertainty and to assess the net effect on the uncertainty of the risk estimates. It is the judgment of the staff that the risk uncertainty bounds could be well over a factor of 10 but not as large as a factor of 100. The risk estimates are equal to the integrals of the corresponding probability distributions of the consequences (CCDFs). As a result, errors in probabilities and consequences are partially offset. Because of the magnitude of uncertainties, the staff has concluded that estimates of the probabilities, consequences, and risks do not provide an accident perspective unless the uncertainties are also considered. It follows, therefore, that conclusions relating to the estimated value of a particular risk or consequence (for example, the per-reactor-year chance of early fatality or the number of early fatalities expected for a particular accident sequence, respectively) should be based also on the uncertainties associated with the estimates.

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

(8) Comparison of Hope Creek Risks With Other Plants

To provide a perspective as to how Hope Creek compares in terms of risks from severe accidents with some of the other nuclear power plants that are either operating or that are being reviewed by the staff for possible issuance of a license to operate, the estimated risks from severe accidents for several nuclear power plants (including those for Hope Creek) are shown in Figures 5.13 through 5.21 for three important categories of risk. The values for individual plants are based on three types of estimates: from the RSS (labeled WASH-1400 Average Plant), from independent staff reviews of contemporary probabilistic risk assessments (Indian Point Units 2 and 3, Zion, and Limerick), and from generic applications of RSS methodology to reactor sites for environmental statements by the staff (for 24 nuclear power plants). Figure 5.13 indicates that the calculated risk of early fatality at Hope Creek is less than that at the majority of the plants evaluated, largely because no one lives within 5 km (3 mi) of the plant. Figures 5.16 and 5.19 show that the calculated risk of latent cancer fatalities at Hope Creek is higher than for most of the plants, mostly because of a higher-than-average population density more than 33 km (20 mi) from the plant. Furthermore, any or all of the estimates of risk could be under- or overestimates.

5.9.4.6 Conclusions

The foregoing sections consider the potential environmental impacts from accidents at the Hope Creek station. These have covered a broad spectrum of possible accidental releases of radioactive materials into the environment by atmospheric and liquid pathways. Included in the considerations are postulated design-basis accidents and more severe accident sequences that lead to a severely damaged reactor core or core melt. The applicant also considered similar accidents in the ER-OL. The staff, however, did not make use of the applicant's analysis.

The environmental impacts that have been considered include potential radiation exposures to individuals and to the population as a whole, the estimated likelihood of core-melt accidents, the risk of near- and long-term adverse health effects that such exposures could entail, and the potential economic and societal consequences of accidental contamination of the environment. These impacts could be severe, but the likelihood of their occurrence is judged to be small and comparable to that of other reactors. This conclusion is based on (1) the fact that considerable experience has been gained with the operation of similar facilities without significant degradation of the environment, (2) the fact that, to obtain a license to operate, the Hope Creek station must comply with the applicable Commission regulations and requirements, and (3) a probabilistic assessment of the risk based on the methodology developed in the RSS, improvements on the RSS methodology, and a brief sensitivity analysis of offsite emergency response modeling. The overall assessment of environmental risk of accidents, assuming protective actions, shows that the risks of population exposure and latent cancer fatality are within a factor of 40 of those from normal operation. Accidents have a potential for early fatalities and economic costs that cannot arise from normal operations; however, the risks of early fatality from potential accidents at the site are small in comparison with risks of early fatality from other human activities in a comparably sized population, and the accident risk will not add significantly to population exposure and cancer risks. Accident risks from Hope Creek are expected to be a small fraction of the risks the general public incurs from other sources. Further, the best-estimate calculations show that the risks of potential reactor accidents at Hope Creek are within the range of such risks from other nuclear power plants.

On the basis of the foregoing considerations of environmental impacts of accidents, which have not been found to be significant, the staff has concluded that there are no special or unique circumstances about the Hope Creek site and environs that would warrant consideration of alternatives for Hope Creek.

5.10 Impacts From the Uranium Fuel Cycle

The Uranium Fuel Cycle rule, 10 CFR 51.20 (44 FR 45362), reflects the latest information relative to the reprocessing of spent fuel and to radioactive waste management as discussed in NUREG-0116, "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," and NUREG-0216, which presents staff responses to comments on NUREG-0116. The rule also considers other environmental factors of the uranium fuel cycle, including aspects of mining and milling, isotopic enrichment, fuel fabrication, and management of low- and high-level wastes. These are described in the AEC report WASH-1248, "Environmental Survey of the Uranium Fuel Cycle." The staff was also directed to develop an explanatory narrative that would convey in understandable terms the significance of releases in the table. The narrative was also to address such important fuel cycle impacts as environmental dose commitments and health effects, socioeconomic impacts, and cumulative impacts, where these are appropriate for generic treatment. A proposed explanatory narrative was published in the Federal Register on March 4, 1981 (46 FR 15154-15175). Appendix C to this report contains a number of sections that address those impacts of the LWR-supporting fuel cycle that reasonably appear to have significance for individual reactor licensing sufficient to warrant attention for NEPA purposes.

Table S-3 of the final rule is reproduced in its entirety as Table 5.20 herein.* Specific categories of natural resource use included in the table relate to land use, water consumption and thermal effluents, radioactive releases, burial of transuranic and high- and low-level wastes, and radiation doses from transportation and occupational exposures. The contributions in the table for reprocessing, waste management, and transportation of wastes are maximized for either of the two fuel cycles (uranium only and no recycle); that is, the cycle that results in the greater impact is used.

Appendix C to this report contains a description of the environmental impact assessment of the uranium fuel cycle as related to the operation of the Hope Creek facility. The environmental impacts are based on the values given in Table S-3 (Table 5.20), and on an analysis of the radiological impact from radon-222 and technetium-99 releases. The staff has determined that the environmental impact of this facility on the U.S. population from radioactive gaseous and liquid releases (including radon and technetium) resulting from the uranium fuel cycle is very small when compared with the impact of natural background radiation. In addition, the nonradiological impacts of the uranium fuel cycle have been found to be acceptable.

5.11 Decommissioning

The purposes of decommissioning are (1) to safely remove nuclear facilities from service and (2) to remove or isolate the associated radioactivity from the environment so that part of the facility site that is not permanently committed can be released from other uses. Alternative methods of accomplishing these purposes and environmental impacts of each method are discussed in NUREG-0586.

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*The U.S. Supreme Court has upheld the validity of the S-3 rule in Baltimore Gas & Electric Co., et al. v. Natural Resources Defense Council, Inc., No. 82-524, issued June 6, 1983, 51 U.S. Law Week, 4678.

Section 5.3 of NUREG-0586 presents estimates of radiation doses to members of the public and to plant workers for decommissioning of a reference boiling water reactor.

Since 1960, 68 nuclear reactors - including 5 licensed reactors that had been used for the generation of electricity - have been or are in the process of being decommissioned. Although, to date, no large commercial reactor has undergone decommissioning, the broad base of experience gained from smaller facilities is generally relevant to the decommissioning of any type of nuclear facility.

Radiation doses to the public as a result of end-of-life decommissioning activities should be small; they will come primarily from the transportation of waste to appropriate repositories. Radiation doses to decommissioning workers should be well within the occupational exposure limits imposed by regulatory requirements. The NRC is currently conducting generic rulemaking that will develop a more explicit overall policy for decommissioning commercial nuclear facilities. Specific licensing requirements are being considered that include the development of decommissioning plans and financial arrangements for decommissioning nuclear facilities.

5.12 Noise

Noise levels generated by station operation were discussed in FES-CP, Section 5.5.4. Because the nearest residence is 5.5 km (3.3 mi) (ER-OL, Table 2.1-8) from the facility, the NRC staff believes that area residents will not be adversly affected by noise resulting from station operation.

5.13 Emergency Planning Impacts

In connection with the promulgation of the Commission's upgraded emergency planning requirements, the NRC staff issued NUREG-0658, "Environmental Assessment for Effective Changes to 10 CFR Part 50 and Appendix E to 10 CFR Part 50; Emergency Planning Requirements for Nuclear Power Plants." The staff believes the only noteworthy potential source of impacts to the public resulting from normal operations would be associated with the testing of the early notification system. The test requirements and noise levels will be consistent with those used for existing alert systems; therefore, the NRC staff concludes that the noise impacts from the system will be infrequent and insignificant.

5.14 Environmental Monitoring

5.14.1 Terrestrial Monitoring

Vegetation at the site was surveyed from 1972 through 1974 to determine the distribution and relative abundance of vascular plants (ER-OL, Section 2.2.1.2). Surveys of amphibians, reptiles, birds, and mammals were also conducted during this time. From June 1974 through December 1978, the distribution and abundance of waterfowl were determined within 8 km (5 mi) of Artificial Island (ER-OL, Section 2.2.1.3). Several reports prepared by the applicant (ER-OL, Section 2.2) present the results of these studies. The nesting success of ospreys, a species listed as endangered by the State of New Jersey, has been monitored since 1974 (ER-OL, Section 2.2.1.3).

The primary potential impact of station operation on terrestrial resources derives from cooling tower drift. Since issuance of the FES-CP, one unit and an associated cooling tower have been cancelled, reducing the potential impact by about one-half. In addition, more operating experience has been gained with natural draft cooling towers at other power plants (Section 5.5.1.1). This experience shows that significant impacts on terrestrial resources will likely not occur at Hope Creek if the cooling tower functions properly and is adequately maintained. To ensure proper cooling tower operation, the staff requires, as stated in the FES-CP (p. v), that the applicant measure the drift rate at the initiation of operation and periodically thereafter. The results will be reported to the NRC staff. Also, the applicant has committed to conduct an aerial photography program designed to detect any effects of cooling tower drift on vegetation. The details of this program will be specified in the Environmental Protection Plan that will be included as Appendix B of the operating license.

5.14.2 Aquatic Monitoring

The certifications and permits required under the Clean Water Act provide mechanisms for protecting water quality; therefore, aquatic biota also are protected indirectly by these mechanisms. Operational monitoring of effluents will be required by the NPDES permit issued by the State of New Jersey.

The construction permit required that the applicant conduct a preoperational ecological monitoring program for a period of 10 years subsequent to July 1, 1975. Also the construction permit required that reports be made to the NRC if operation of Salem was found to cause significant adverse effects on the aquatic ecology of the Delaware River Estuary. The NRC, however, will rely on the State of New Jersey, under the authority of the Clean Water Act, for the protection of water quality and aquatic biota and for any associated nonradiological monitoring that may be required during plant operation.

An Environmental Protection Plan will be included as Appendix B to the Hope Creek operating license. This plan will include requirements for prompt reporting by the applicant of important events that potentially could result in significant environmental impacts causally related to plant operation (for example, fish kills, mortality of any species protected by the Endangered Species Act of 1973 as amended, an increase in nuisance organisms or conditions, or unanticipated or emergency discharge of water or chemical substances).

5.14.3 Atmospheric Monitoring

The FES-CP did not contain a description of the onsite meteorological measurements program. Meteorological measurements have been made on Artificial Island since 1969, originally in support of the application of the Salem Generating Station. The 91-m (300-ft) meteorological tower is located about 1,500 m (4,920 ft) east-southeast of the Hope Creek plant complex. Wind speed and wind direction are measured at the 10-m (33-ft), 45.7-m (150-ft), and 91.4-m (300-ft) levels, and vertical temperature gradient was measured between the 10-m and 45.7-m levels and between the 10-m and 91.4-m levels. Ambient dry bulb and dew point temperatures are measured at the 10-m level, and precipitation and solar radiation are measured near the ground.

Two years (June 1969-May 1971) of data from this tower were submitted with the operating license application for the Salem facility. Five years (January 1977-December 1981) of data from this program were included in the ER-OL for Hope Creek. Meteorological data from the two collection periods have been compared. The 5 years of onsite data have been combined into joint frequency distribution of wind speed and wind direction by atmospheric stability for use in the atmospheric dispersion assessment described in Appendix D. Wind speed and wind direction data for these assessments were based on measurements at the 10-m level, and atmospheric stability was defined by the measurement of vertical temperature gradient between the 10-m and 45.7-m levels.

Analog strip charts have been used to record meteorological data. The measurements system is checked daily in accordance with Technical Specification requirements for the Salem Generating Station. Calibration of the system was initially performed semiannually and changed to triannually in March 1978. Since 1980, the measurements system has been calibrated quarterly. Joint data recovery of wind speed and wind direction at the 10-m level by atmospheric stability (defined by the vertical temperature gradient between the 10-m and 45.7-m levels) was 84% for the 5-year period (January 1977-December 1981). Although data recovery was below the recommended 90%, the staff has used the joint frequency data described above in the assessment of atmospheric dispersion characteristics presented in Appendix D. Because the periods of data were sufficiently random, the 5-year period of record is expected to reasonably reflect expected diurnal, seasonal, and annual airflow and stability patterns at the Hope Creek site. The 5-year period of record is also expected to reasonably represent occurrences of extreme atmospheric conditions of importance for assessments of local transport and diffusion characteristics. The frequencies of occurrence of moderately stable and extremely stable conditions at Hope Creek agree reasonably well with other sites in the northeastern United States. Dose consequence assessments based on available onsite meteorological data are expected to be reasonably conservative.

The applicant claims that the entire onsite meteorological measurements system complies with the accuracy specifications presented in RG 1.23, "Onsite Meteorological Programs." However, the applicant has not yet provided estimates of the overall system accuracy for each parameter measured. The types of wind speed and direction sensors and recording equipment identified by the applicant have been used by other applicants and licensees to meet the accuracy specifications of RG 1.23. The applicant's method, using a matched pair of thermistors for determining vertical temperature gradient, is uncommon. Additional information is required from the applicant to demonstrate that the accuracies of meteorological measurements comply with the system accuracy specifications presented in RG 1.23.

The meteorological measurements program during plant operation will include those parameters currently measured. Meteorological parameters are to be available for display through the radiation monitoring system central radiation processor (CRP), although the method of display has not been specified. Calculations of atmospheric transport and diffusion are also to be available through the CRP, although the models and/or methodology have not been described.

5.15 References

Adams, S. M., B. L. Kimmel, and G. R. Ploskey, "Sources of Organic Matter for Reservoir Fish Production: A Trophic Dynamics Analysis," Can. J. Fish. Aquat. Sci. 40:1480-1495, 1983.

Advisory Committee on the Biological Effects of Ionizing Radiations, BEIR I, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," National Academy of Sciences/National Research Council, November 1972.

---, BEIR III, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," National Academy of Sciences/National Research Council, July 1980.

American Cancer Society, "Cancer Facts and Figures 1979," 1978.

---, "Cancer Facts and Figures," 1981.

Blaylock, B. G., and J. P. Witherspoon, "Radiation Doses and Effects Estimated for Aquatic Biota Exposed to Radioactive Releases from LWR Fuel-Cycle Facilities," Nuclear Safety, 17:351, 1976.

Brungs, U. A., "Literature Review of the Effects of Residual Chlorine on Aquatic Life," J. Water Pollut. Contr. Fed. 45:2180-2193, 1973.

Carney, S. M., M. F. Sorensen, and E. M. Martin, "Distribution of Waterfowl Species Harvested in States and Counties During 1971-1980 Hunting Seasons," U.S. Fish and Wildlife Service Special Scientific Report, Wildlife No. 254, Washington, D.C., 1983.

Carson, J. E., "Atmospheric Impacts of Evaporative Cooling Systems," ANL/ES-53, Argonne National Laboratory, Argonne, Illinois, 1976.

Clark G. L., "Dynamics of Production in a Marine Area," Ecol. Monogr. 16(4): 322-335, 1946.

Code of Federal Regulations, Title 10, "Energy," and Title 40, "Protection of Environment," U.S. Government Printing Office, Washington, D.C.

Codell, R., NRC staff, Written Staff Testimony on Commission Question 1, Section III.D on Liquid Pathway Considerations for the Indian Point ASLB Special Hearing, June 1982-April 1983.

Comptroller General of the United States, "Report to the Congress," EMD-81-106, August 26, 1981.

Crick, M. J., and G. S. Linsley, "An Assessment of the Radiological Impact of the Windscale Reactor Fire," NRPB-R135, National Radiological Protection Board, 1982.

Federal Insurance Administration, "Flood Insurance Study for the Township of Lower Alloways Creek, Salem County, New Jersey." October 18, 1982.

Gulland, J. A., "Food Chain Studies and Some Problems in World Fisheries," in J. H. Steele (ed.), Marine Food Chains, University of California Press, Los Angeles, California, 1970, pp. 296.-315.

Harris, V. A., M. L. B. Winters, and J. Y. Yang, "Accident Mitigation: Alternative Methods for Isolating Contaminated Groundwater." Division of Environmental Impact Studies, Argonne National Laboratory, Argonne, Illinois, September 1982a.

Harris, V. A., D. Y. Yang, M. C. Bynoc, and J. S. Warkentien, "Accident Mitigation: Slurry Wall Barriers," Argonne National Laboratory, Argonne, Illinois, May 1982b.

Harvey, H. W., "On the Production of Living Matter in the Sea off Plymouth," J. Mar. Biol. Asoc. U.K. 29:97-137, 1950.

Heinle, D. R., "Production of a Calanoid Copepod, Acartia tonasa, in the Patuxent River Estuary," Ches. Sci. 7:59-74, 1966.

International Commission on Radiological Protection (ICRP), "Recommendations of the International Commission on Radiological Protection," ICRP Publication 26, January 1977.

Isherwood, D., "Preliminary Report on Retardation Factors and Radionuclide Migration," Lawrence Livermore Laboratory, August 1977.

Land, C. E., "Estimating Cancer Risks From Low Doses of Ionizing Radiation," Science 209:1197-1203, September 12, 1980.

Lee, J. M., Jr., J. H. Brunke, G. E. Lee, G. L. Reiner, and F. L. Shon, "Electrical and Biological Effects of Transmission Lines: A Review." U.S. Department of Energy, Bonneville Power Administration, Portland, Oregon, 1982.

Leitzell, T. L., National Marine Fisheries Service, Washington, D.C., letter to W. H. Regan, NRC, with attachment "Endangered Species Act of 1973 (As Amended), Section 7, Consultation-Biological Opinion," 1980.

Lind, O. T., Handbook of Common Methods in Limnology, C. V. Mosby Co., St. Louis, Missouri, 1979.

Lindsay, J. A., and N. J. Morrison, "A Study of Zooplankton in the Delaware River in the Vicinity of Artificial Island in 1972," in An Ecological Study of the Delaware River in the Vicinity of Artificial Island, Progress Report for the Period January-December 1972, Ichthyological Associates, 1974, pp. 482-555.

Merkens, J. C., "Studies on the Toxicity of Chlorine and Chloramines to Rainbow Trout," Water Waste Treat. J. 7:150-151, 1958.

Mittl, R. L., PSE&G, letter to A. Schwencer, "Hope Creek Generating Station - Environmental Report - Operating License Stage - Docket No. 50-354," May 29, 1984.

Mulchi, C. L., and J. A. Armbruster, "Response of Corn and Soybean to Simulated Saline Aerosol Drift From Brackish Water Cooling Towers," J. Environ. Qual. 10:541-547, 1981.

National Council or Radiation Protection and Measurements (NCRP), "Review of the Current State of Radiation Protection Philosophy," NCRP Report No. 43, January 1975.

National Research Council, "Energy in Transition 1975-2010," Final Report of the Committee on Nuclear and Alternative Energy Systems (CONAES), Chapter 9, 1979.*

New Jersey Department of Environmental Protection, "Final Report on Salt Depositon From the Hope Creek Cooling Towers and the Potential Effect of the Salt on Crops and Livestock," Trenton, New Jersey, 1980.

Nieves, L. A., et al., "Estimating the Economic Costs of Radiation Induced Health Effects," Pacific Northwest Laboratory Report PNL-4664, 1983.

Oak Ridge National Laboratory, "Descriptions of Selected Accidents That Occurred at Nuclear Reactor Facilities," ORNL/NSIC-176, Oak Ridge, Tennessee, April 1980.

President's Commission on the Accident at Three Mile Island, "Report of the President's Commission on the Accident at Three Mile Island," Commission Findings B, Health Effects, Washington, D.C., October 1979.

Public Service Electric and Gas Company (PSE&G) "Ecological Studies of the Delaware River in the Vicinity of Artificial Island: Progress Reports for the Periods January-December 1973-1977," Newark, New Jersey, 1973-1977.

---, "An Ecological Study of the Delaware River Near Artificial Island - 1968-1975: A Summary," NRC Docket No. 50-272, Newark, New Jersey, 1980a.

---, "An Interpretive Analysis of Entrainment and Impingement - April 1977-December 1978," NRC Docket No. 50-272, Newark, New Jersey, 1980b.

Rowsome, F. H., NRC staff, Direct Testimony on the Accuracy of the Indian Point Risk Assessment, 1982.

Stout, I. J., and G. W. Cornwell, "Nonhunting Mortality of Fledged North American Waterfowl," J. Wildl. Manage. 40:681-693, 1976.

Talbot, J. J., "A Review of Potential Biological Impacts of Cooling Tower Salt Drift," Atmos. Environ. 13:395-405, 1979.

Thompson and Beckerley, The Technology of Nuclear Reactor Safety, Vol. 1, The MIT Press, Cambridge, Massachusetts, 1964.

United Kingdom Atomic Energy Office, "Accident at Windscale," Her Majesties Stationary Office Cmnd. 302, 1957.

United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), "Ionizing Radiation: Sources and Biological Effects," 1982.

*This report was also published in 1980 by W. H. Freeman and Company. Pages cited will differ.

U.S. Atomic Energy Commission, WASH-1248, "Environmental Survey of the Uranium Fuel Cycle," April 1974.

U.S. Department of Health and Human Services, "Accidental Radioactive Contamination of Human Food and Animal Feeds; Recommendations for State and Local Agencies," 47 FR 47073, October 22, 1982.

U.S. Environmental Protection Agency, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA-520/1-75-001, Washington, D.C., September 1975.

---, "Quality Criteria for Water," EPA-440/9-76-023, 1976.

U.S. Nuclear Regulatory Commission, "Draft Environmental Statement Related to the Construction of Atlantic Generating Station Units 1 and 2," Docket Nos. STN 50-477 and STN 50-478, 1976.

---, "Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1979," 45 FR 40101-40104, June 13, 1980.

---, "10 CFR Part 51, Appendix A, Narrative Explanation of Table S-3, Uranium Fuel Cycle Environmental Data," proposed rule, 46 FR 15154-15175, March 4, 1981.

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

---, NUREG-0016, "Calculation of Radioactive Materials in Gaseous and Liquid Effluents From Boiling Water Reactors," Revision 1, January 1979.

---, NUREG-0116 (Supplement 1 to WASH-1248), "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," October 1976.

---, NUREG-0216 (Supplement 2 to WASH-1248), "Public Comments and Task Force Responses Regarding the Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," March 1977.

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

---, NUREG-0440, "Liquid Pathway Generic Study," February 1978.

---, NUREG-0555, "Environmental Standard Review Plans for the Environmental Review of Construction Permit Applications for Nuclear Power Plants," May 1979.

---, NUREG-0558, "Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station," May 1979.

---, NUREG-0586, "Draft Generic Environmental Statement on Decommissioning Nuclear Facilities," January 1981.

---, NUREG-0651, "Evaluation of Steam Generator Tube Rupture Accidents," March 1980.

---, NUREG-0654, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," November 1980 (also issued as FEMA-REP-1).

---, NUREG-0658, "Environmental Assessment for Effective Changes to 10 CFR 50 and Appendix E to 10 CFR 50; Emergency Planning Requirements for Nuclear Power Plants," August 1980.

---, NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," Vol. I, May 1980.

---, NUREG-0671, "Assessment of the Impacts of the Salem and Hope Creek Stations on the Shortnose Sturgeon, Acipenser brevirostrum LeSueur," 1980.

---, NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors, 1981," Vol. 3, November 1982.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980; Supplement 1, January 1983.

---, NUREG-0769, "Final Environmental Statement Related to the Operation of Enrico Fermi Atomic Power Plant, Unit No. 2," Addendum No. 1, March 1982.

---, NUREG-0772, "Technical Bases for Estimated Fission Product Behavior During LWR Accident," June 1981.

---, NUREG-0773, "The Development of Severe Reactor Accident Source Terms: 1957-1981," R. Blond, et al., 1982.

---, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," July 1981 (formerly issued as NUREG-75/087).

---, NUREG/CR-0400, "Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission," September 1978.

---, NUREG/CR-0912, "Geosciences Data Base Handbook for Modeling a Nuclear Waste Repository," Lawrence Livermore Laboratory, January 1981.

---, NUREG/CR-1250, Special Inquiry Group, "Three Mile Island - A Report to the Commissioners and the Public," Vol. I, Summary Section 9, Mitchell Rogovin, Director, January 1980.

---, NUREG/CR-1916, "A Risk Comparison," A. Coppola, et al., Brookhaven National Laboratory, February 1981.

---, NUREG/CR-2300, "PRA Procedures Guide," American Nuclear Society, January 1983.

---, NUREG/CR-2591, "Estimated Potential Industrial Impacts of a Nuclear Reactor Accident: Methodology and Case Studies," J. V. Cartwright, et al., 1982.

---, NUREG/CR-2965, "Copper in the Intake and Discharge Zones of the Surry and Salem Nuclear Power Stations," F. L. Harrison, D. J. Bishop, D. W. Rice, and J. P. Knezovich, 1983.

Vollenweider, R. A. (ed.), A Manual on Methods for Measuring Primary Production in Aquatic Environments, IPB Handbook No. 12, Blackwell Scientific Publishers, Oxford, 1971.

Wilber, K. R., and R. O. Webb, "Environmental Effects of Cooling Towers," AIF/NESP-026, Atomic Industrial Forum, Inc., 1983.

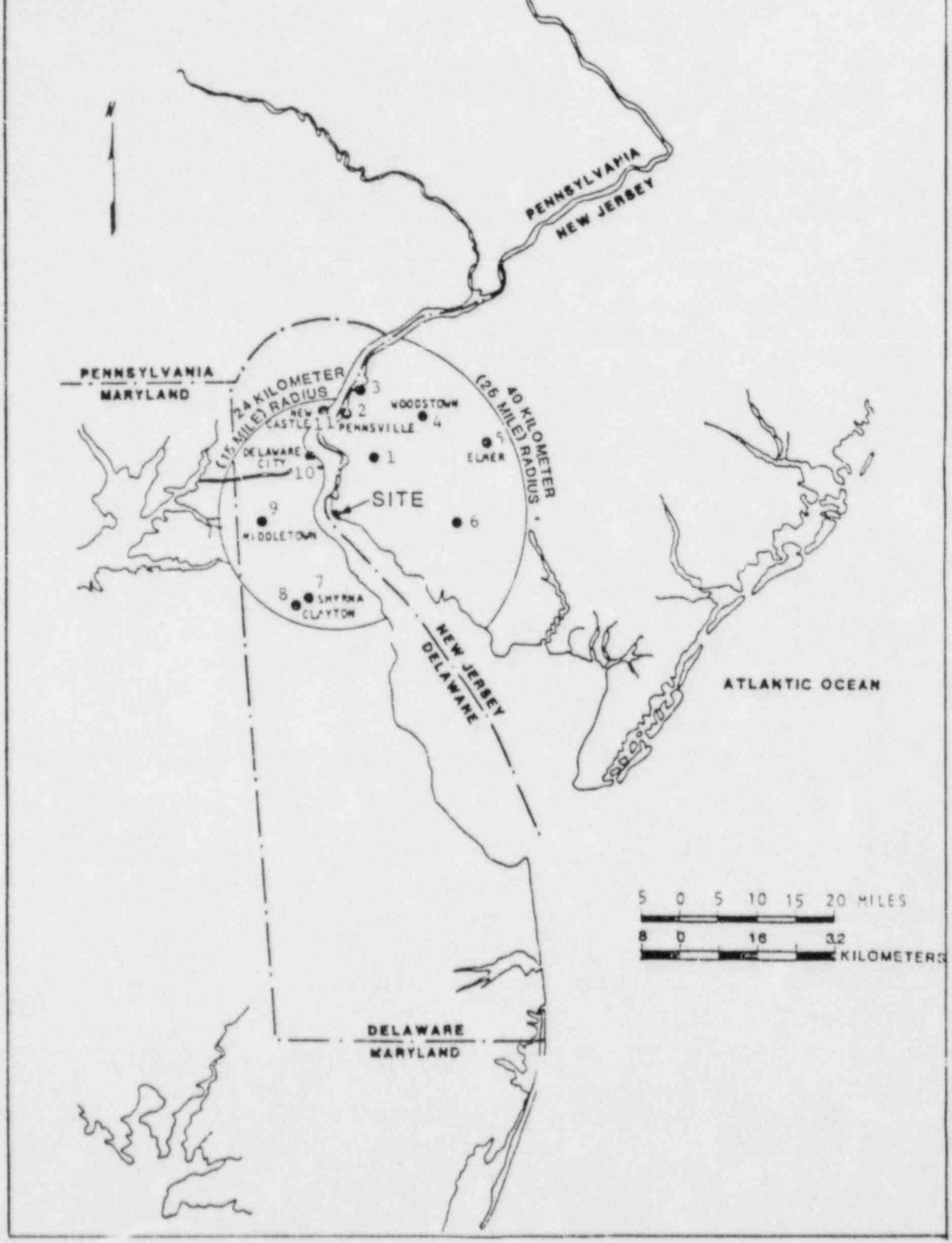

Figure 5.1 Public water supplies in the site region
Source: ER-OL, Figure 5.3-1

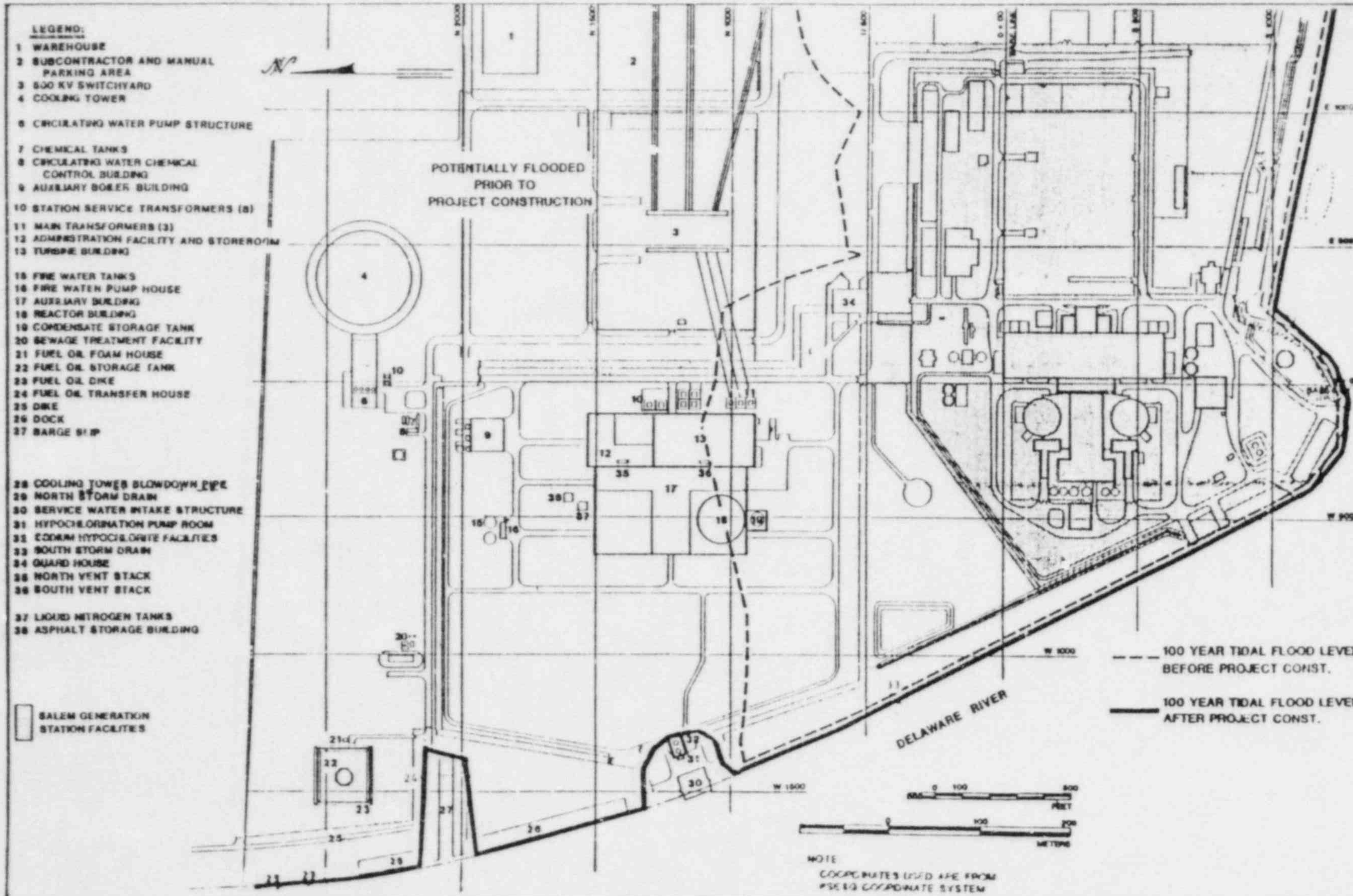

Figure 5.2 100-year tidal flood boundary
Source: ER-OL, Figure 5.3-2

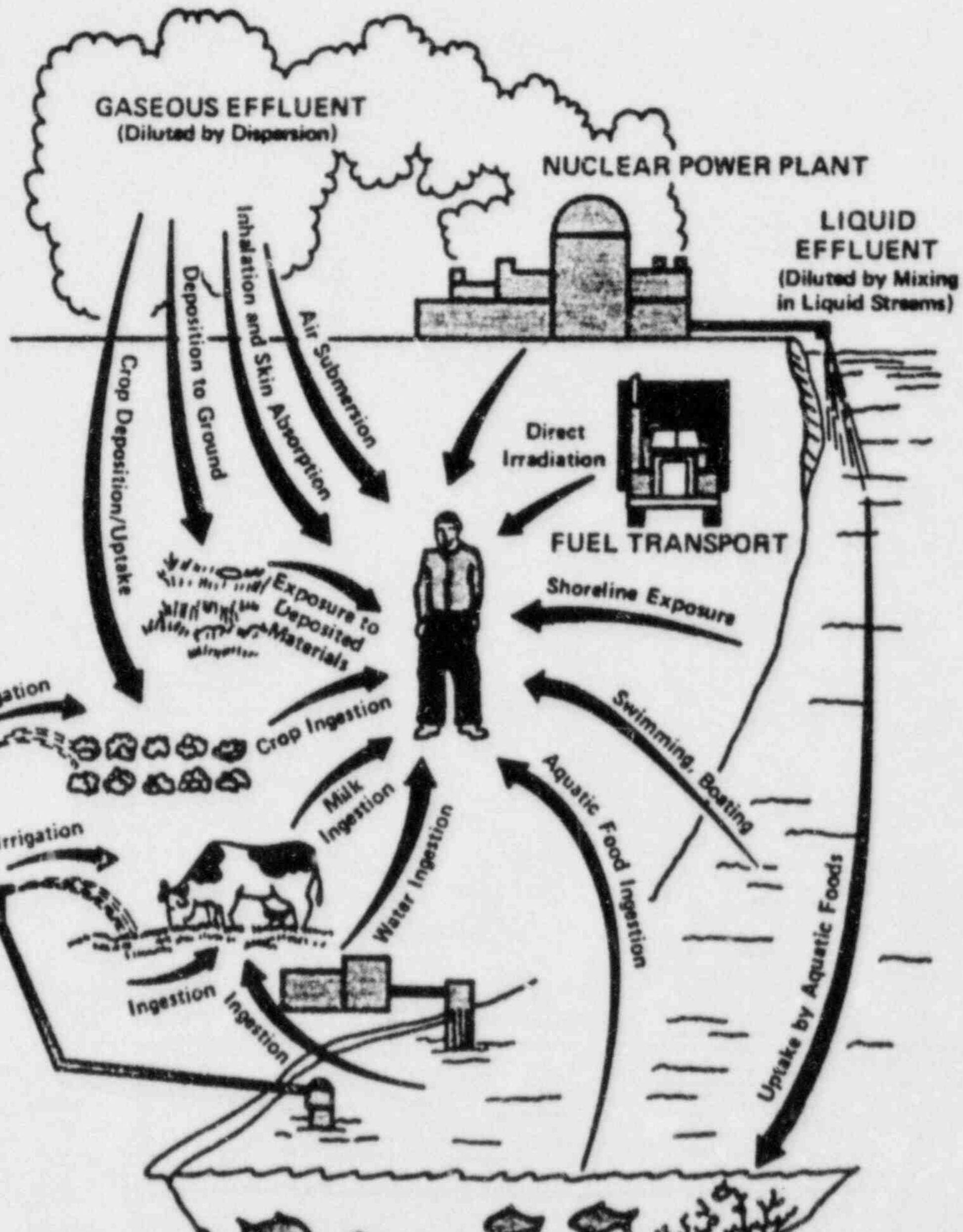

Figure 5.3 Potentially meaningful exposure pathways to individuals

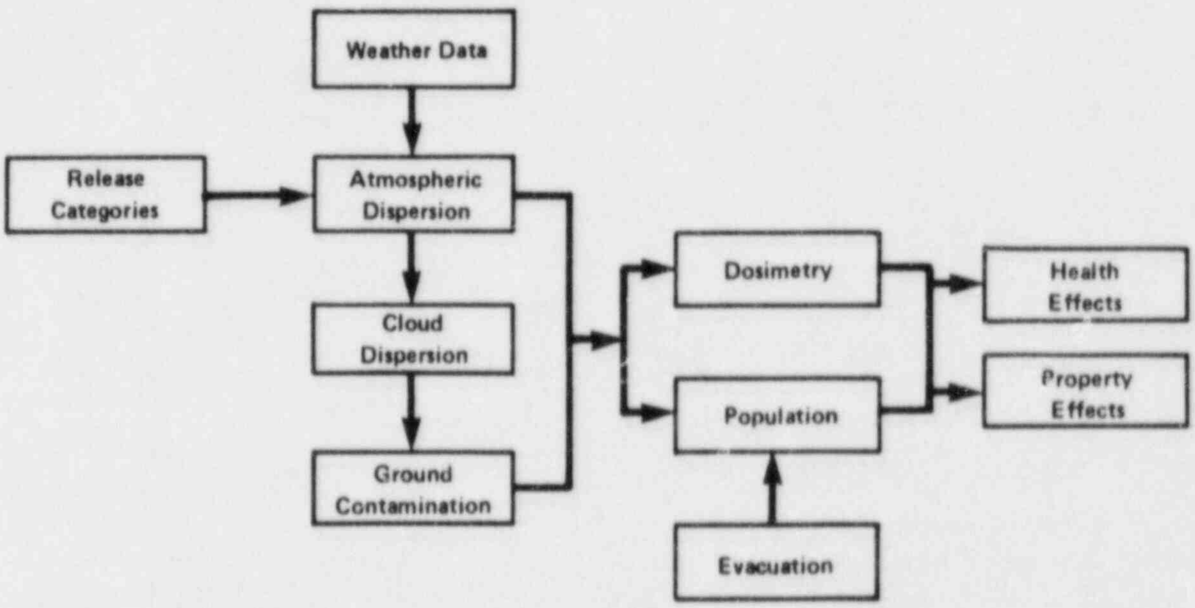

Figure 5.4 Schematic outline of atmospheric pathway consequence model

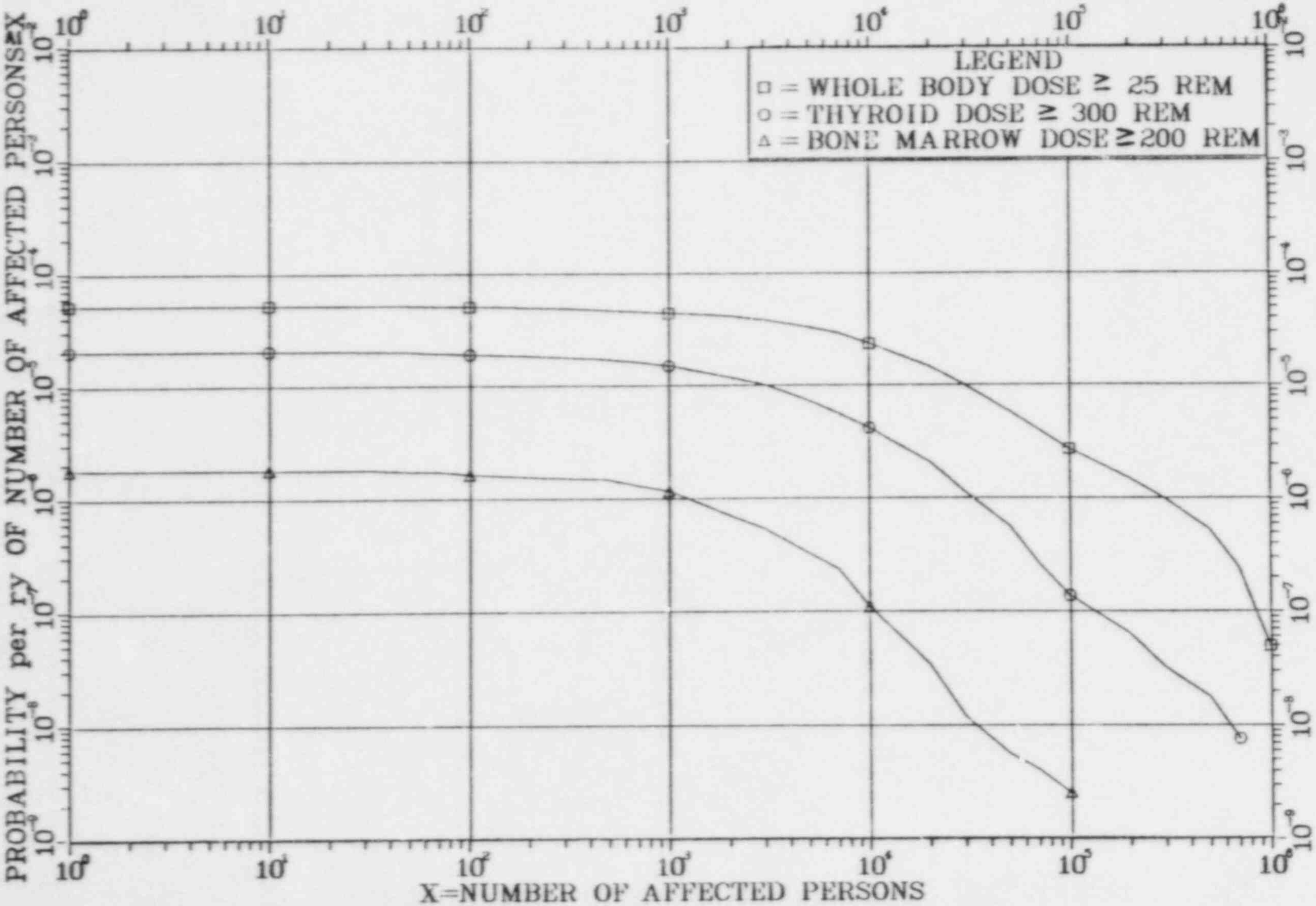

Figure 5.5 Probability distributions of individual dose impacts
NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties

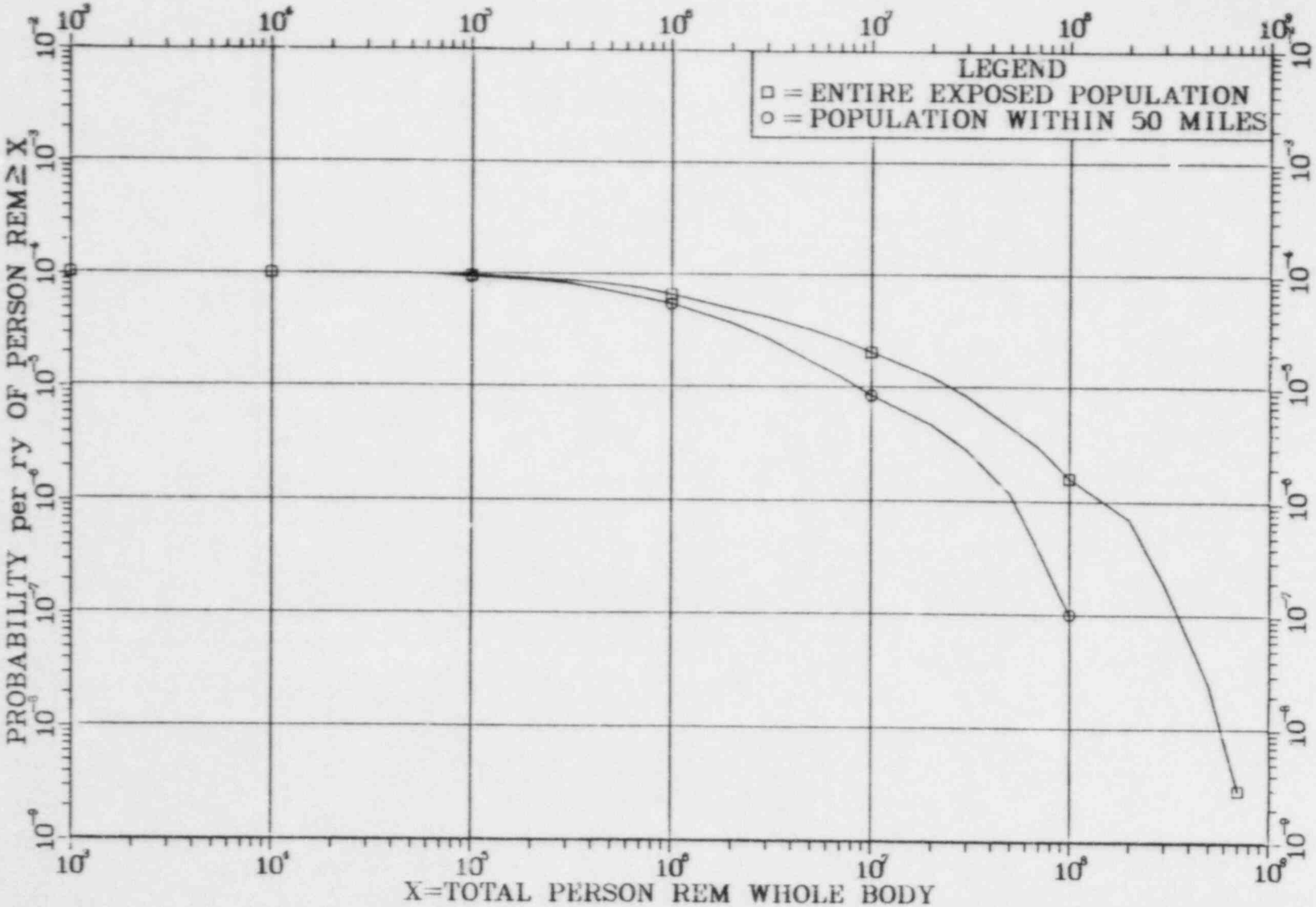

Figure 5.6 Probability distributions of population exposures
NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties.

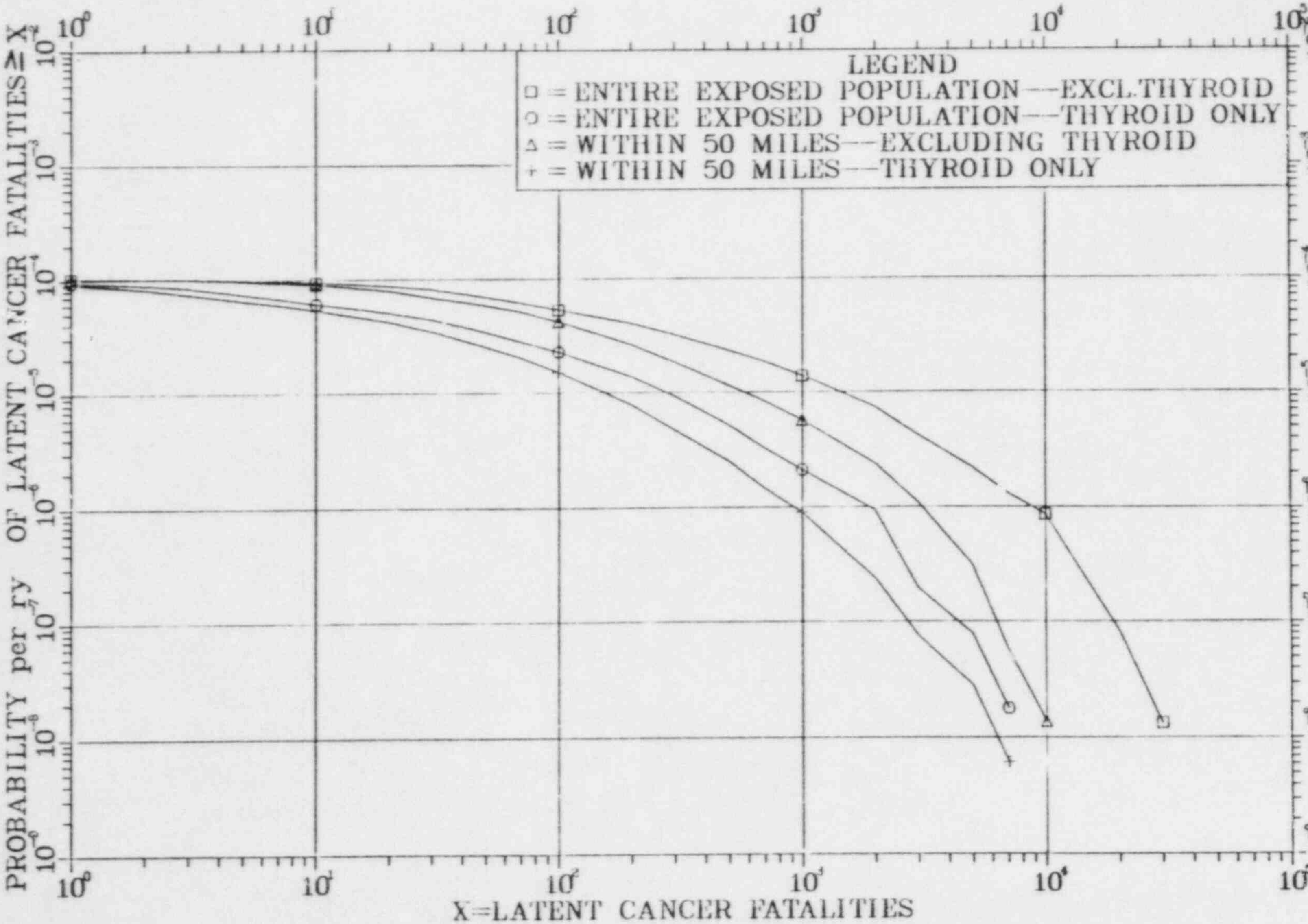

Figure 5.7 Probability distributions of cancer fatalities
NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties.

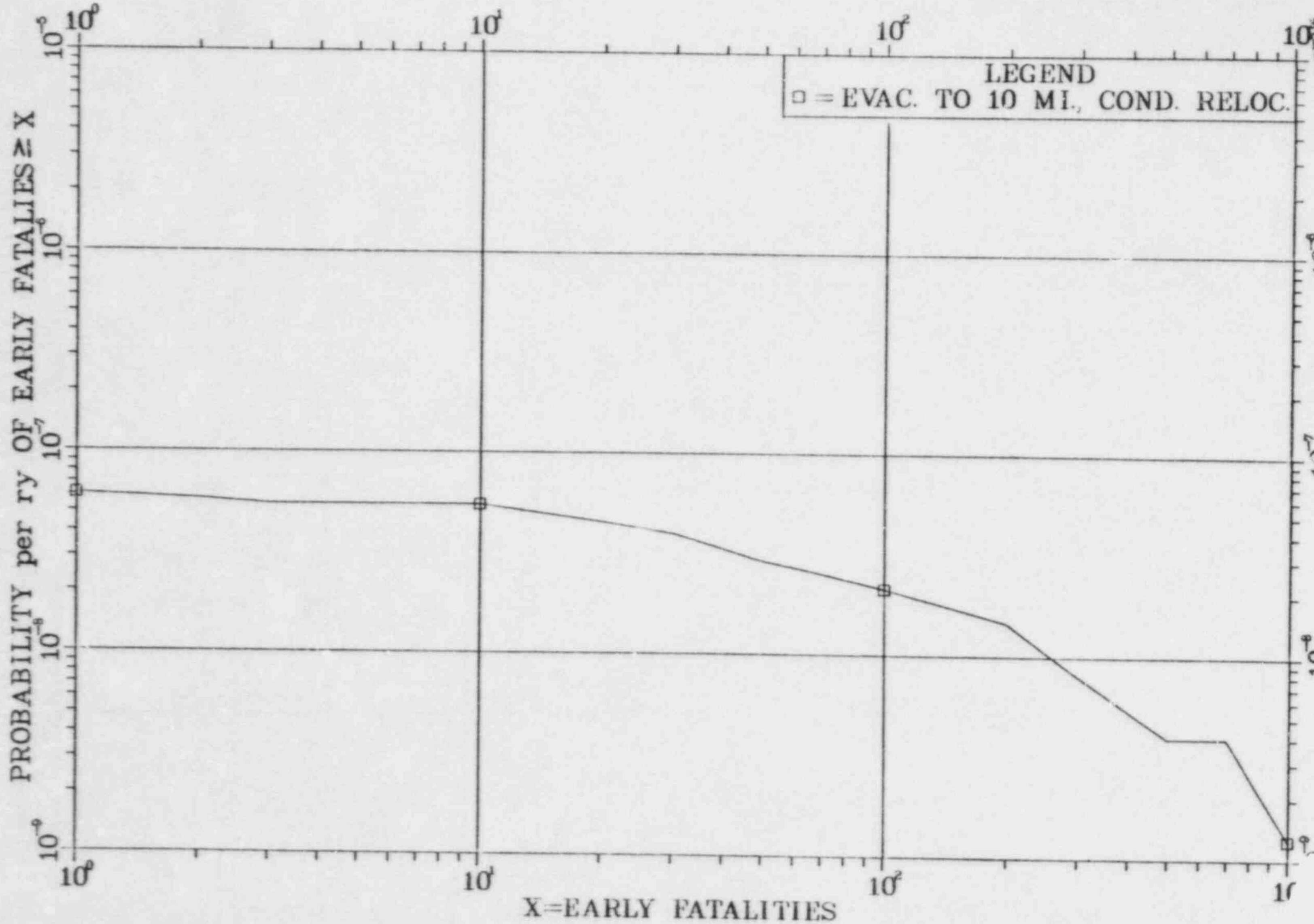

Figure 5.8 Probability distribution of early fatalities
NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties.

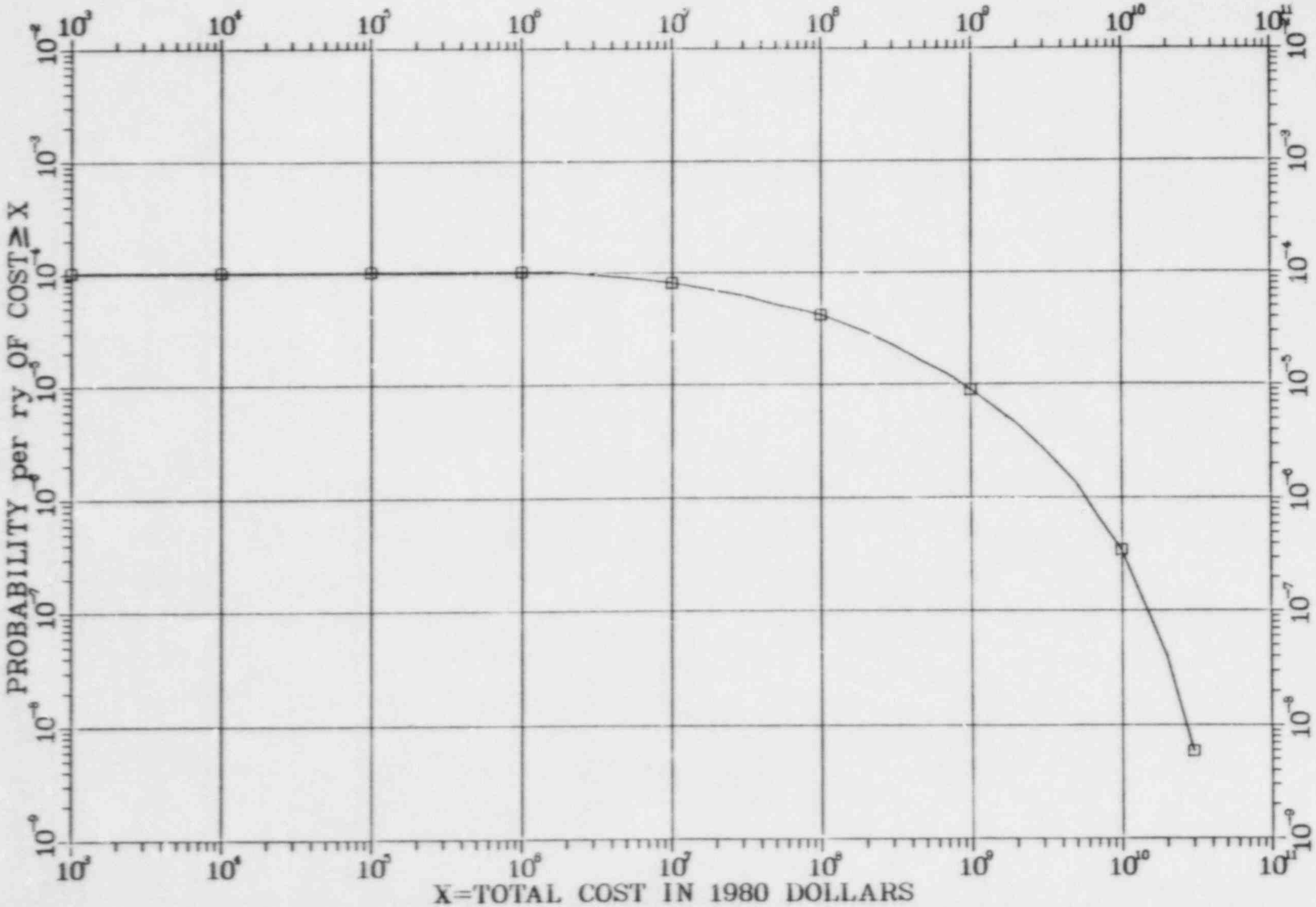

Figure 5.9 Probability distribution of mitigation measures cost
NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties.

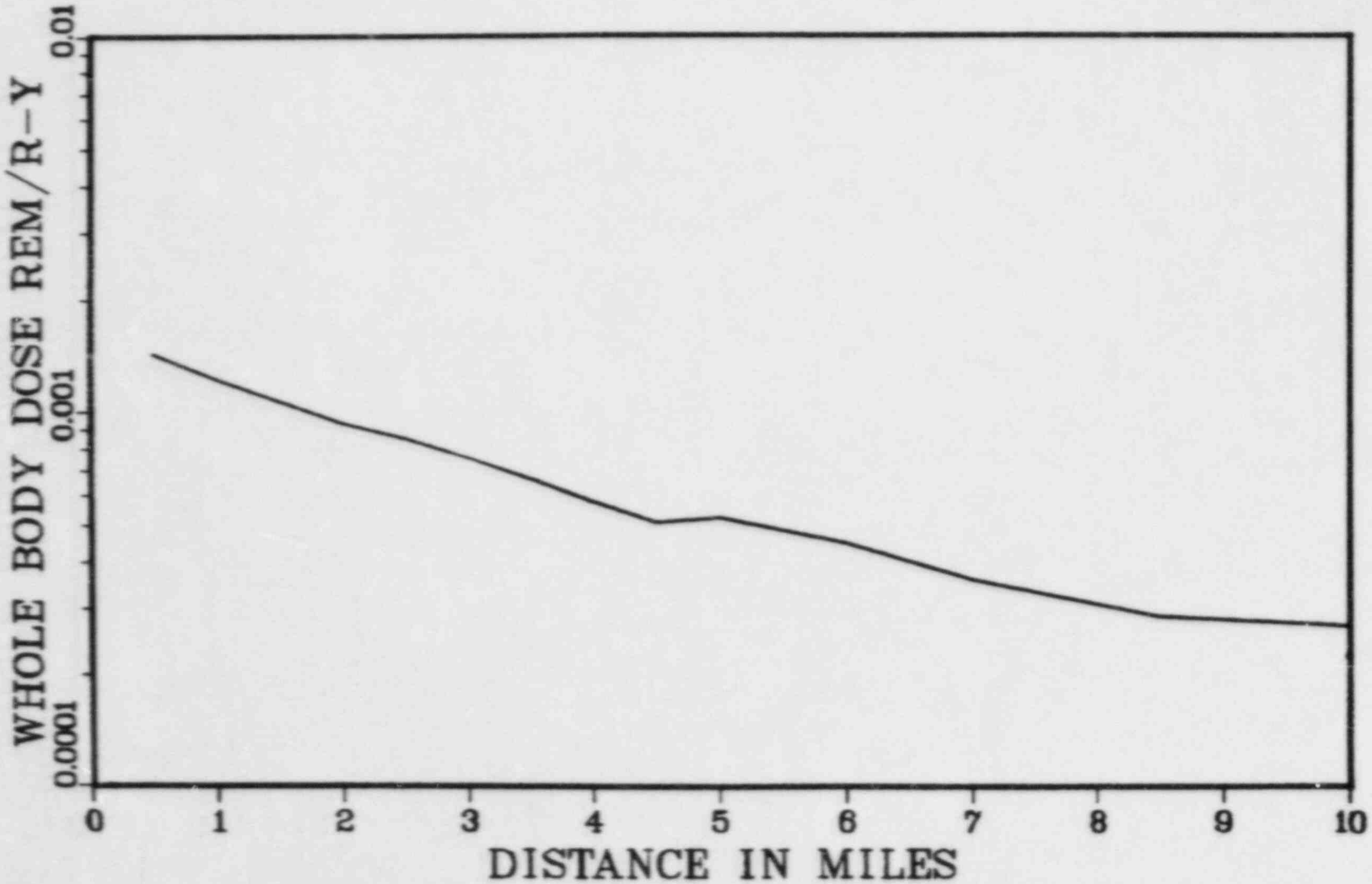

Figure 5.10 Risk of individual dose (to those downwind) versus distance
NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties

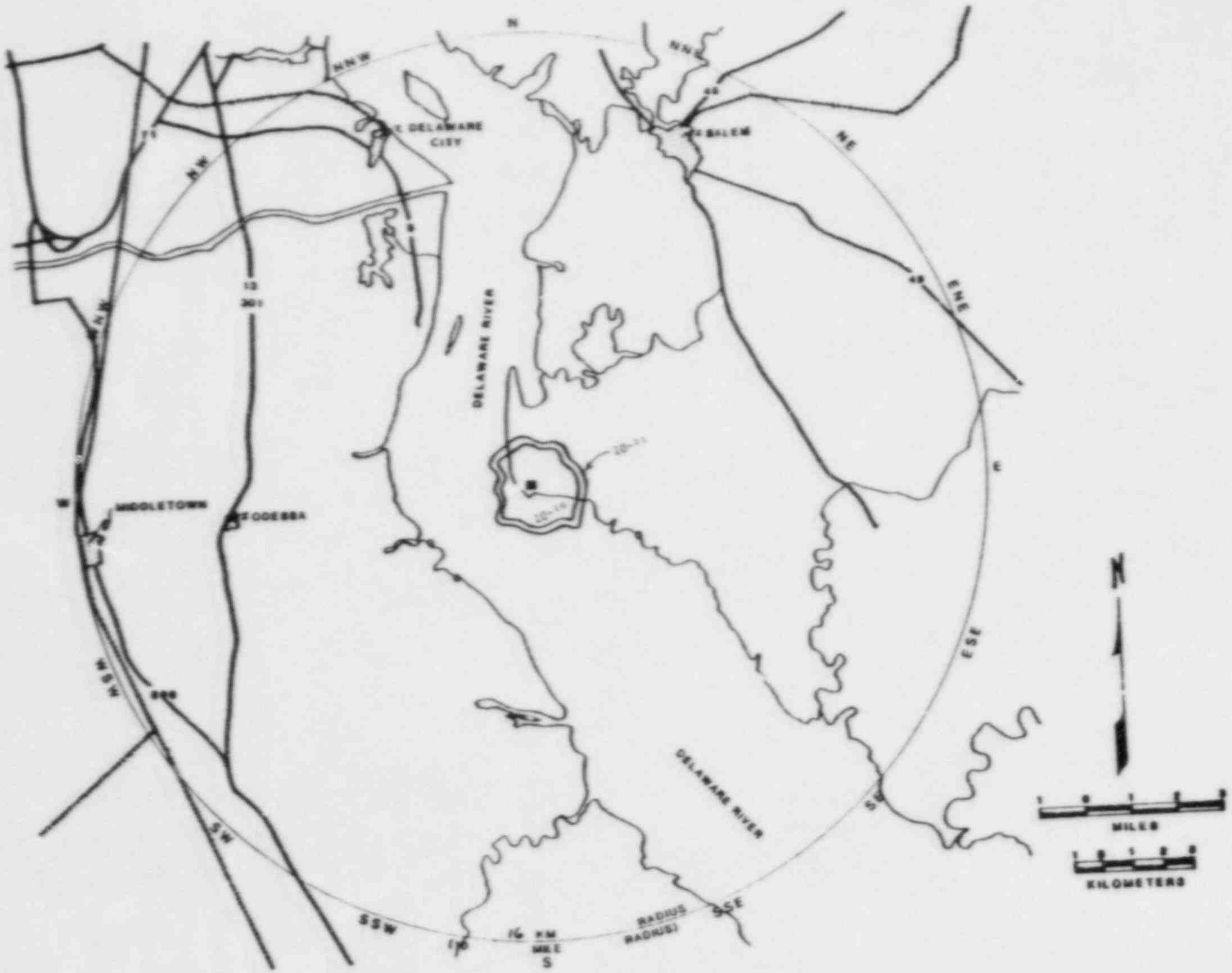

Figure 5.11 Lines of equal risk (isopleths) per reactor-year of early fatality to an individual
NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties.
Source: U.S. Geological Survey Map: Wilmington, Delaware, 1966

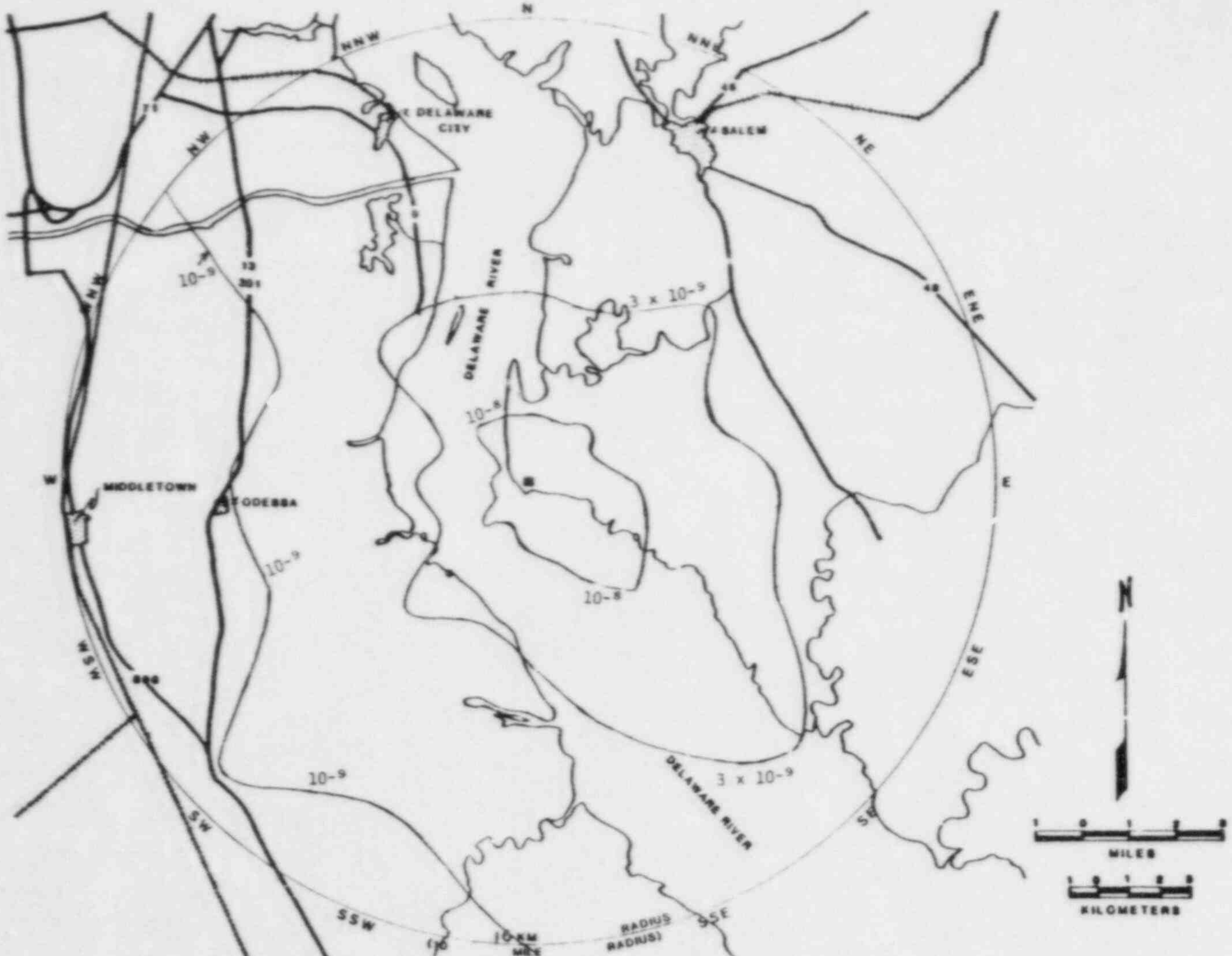

Figure 5.12 Lines of equal risk (isopleths) per reactor-year of latent cancer fatality to an individual
NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties.
Source: U.S. Geological Survey Map: Wilmington, Delaware, 1964

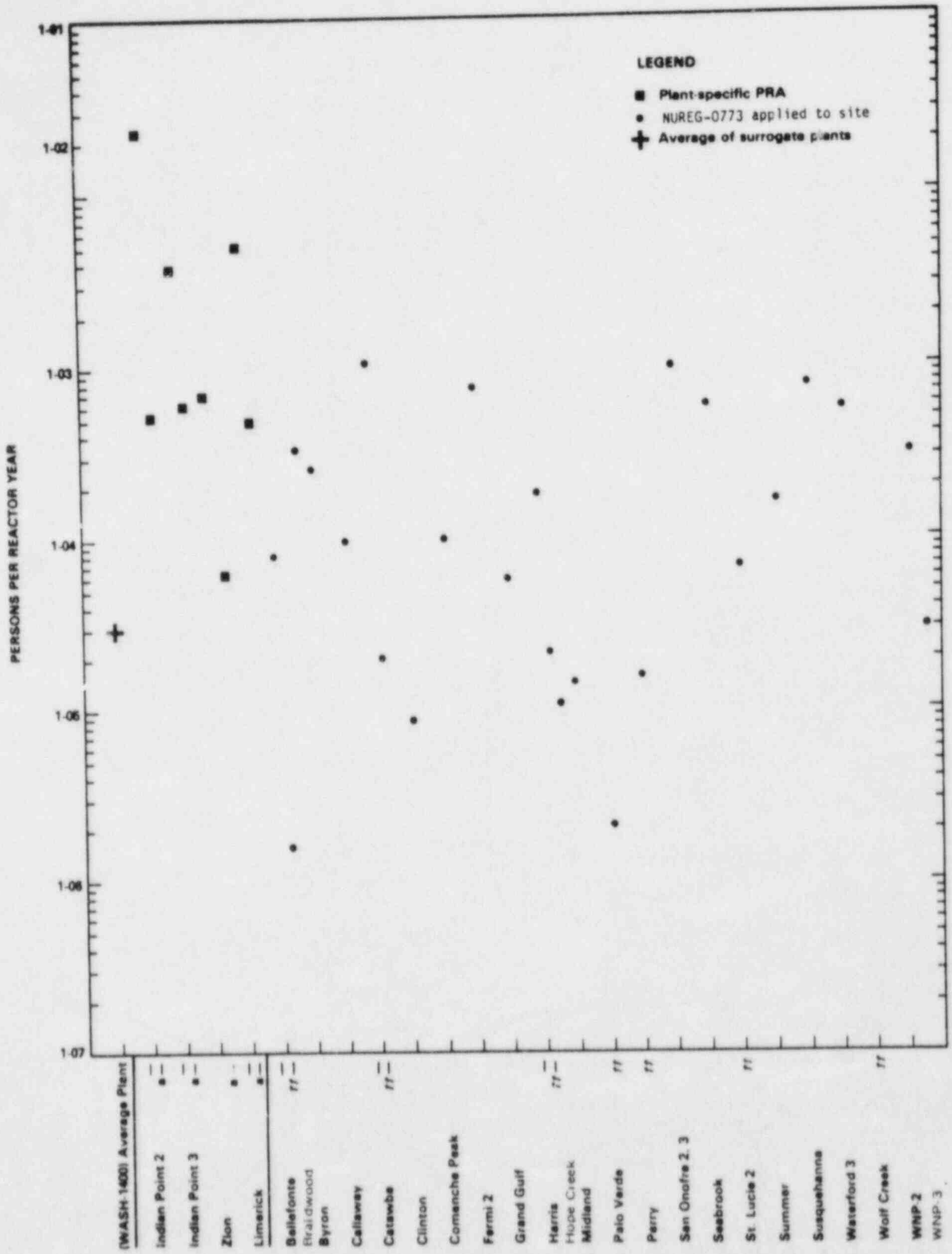

Figure 5.13 Estimated early fatality risk with supportive medical treatment (persons) from severe reactor accidents for several nuclear power plants either operating or receiving consideration for issuance of license to operate. See footnotes at end of Figure 5.21.

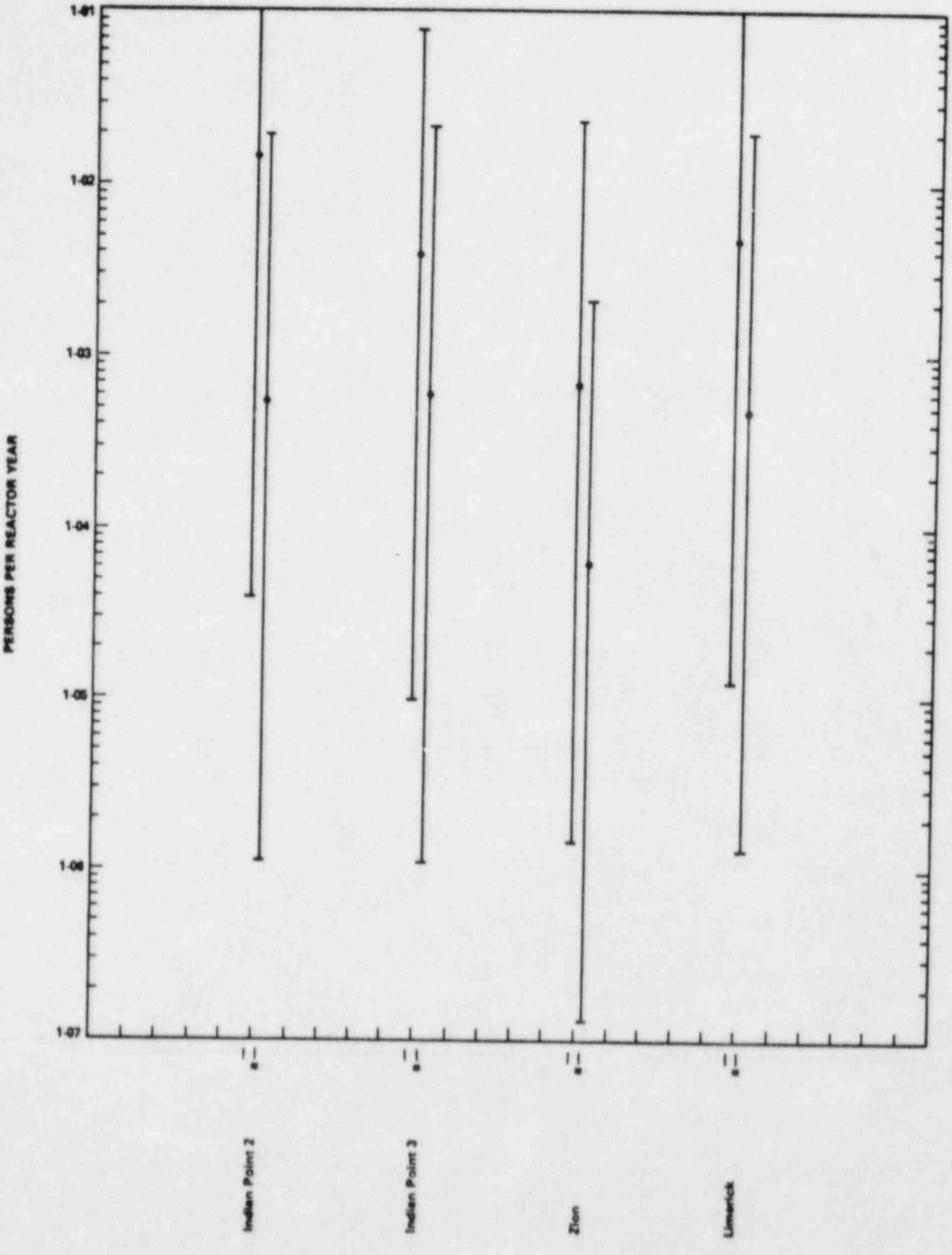

Figure 5.14 Estimated early fatality risk with supportive medical treatment (persons) from severe reactor accidents for nuclear power plants having plant-specific PRAs, showing estimated range of uncertainties. See footnotes at end of Figure 5.21.

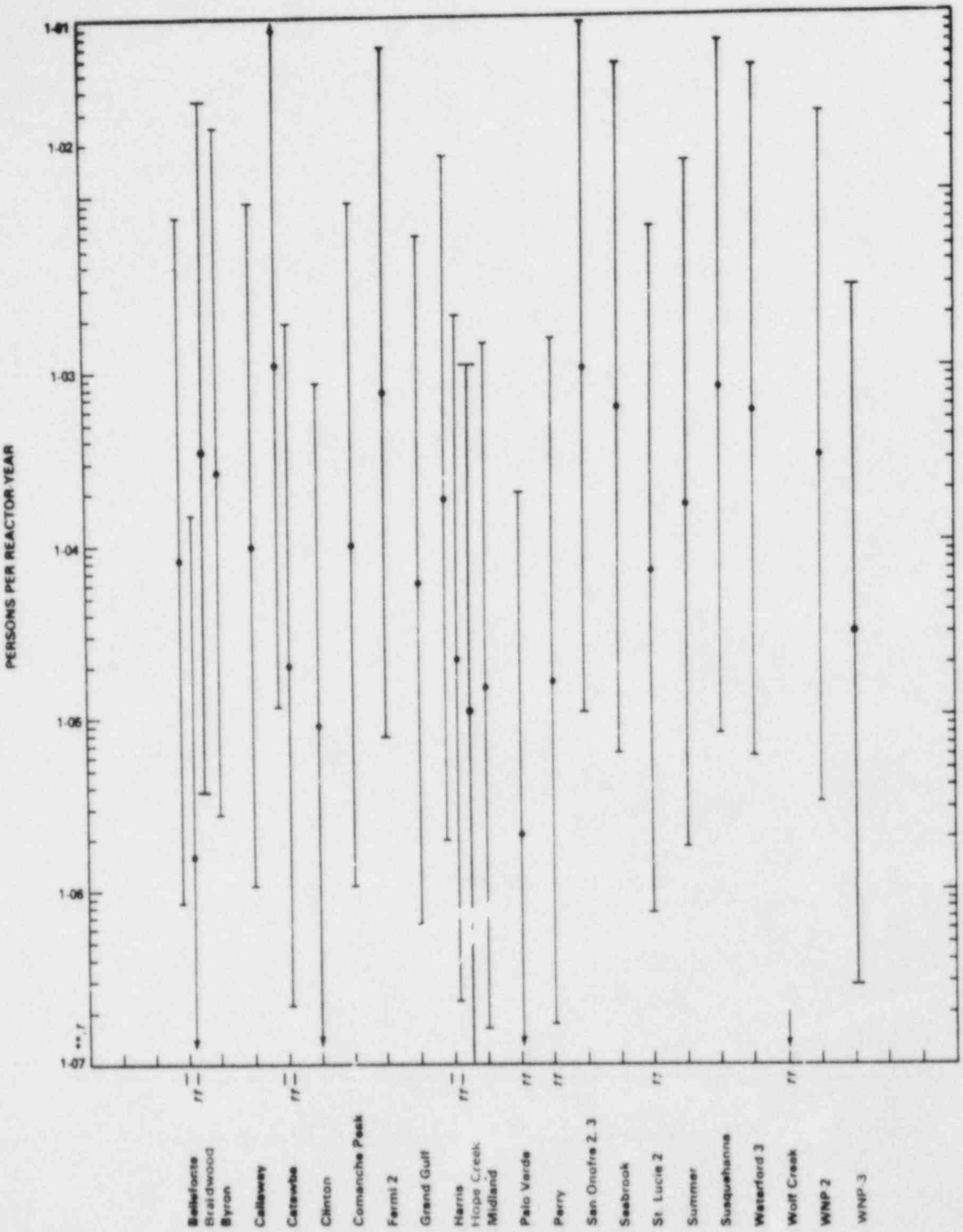

Figure 5.15 Estimated early fatality risk with supportive medical treatment (persons) from severe reactor accidents for several nuclear power plants either operating or receiving consideration for issuance of license to operate for which site-specific applications of rebaselined accident releases have been used to calculate offsite consequences. Bars are drawn to illustrate effect of uncertainty range discussed in text. See footnotes at end of Figure 5.21.

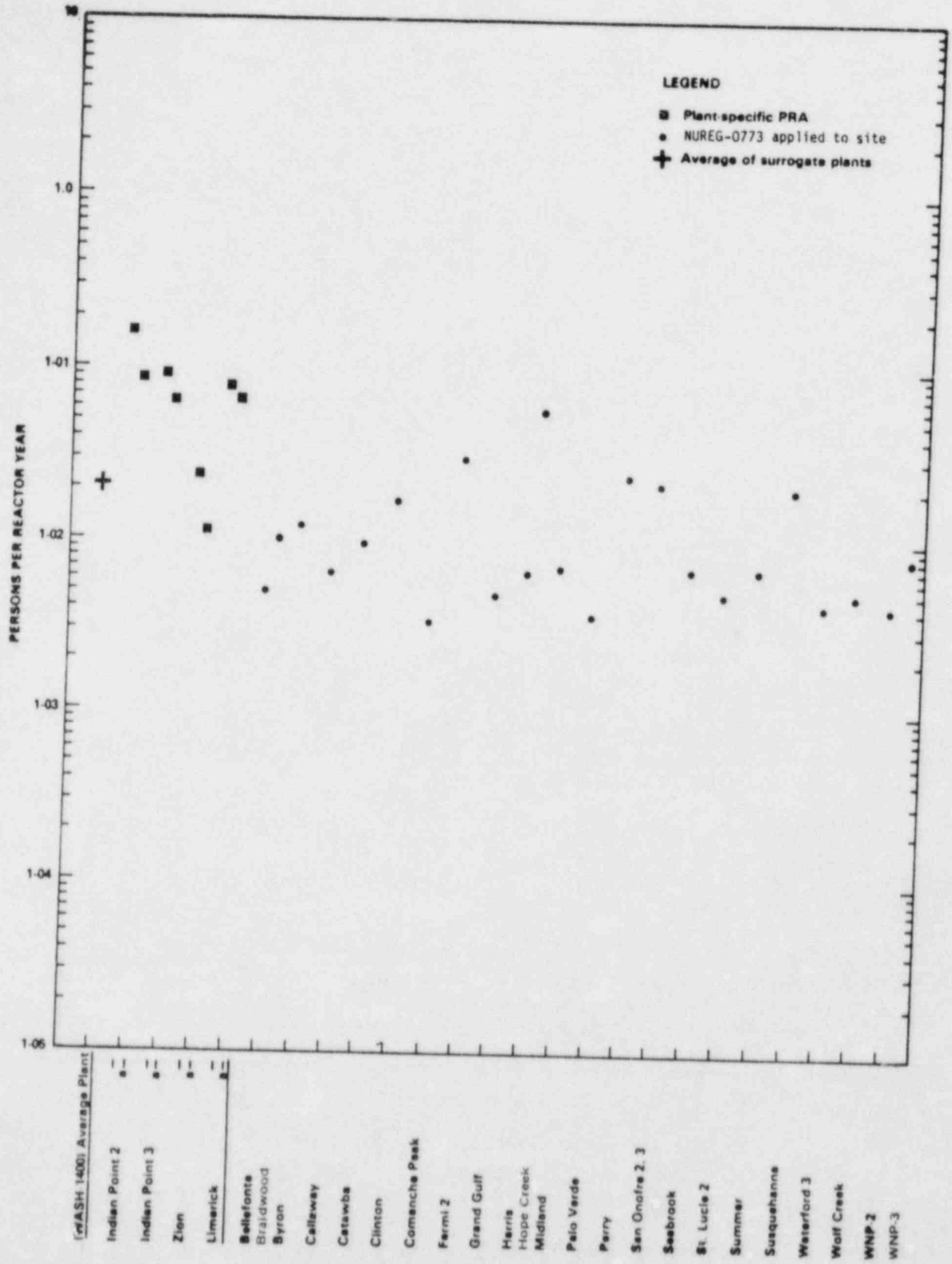

Figure 5.16 Estimated latent cancer fatality risk (persons), excluding thyroid, from severe reactor accidents for several nuclear power plants either operating or receiving consideration for issuance of license to operate. See footnotes at end of Figure 5.21.

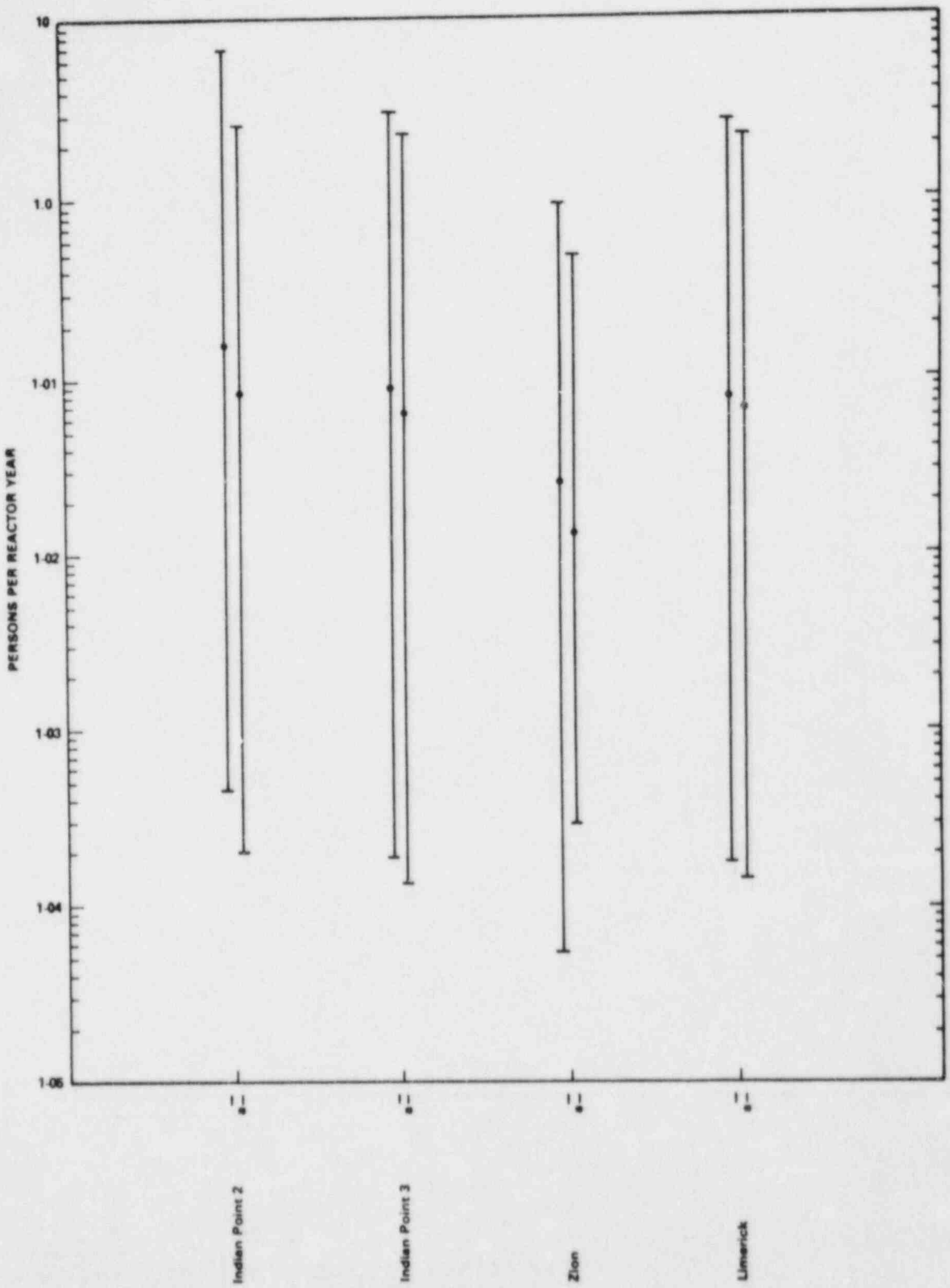

Figure 5.17 Estimated latent cancer fatality risk (persons), excluding thyroid, from severe reactor accidents for nuclear power plants having plant-specific PRAs, showing estimated range of uncertainties. See footnotes at end of Figure 5.21.

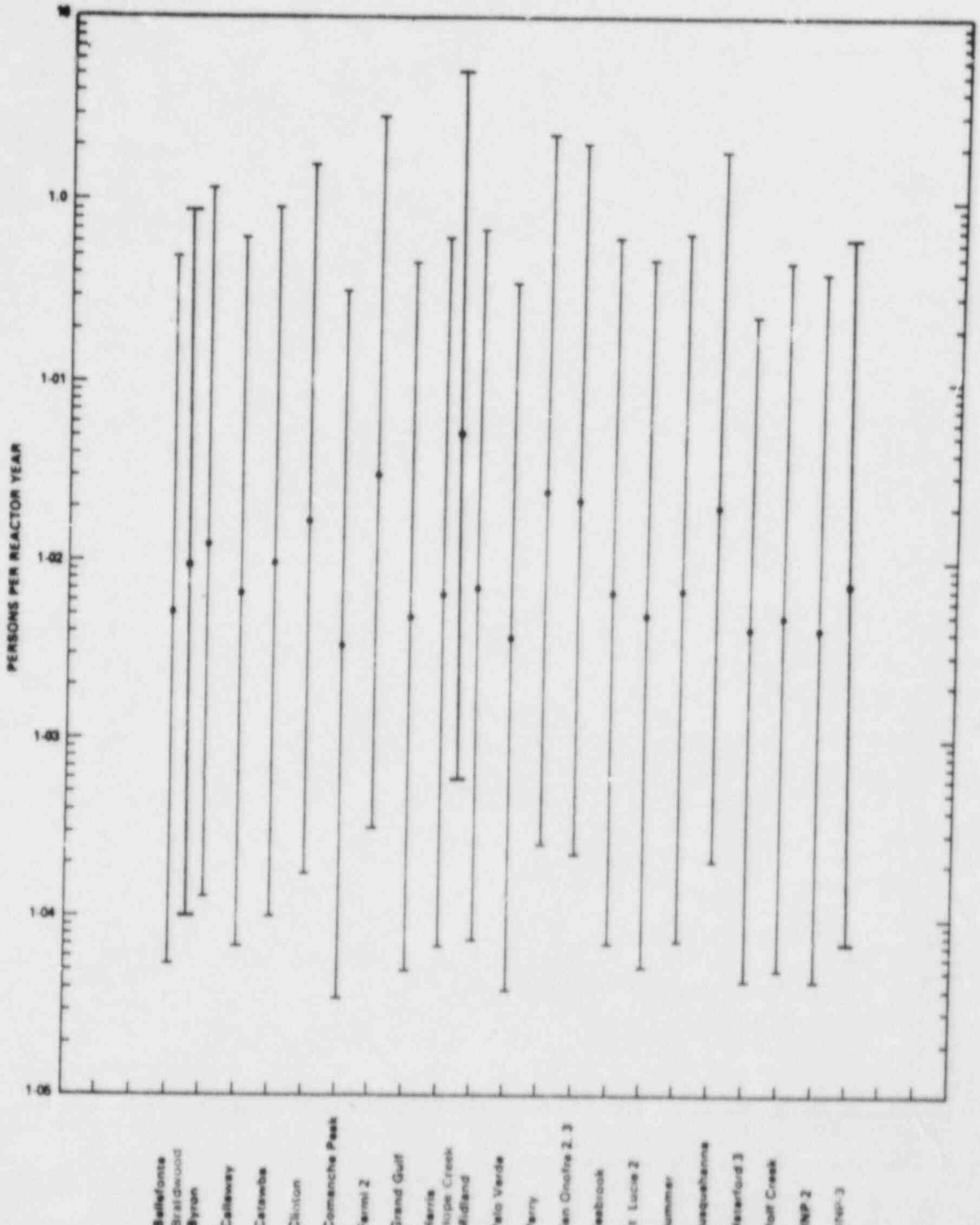

Figure 5.18 Estimated latent cancer fatality risk (persons), excluding thyroid, from severe reactor accidents for several nuclear power plants either operating or receiving consideration for issuance of license to operate for which site-specific applications of rebaselined accident releases have been used to calculate off-site consequences. Bars are drawn to illustrate effect of uncertainty range discussed in text. See footnotes at end of Figure 5.21.

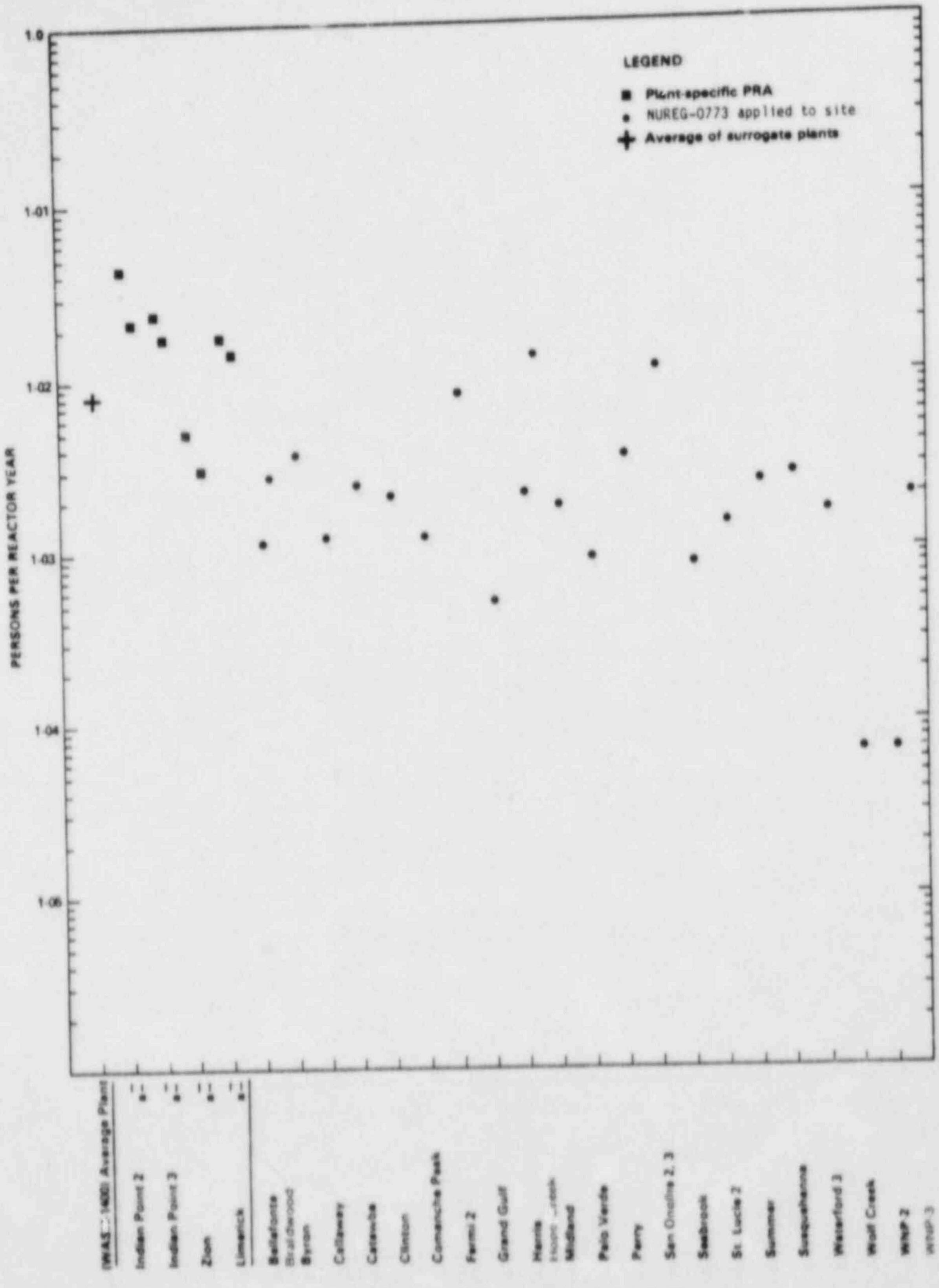

Figure 5.19 Estimated latent thyroid cancer fatality risk (persons) from severe reactor accidents for several nuclear power plants either operating or receiving consideration for issuance of license to operate. See footnotes at end of Figure 5.21.

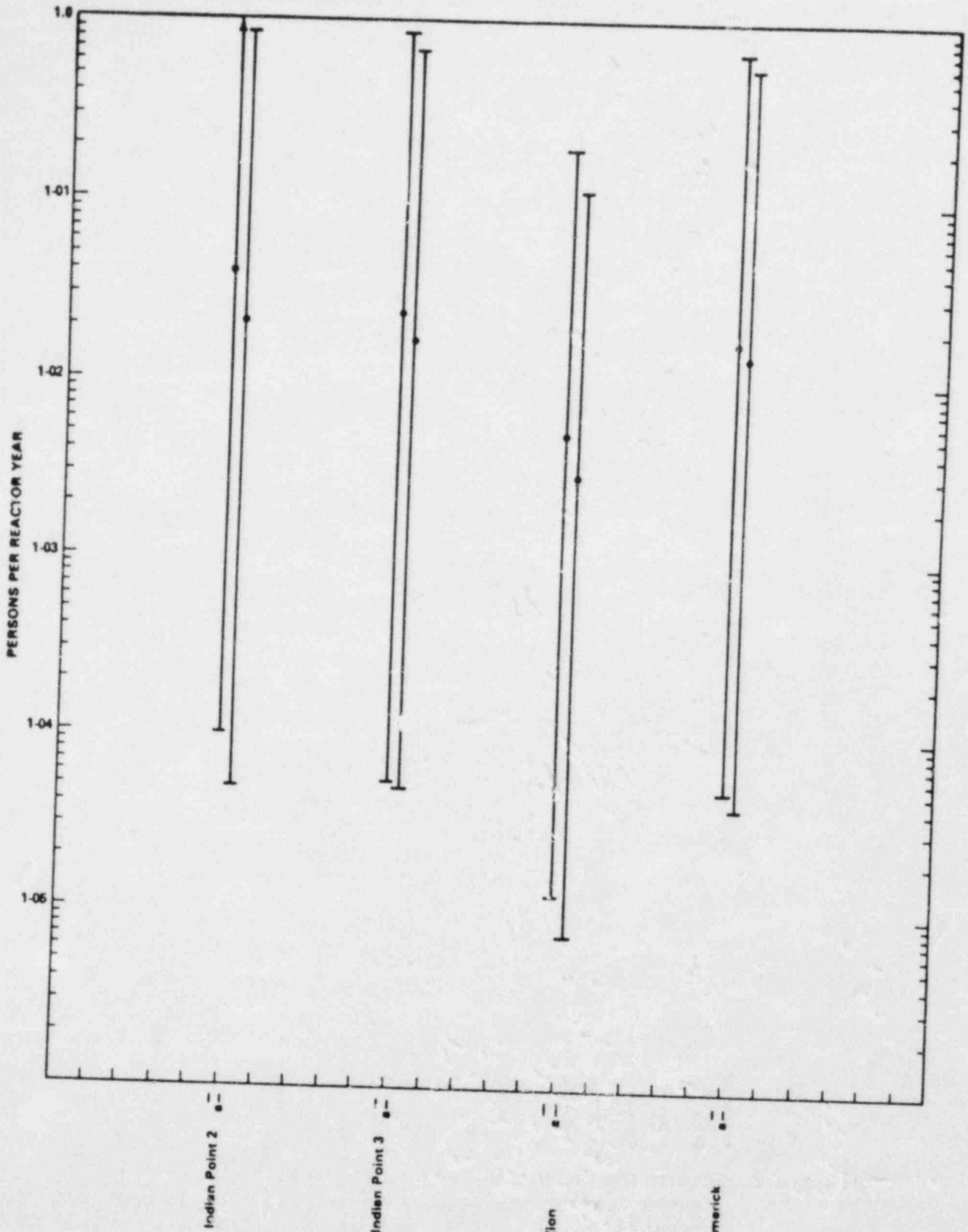

Figure 5.20 Estimated latent thyroid cancer fatality risk (persons) from severe reactor accidents for nuclear power plants having plant-specific PRAs, showing estimated range of uncertainties. See footnotes at end of Figure 5.21.

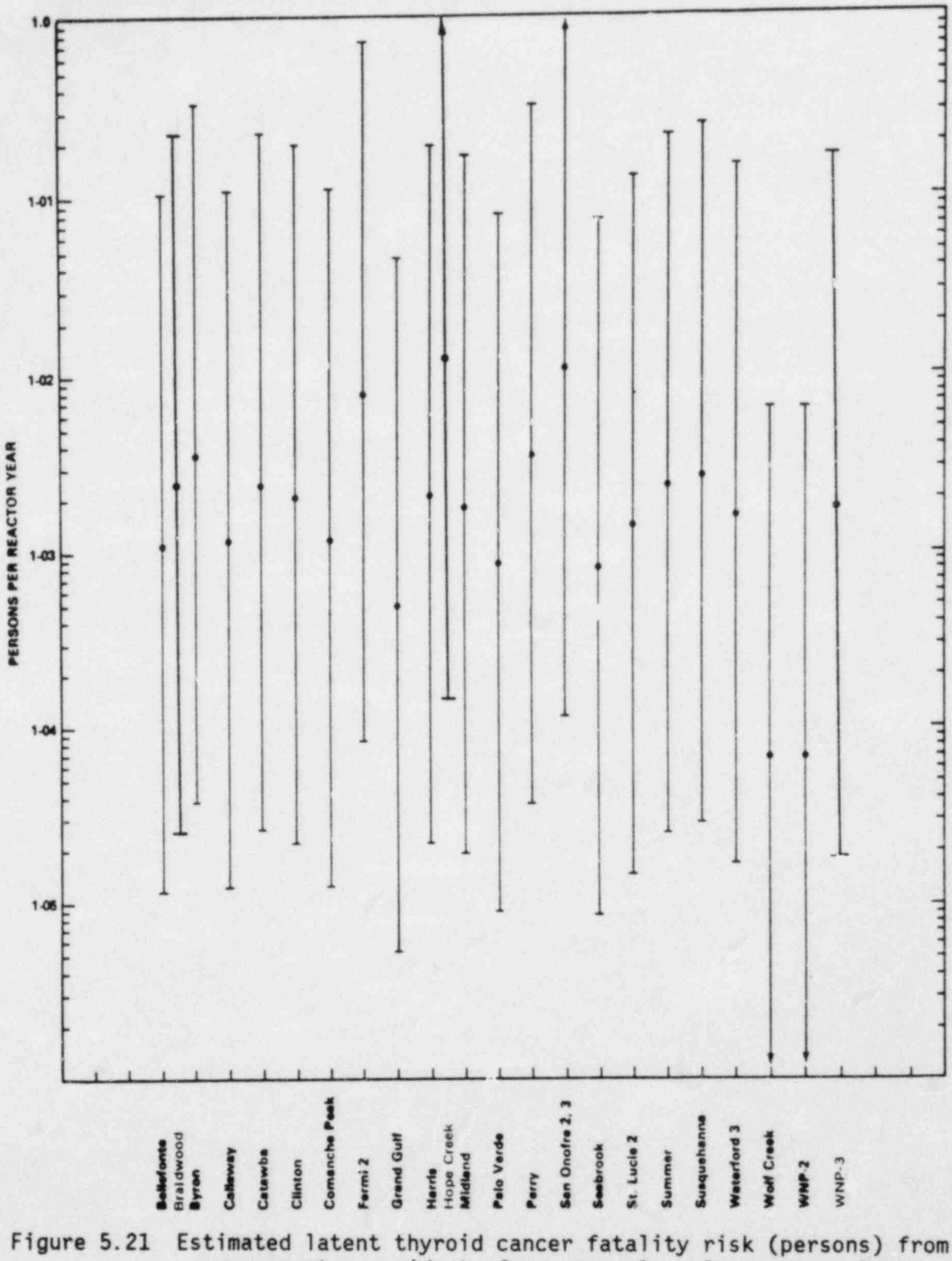

Figure 5.21 Estimated latent thyroid cancer fatality risk (persons) from severe reactor accidents from several nuclear power plants either operating or receiving consideration for issuance of license to operate for which site-specific applications of rebaselined accident releases have been used to calculate offsite consequences. Bars are drawn to illustrate effect of uncertainty range discussed in text. See footnotes on following page.

Notes for Figures 5.13 through 5.21

†Assumes evacuation to 25 mi.

††With evacuation within 10 mi and relocation from 10 to 25 mi.

[a]Excluding severe earthquakes and hurricanes.

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

Except for Indian Point, Zion, Limerick, Braidwood, Hope Creek, and WNP-3, risk analyses for other plants in these figures are based on WASH-1400 generic source terms and probabilities for severe accidents and do not include external event analyses. The staff briefly reviewed Braidwood, Hope Creek, and WNP-3 to determine plant-specific release category probabilities considering internal events only. Any or all of the values could be under- or overestimates of the true risks.

$1\text{-}01 = 1 \times 10^{-1}$.

Table 5.1 Public water supplies in the site region

No.	Distance: Kilometer	Distance: Mile	Town	Population served	Average output: Million liters/day	Average output: Million gallons/day	Source of water
1	14.5	9	Salem, New Jersey	9,000	6.4	1.7	About 2/3 of water consumed is surface water pumped from the Quinton pumping station about 3 mi east of town and 9 mi northeast of the site. Remainder is obtained from four wells, ranging in depth from 80 ft to 168 ft, located east of Salem.
2	22.5	14	Pennsville, New Jersey	10,500	-	-	Four wells ranging in depth from 105 ft to 240 ft. The wells are probably completed in the Magothy Formation.
3	27.4	17	Penns Grove, New Jersey	8,000	-	-	Two wells, 292 ft and 360 ft deep. The water probably comes from the Potomac Group.
4	27.4	17	Woodstown, New Jersey	3,000	-	-	Eight wells; six are about 100 ft deep and the others are about 300 ft and 350 ft deep.
5	35.4	22	Elmer, New Jersey	2,500	-	-	Three wells; two are 80 ft deep and the third is 500 ft deep. The shallow wells probably tap the Mount Laurel-Wenonah Formation.
6	25.7	16	Bridgeton, New Jersey	22,000	-	-	A total of 12 wells, some of which are no longer in use, range in depth from 75 ft to 129 ft. They are completed in the Cohonsey Sand.
7	17.7	11	Smyrna, Delaware	-	1.0	0.27	Two wells, 20 ft and 95 ft deep, supply the town. The shallower well is used for standby purposes.
8	20.9	13	Clayton, Delaware	825	4.5	1.2	One well, 272 ft deep, is the source of water supply.
9	16.1	10	Middletown, Delaware	2,000	0.8	0.2	Three wells, having depths of 100 ft, 200 ft, and 500 ft, supply the town.

Table 5.1 (Continued)

No.	Distance: Kilometer	Distance: Mile	Town	Population served	Average output: Million liters/day	Average output: Million gallons/day	Source of water
10	14.5	9	Delaware City, Delaware	1,500	-	-	Two wells, one 26 ft deep in the Wenonah Formation and the other in the Magothy Formation, supply the town.
11	22.5	14	New Castle, Delaware	-	-	-	The town obtains water from a shallow infiltration gallery system located in Pleistocene deposits.

NOTE: 1 ft = 0.3048 m; 1 mi = 1.609 km.

Source: ER-OL, Table 5.3-1.

Table 5.2 Estimated and measured flood levels

Flood event	Still water level (MSL): Meter	Still water level (MSL): Foot
Estimated		
10-year flood*	2.1	7.0
50-year flood*	2.5	8.2
100-year flood*	2.7	8.9
500-year flood*	4.0	13.2
Probable maximum hurricane surge	7.6	24.8
Measured		
Storm, November 25, 1950**	2.6	8.5
Hurricane, August 1933**	2.4	8.0
Storm, March 6, 1962**	2.3	7.5

*Federal Insurance Administration Study for Township of Alloways Creek, Salem County, New Jersey.
**Reedy Point Tidal Station.

Table 5.3 Annual average densities of eggs, number potentially entrained per year, number surviving to end of the 0- to 1-year age class, and loss of potential fishery production

	Species			
Parameter	Anchovy	Weakfish	Silversides	Other
Average egg density (no./m^3)				
1974	2.926	0.007	-	0.002
1975	0.520	0.001	-	-
1976	3.16	0.002	-	-
1977	13.730	0.057	0.008	0.001
Entrainment (no./year)				
1974	1.913×10^{8}	4.58×10^{5}	-	1.31×10^{5}
1975	3.40×10^{7}	6.54×10^{4}	-	-
1976	2.066×10^{8}	1.31×10^{5}	-	-
1977	8.977×10^{8}	3.73×10^{6}	5.23×10^{5}	6.54×10^{4}
No. surviving to 0- to 1-year age class				
1974	2.68×10^{4}	64.0	-	18.0
1975	4.76×10^{3}	9.0	-	-
1976	2.89×10^{4}	18.0	-	-
1977	1.25×10^{4}	522.0	73.0	9.0
Average weight during 1st year (g (lb))	2.7 (0.006)	3.2 (0.007)	2.7 (0.006)	3.6 (0.008)
Potential annual production lost (kg (lb))				
1974	73 (161)	0.4 (1)	-	<2 (<4)
1975	13 (29)	0.4 (1)	-	-
1976	78 (173)	0.4 (1)	-	-
1977	342 (754)	2 (4)	<0.4 (<1)	<0.4 (<1)
Mean	130 (287)	1 (2)	<0.4 (<1)	<0.4 (<1)

Table 5.4 Annual average densities of fish larvae during 1974-1977, number potentially entrained per year, number surviving to end of the 0- to 1-year age class, and loss of potential fishery production

Parameter	Species					
	Anchovy	Goby	Weakfish	Silversides	Croaker	Other
Average larval density (no./m^3)						
1974	0.735	0.148	0.001	-	0.002	0.005
1975	0.683	0.310	0.021	0.002	0.009	-
1976	1.699	0.416	0.006	0.005	0.020	0.005
1977	10.902	0.121	0.357	0.017	-	0.007
Entrainment (no./year)						
1974	4.81×10^7	9.68×10^6	6.54×10^4	-	1.31×10^5	3.27×10^6
1975	4.47×10^7	2.03×10^7	1.37×10^6	1.31×10^5	5.88×10^5	-
1976	1.11×10^8	2.72×10^7	3.92×10^5	3.27×10^6	1.31×10^6	3.27×10^6
1977	7.13×10^8	7.91×10^6	2.33×10^7	1.11×10^6	-	4.58×10^6
No. surviving to 0- to 1-year age class						
1974	6.734×10^4	1.355×10^4	92	-	183	4,578
1975	6.258×10^4	2.842×10^4	1,918	183	823	-
1976	1.551×10^5	3.808×10^4	549	4,578	1,834	4,578
1977	9.982×10^5	1.11×10^4	3.262×10^4	1,554	-	641
Average weight during 1st year (g (lb))	2.7 (0.006)	2.7 (0.006)	3.2 (0.007)	2.7 (0.006)	6.3 (0.014)	3.6 (0.008)
Potential annual production lost (kg (lb))						
1974	183 (404)	37 (81)	0.4 (1.0)	-	1 (3)	17 (37)
1975	170 (375)	77 (170)	6 (13)	0.4 (1)	5 (12)	-
1976	423 (932)	103 (228)	2 (4)	12 (27)	12 (26)	17 (37)
1977	2,717 (5,989)	27 (60)	103 (228)	4 (9)	-	2 (5)
Mean	873 (1,925)	61 (134)	28 (62)	4 (9)	5 (10)	9 (20)

Table 5.5 Potential fishery production lost from entrainment of eggs and larvae of the major fish species in the Delaware River near the Hope Creek site

Species	Annual average (kg (lb))			
	Eggs	Larvae	Total	Standard deviation
Anchovy	135 (287)	873 (1,925)	1,003 (2,212)	±1,379 (3,040)
Goby	-	61 (134)	61 (134)	±35 (78)
Weakfish	0.9 (2.0)	28 (62)	28 (62)	±51 (113)
Croaker	-	5 (10)	5 (10)	±5 (12)
Silversides	0.4 (1.0)	4 (9.0)	5 (10)	±6 (13)
Other	0.4 (1.0)	9 (20)	10 (21)	±9 (20)

NOTE: The standard deviation of the total annual average was calculated from the values of 1974-1977.

Table 5.6 Total potential of fishery production lost as a result of entrainment of phytoplankton, zooplankton, and icthyoplankton at the Hope Creek station

Organism	Total lost	
	Kilogram	Pound
Ichthyoplankton	1,112	2,452
Phytoplankton	1,377	3,036
Zooplankton	1,312	2,892

NOTE: Total entrainment impact equals 3,801 kg (8,380 lb) of potential fishery production.

Table 5.7 Actual and estimated numbers and weights of the organisms commonly impinged during 1977 and 1978 at the Salem Generating Station

Species	Actual no.	Estimated no.	Estimated wt. (kg)*	Length (mm) min-max
1977**				
Blueback herring	217	38,900	160	58-113
Bay anchovy	11,307	5,507,200	14,210	23-93
White perch	2,568	360,100	3,110	38-223
Weakfish	8,086	2,435,900	6,990	23-168
Spot	6,937	1,664,500	10,420	23-193
Total of common fish	29,115	10,006,600	34,890	-
Total of all fish	41,845	12,773,100	52,600	-
Blue crab	1,720	420,100	19,470	13-198
1978***				
Blueback herring	3,460	305,300	960	38-278
Bay anchovy	14,544	1,819,500	4,370	13-98
White perch	5,743	592,400	8,210	38-293
Weakfish	51,018	8,104,300	11,350	18-253
Spot	1,186	119,300	1,770	21-198
Total of common fish	75,951	10,940,800	26,660	-
Total of all fish	93,884	12,837,400	40,120	-
Blue crab	3,010	372,600	9,290	8-208

*To convert to pounds, multiply kilograms by 2.20.
**532 samples; 1,596 minimum sampled.
***2,195 samples; 3,791 minimum sampled.

Source: PSEG, 1980b, Table 5.3.

Table 5.8 Estimated real estate and gross receipts and franchise taxes* to be paid on Hope Creek Generating Station (millions of dollars)

Year	Real estate taxes to be paid to Lower Alloways Creek	Gross receipts, franchise taxes to be paid to state
1987	94	0.789
1988	100	0.844
1989	97	0.899
1990	116	0.958
1991	110	1.020
1992	117	1.086

*Dollars are valued in year stated.

Table 5.9 Incidence of job-related mortalities

Occupational group	Mortality rates (premature deaths per 10^5 person-years)
Underground metal miners*	~1300
Uranium miners*	420
Smelter workers*	190
Mining**	61
Agriculture, forestry, and fisheries**	35
Contract construction**	33
Transportation and pulbic utilities**	24
Nuclear-plant worker***	23
Manufacturing**	7
Wholesale and retail trade**	6
Finance, insurance, and real estate**	3
Services**	3
Total private sector**	10

*The President's Report on Occupational Safety and Health, "Report on Occupational Safety and Health by the U.S. Department of Health, Education, and Welfare," E. L. Richardson, Secretary, May 1972.

**U.S. Bureau of Labor Statistics, "Occupational Injuries and Illness in the United States by Industry, 1975," Bulletin 1981, 1978.

***The nuclear-plant workers' risk is equal to the sum of the radiation-related risk and the nonradiation-related risk. The estimated occupational risk associated with the industry-wide average radiation dose of 0.8 rem is about 11 potential premature deaths per 10^5 person-years due to cancer, based on the risk estimators described in the following text. The average non-radiation-related risk for seven U.S. electrical utilities over the period 1970-1979 is about 12 actual premature deaths per 10^5 person-years as shown in Figure 5 of the paper by R. Wilson and E. S. Koel, "Occupational Risks of Onatario Hydro's Atomic Radiation Workers in Perspective," presented at Nuclear Radiation Risks, a Utility-Medical Dialog, sponsored by the International Institute of Safety and Health in Washington, D.C., September 22-23, 1980. (Note that the estimate of 11 radiation-related premature cancer deaths describes a potential risk rather than an observed statistic.)

Table 5.10 (Summary Table S-4) Environmental impact of transportation of fuel and waste to and from one light-water-cooled nuclear power reactor[1]

NORMAL CONDITIONS OF TRANSPORT

	Environmental impact
Heat (per irradiated fuel cask in transit)	250,000 Btu/hr.
Weight (governed by Federal or State restrictions)	73,000 lbs. per truck; 100 tons per cask per rail car
Traffic density:	
Truck	Less than 1 per day.
Rail	Less than 3 per month.

Exposed population	Estimated number of persons exposed	Range of doses to exposed individuals [2] (per reactor year)	Cumulative dose to exposed population (per reactor year) [3]
Transportation workers	200	0.01 to 300 millirem	4 man-rem.
General public:			
Onlookers	1,100	0.003 to 1.3 millirem	3 man-rem.
Along Route	600,000	0.0001 to 0.06 millirem	

ACCIDENTS IN TRANSPORT

	Environmental risk
Radiological effects	Small
Common (nonradiological) causes	1 fatal injury in 100 reactor years; 1 nonfatal injury in 10 reactor years; $475 property damage per reactor year.

[1] Data supporting this table are given in the Commission's "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants," WASH-1238, December 1972, and Supp. I, NUREG-75/038 April 1975. Both documents are available for inspection and copying at the Commission's Public Document Room, 1717 H St. NW., Washington, D.C., and may be obtained from National Technical Information Service, Springfield, Va. 22161. WASH-1238 is available from NTIS at a cost of $5.45 (microfiche, $2.25) and NUREG-75/038 is available at a cost of $3.25 (microfiche, $2.25).

[2] The Federal Radiation Council has recommended that the radiation doses from all sources of radiation other than natural background and medical exposures should be limited to 5,000 millirem per year for individuals as a result of occupational exposure and should be limited to 500 millirem per year for individuals in the general population. The dose to individuals due to average natural background radiation is about 130 millirem per year.

[3] Man-rem is an expression for the summation of whole body doses to individuals in a group. Thus, if each member of a population group of 1,000 people were to receive a dose of 0.001 rem (1 millirem), or if 2 people were to receive a dose of 0.5 rem (500 millirem) each, the total man-rem dose in each case would be 1 man-rem.

[4] Although the environmental risk of radiological effects stemming from transportation accidents is currently incapable of being numerically quantified, the risk remains small regardless of whether it is being applied to a single reactor or a multireactor site.

Table 5.11 Reproduction of applicant's preoperational radiological environmental monitoring program summary table*

EXPOSURE PATHWAY	STATION CODE	LOCATION[(3)]	COLLECTION METHOD AND FREQUENCY	TYPE AND FREQUENCY OF ANALYSES
I. AIRBORNE				
(a) Particulates	2S2	0.4 mi NNE of vent	Sample collected every week along with filter change using a low volume air sampler	Gross beta analysis performed on[(2)] each weekly sample Gamma scan analysis performed on a sample composited over a calendar quarter
	10D1	3.9 mi SSW of vent		
	16E1	4.1 mi NNW of vent		
	1F1	5.8 mi N of vent		
	2F2	8.7 mi NNE of vent		
	3H3[(1)]	110 mi NE of vent		

(1) Control station.
(2) Gamma spectrometry is performed if gross beta exceeds ten times the control station value.
(3) 1.61 km = 1 mile.

Source: ER-OL, Table 6.1-3, through Amendment 3, April 30, 1984

Table 5.11 (Continued)

EXPOSURE PATHWAY	STATION CODE	LOCATION[3]	COLLECTION METHOD AND FREQUENCY	TYPE AND FREQUENCY OF ANALYSES
(b) Iodine	2S2	0.4 mi NNE of vent	A TEDA impregnated charcoal flow-through cartridge is connected to air particulate air sampler and is collected weekly	Iodine 131 analyses are performed weekly
	10D1	3.9 mi SW of vent		
	16E1	4.1 mi NNW of vent		
	2F2	NJ; 8.7 mi NNE of vent		
	3H3[1]	110 mi NE of vent		

(1) Control station.
(3) 1.61 km = 1 mile.

Source: ER-OL, Table 6.1-3, through Amendment 3, April 30, 1984

Table 5.11 (Continued)

EXPOSURE PATHWAY	STATION CODE	LOCATION[3]	COLLECTION METHOD AND FREQUENCY	TYPE AND FREQUENCY OF ANALYSES
II. SOIL				
	2S2	0.4 mi NNE of vent	10 soil plugs to a depth of 15 cm (6 in.) over an area of 2.3 m^2 ($25\ ft^2$) are composited and sealed in a plastic bag at each location[4]. A sample will be collected from each location once every 3 years	Gamma spectrometry performed on collection SR-90 analyses on one sample from each location on collection
	5D1	3.5 mi E of vent		
	10D1	3.9 mi SSW of vent		
	2E1	4.4 mi NNE of vent		
	16E1	4.1 mi NNW of vent		
	1F1	5.8 mi N of vent NJ; 8.7 mi NNE of vent		
	2F1	5 mi NNE of vent		
	3G1[1]	16.6 mi NE of vent		
	3H3[1]	10 mi NE of station		

(1) Control station.
(2) Gamma spectrometry is performed if gross beta exceeds ten times the control station value.
(3) 1.61 km = 1 mile.
(4) Soil samples are taken in accordance with procedures outlined in HASL-300 (Rev. 5/73). If a suitable sample cannot be obtained at a location, a sample is obtained from a new location. The NRC is notified in writing of the new sample location.

Source: ER-OL, Table 6.1-3, through Amendment 3, April 30, 1984

Table 5.11 (Continued)

EXPOSURE PATHWAY	STATION CODE	LOCATION	COLLECTION METHOD AND FREQUENCY	TYPE AND FREQUENCY OF ANALYSES
III. DIRECT				
	2S2	0.4 mi NNE of vent	Four dosimeters will be collected from each location monthly and quarterly	Gamma dose-monthly Gamma dose-quarterly
	5S1	1.0 mi E of vent		
	6S2	0.2 mi EES of vent		
	7S1	0.12 mi SE of vent		
	10S1	0.14 mi SSW of vent		
	11S1	0.09 mi Sw of vent		
	5D1	3.5 mi E of vent		
	10D1	3.9 mi WNW of vent		
	14D1	3.4 mi WNW of vent		
	2E1	4.4 mi NNE of vent		
	3E1	4.1 mi NE of vent		
	13E1	4.2 mi NE of vent		
	16E1	4.1 mi NNW of vent		
	1F1	5.8 mi N of vent		
	2F2	8.7 mi NNE of vent		
	5F1	8.0 mi E of vent		
	6F1	6.4 mi ESE of vent		
	7F2	9.1 mi Se of vent		
	11F1	5.2 mi SW of vent		
	13F1	9.8 mi W of vent		
	3G1 (1)	17 mi NE of vent		
	2H1 (1)	34 mi NNE of vent		
	3H1 (1)	32 mi NE of vent		
	3H3 (1)	110 mi NE of vent		

(1) Control station.

Source: ER-OL, Table 6.1-3, through Amendment 3, April 30, 1984

Table 5.11 (Continued)

EXPOSURE PATHWAY	STATION CODE	LOCATION	COLLECTION METHOD AND FREQUENCY	TYPE AND FREQUENCY OF ANALYSES
	4D2	3.7 mi ENE of vent	Four dosimeters will be collected from each location quarterly	Gamma dose-quarterly
	9E1	4.2 mi S of vent		
	11E2	5.0 mi SW of vent		
	12E1	4.4 mi WSW of vent		
	2F5	7.4 mi NNE of vent		
	3F2	5.1 mi NE of vent		
	3F3	8.6 mi NE of vent		
	10F2	5.8 mi SSW of vent		
	12F1	9.4 mi WSW of vent		
	13F2	6.5 mi W of vent		
	13F3	9.3 mi W of vent		
	14F2	6.6 mi WNW of vent		
	15F3	5.4 mi NW of vent		
	16F2	8.1 mi NNW of vent		
	16G1 (1)	14.8 mi NNW of vent		
	1G3 (1)	18.5 mi N of vent		
	10G1 (1)	11.6 mi SSW of vent		

(1) Control station.

Source: ER-OL, Table 6.1-3, through Amendment 3, April 30, 1984

Table 5.11 (Continued)

EXPOSURE PATHWAY	STATION CODE	LOCATION(3)	COLLECTION METHOD AND FREQUENCY	TYPE AND FREQUENCY OF ANALYSES
IV. WATER				
(a) Surface	11A1	Approximately 200m (650 ft) SW of vent	Two-gallon (4) sample to be collected monthly providing winter icing conditions allow sample collection	Gamma scan on each monthly Sample Tritium analyses are done monthly
	12C1(1)	2.5 mi WSW of vent		
	7E1	1 mi W of Mad Horse Creek; 4.5 mi SE of vent		
	1F2	7.1 mi N of vent		
	16F1	6.9 mi NNW of vent at the mouth of the C&D canal		
(b) Ground	4S1	On-site	Two-gallon grab sample is collected monthly	Gamma scan on samples composited over a calendar quarter
	5D1	3.5 mi E of vent		
	3E1(1)	4.0 mi NE of vent		
(c) Drinking	2F3 (raw)	Salem Water Co.; 8 mi NNE of vent	50 ml aliquot is taken daily and composited to a monthly sample of two gallons	Gross beta monthly Gamma scan - QC Tritium analyses are done monthly
	2F3 (treated)			

(1) Control station.
(3) 1.61 km = 1 mile.
(4) 2 gallons = 7.6 liters.

Source: ER-OL, Table 6.1-3, through Amendment 3, April 30, 1984

Table 5.11 (Continued)

EXPOSURE PATHWAY	STATION CODE	LOCATION(3)	COLLECTION METHOD AND FREQUENCY	TYPE AND FREQUENCY OF ANALYSES
V. AQUATIC				
(a) Benthos	7E1	1 mi W of Mad Horse Creek; 4.5 mi SE of vent	A benthos sample consisting of benthic organisms and associated sediment is taken semiannually	Gamma spectrometry of each sample semiannually; SR-90 semiannually on sediment
	12C1(1)	2.5 mi WSW of vent		
	11A1	Outfall area; 200m (650 ft) SW of vent		
VI. INGESTION				
(a) Milk	15F1	5.2 mi NW of vent	Four gallon grab sample of fresh milk is collected from each farm semimonthly. Collected weekly if calculated dose exceeds 15 mrem to child's thyroid	Gamma scan semimonthly; I-131 monthly, I-131 weekly if calculated dose exceeds 15 mrem to child's thyroid
	2F4	6.3 mi NNE of vent		
	5F2	6.5 mi E of vent		E470.5
	14F1	5.5 mi WNW of vent		
	3G1(1)	16.6 mi NE of vent		
(b) Fish	11A1	Outfall area; 200m (650 ft) SW of vent	Two key samples of fish are sealed in plastic bag or jar and frozen semiannually or when in season	Gamma scan of edible portion on collection
	12C1(1)	2.5 mi WSW of vent		

(1) Control station.
(3) 1.61 km = 1 mile.

Source: ER-OL, Table 6.1-3, through Amendment 3, April 30, 1984

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Table 5.11 (Continued)

EXPOSURE PATHWAY	STATION CODE	LOCATION	COLLECTION METHOD AND FREQUENCY	TYPE AND FREQUENCY OF ANALYSES
VI. INGESTION				
(c) Crab	11A1	Outfall area; 200m (650 ft) SW of vent	Two keys samples of crab are sealed in a plastic bag or jar and frozen semiannually or when in season	Gamma scan of edible portion on collection
	12C1(1)	West bank opposite Artificial Island 2.5 mi SWS of vent		
(d) Fruits or Vegetation	1G1(1)	10.2 mi N of vent	Samples are collected during the normal harvest season, sealed in plastic, and frozen if perishable. Sufficient sample is collected to yield 500 grams of dry weight	Radioiodine determination of green leafy vegetables on collection
	2E1	4.45 mi NNE of vent		
	2F1	3 mi NNE of vent		Gamma scan on collection

(1) Control station.

Source: ER-OL, Table 6.1-3, through Amendment 3, April 30, 1984

Table 5.11 (Continued)

EXPOSURE PATHWAY	STATION CODE	LOCATION[3] [4]	COLLECTION METHOD AND FREQUENCY	TYPE AND FREQUENCY OF ANALYSES
VI. INGESTION				
(e) Game		Station vicinity east side of estuary	Muskrats are skinned and frozen semiannually	Gamma scan on edible portion only on collection
		West side of estuary,[1] 3-5 mi from vent		
		Within (10 mi) of station	Beef portion of cow is sampled and frozen semiannually[5]	

(1) Control station.
(3) 1.61 km = 1 mile.
(4) Location given at time of collection.
(5) This sample is subject to availability of slaughtered cow.

Source: ER-OL, Table 6.1-3, through Amendment 3, April 30, 1984

Table 5.12 Activity of radionuclides in the Hope Creek reactor core at 3,458 MWt

Group/radionuclide	Radioactive inventory (millions of curies)	Half-life (days)
A. NOBLE GASES		
Krypton-85	0.6	3,950
Krypton-85m	30	0.183
Krypton-87	50	0.0528
Krypton-88	70	0.117
Xenon-133	200	5.28
Xenon-135	40	0.384
B. IODINES		
Iodine-131	90	8.05
Iodine-132	100	0.0958
Iodine-133	200	0.875
Iodine-134	200	0.0366
Iodine-135	200	0.280
C. ALKALI METALS		
Rubidium-86	0.03	18.7
Cesium-134	8	750
Cesium-136	3	13.0
Cesium-137	5	11,000
D. TELLURIUM-ANTIMONY		
Tellurium-127	6	0.391
Tellurium-127m	1	109
Tellurim-129	30	0.048
Tellurim-129m	6	34.0
Tellurium-131m	10	1.25
Tellurium-132	100	3.25
Antimony-127	7	3.88
Antimony-129	40	0.179
E. ALKALINE EARTHS		
Strontium-89	100	52.1
Strontium-90	4	11,030
Strontium-91	100	0.403
Barium-140	200	12.8
F. COBALT AND NOBLE METALS		
Cobalt-58	0.8	71.0
Cobalt-60	0.3	1,920
Molybdenum-99	200	2.8
Technetium-99m	200	0.25
Ruthenium-103	100	39.5
Ruthenium-105	80	0.185
Ruthenium-106	30	366
Rhodium-105	50	1.50

Table 5.12 (Continued)

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

Note: The above grouping of radionuclides corresponds to that in Table 5.14.

Table 5.13 Approximate doses during a 2-hour exposure at the exclusion area boundary* (estimated by the staff)

Accidents and faults	Duration of release	Whole-body dose (rems)	Thyroid dose (rems)
INFREQUENT ACCIDENTS			
Category 2			
Fuel-handling accident	<2 hours	<1	2
LIMITING FAULTS			
Category 3			
Main steamline break	<2 hours	<1	<1
Control rod drop	hours-days	<1	<1
Large-break LOCA	hours-days	<1	15

*901 m (2,955 ft) from the center of the reactor building.

Table 5.14 Summary of the atmospheric release specifications used in consequence analysis for Hope Creek

Release Category[b]	Probability per reactor-yr.	Release time (hr)	Release duration (hr)	Warning time for evacuation (hr)	Energy release (10^6 BTU/hr)	Release height (m)	Fraction of Core Inventory Released[a] Xe-Kr	I	Cs-Rb	Te-Sb	Ba-Sr	Ru[c]	La[d]
TCγ'	$2. \times 10^{-6}$	1.5	2.0	1.0	14.	10.	1.0	0.5	0.7	0.6	0.07	0.05	0.008
TWγ'	$3. \times 10^{-6}$	50.	2.0	40.	14.	10.	1.0	0.1	0.3	0.4	0.02	0.03	0.005
TQUVγ'	$1. \times 10^{-5}$	2.0	0.5	1.0	210.	10.	1.0	0.1	0.3	0.4	0.03	0.03	0.005
AEγ'													
S_1Eγ'													
S_2Eγ'													
TCγ	$8. \times 10^{-6}$	1.5	2.0	1.0	0.	25.	1.0	0.07	0.1	0.1	0.02	0.01	0.002
TWγ	$1. \times 10^{-5}$	50.	2.0	40.	0.	25.	1.0	0.003	0.1	0.08	0.01	0.007	0.001
TQUVγ	$3. \times 10^{-5}$	3.5	0.5	1.0	0.	25.	1.0	0.02	0.06	0.1	0.006	0.007	0.001
AEγ													
S_1Eγ													
S_2Eγ													
BWR 4	$4. \times 10^{-5}$	5.0	2.0	2.0	0.	25.	.6	0.002	0.005	0.004	6×10^{-4}	6×10^{-4}	1×10^{-4}

(a) Background on the isotope groups and release mechanisms is presented in Appendix VII, WASH 1400 (NUREG 75/014). These fractions have been rounded to one figure.

(b) See Appendix F for description of the accident sequences and sequence groups.

(c) Includes Ru, Rh, Co, Mo, Tc.

(d) Includes Y, La, Zr, Nb, Ce, Pr, Nd, Np, Pu, Am, Cm.

Table 5.15 Annual average wind-direction probabilities for the Hope Creek site based on data for the year 1981

Wind blowing toward the direction	Probability (fraction of of the year)
N	0.057
NNE	0.056
NE	0.068
ENE	0.058
E	0.092
ESE	0.106
SE	0.137
SSE	0.076
S	0.054
SSW	0.042
SW	0.051
WSW	0.026
W	0.029
WNW	0.025
NW	0.063
NNW	0.061
Total	1.00

Table 5.16 Emergency response assumptions for Hope Creek

Emergency response characteristic	Value used in CRAC analysis	Comments
Evacuation distance	10 mi	To convert miles to kilometers, multiply by 1.609.
Delay time	1 hr	
Effective evacuation speed	3.4 mi/hr	Same as 1.5 m/s.
Effective downwind distance moved	15 mi	An artificial parameter used only to represent a realistic path length over which radiation exposure to each evacuee is calculated in the CRAC code.
Relocation zone	All areas more than 10 mi from the plant	The area outside the 10-mi plume exposure pathway emergency planning zone.
Relocation time, after plume passage	12 hr	A separate calculation, with a relocation time of 24 hours, was also performed. See Appendix I.
Relocation dose criterion (7-day projected bone marrow dose)	200 rems	
Factors by which unshielded exposures are multiplied to correct for shielding		
Plume exposure during evacuation	1	See Footnote 1.
Groundshine exposure during evacuation	0.5	See Footnote 1.
Plume exposure, other times	0.75	See Footnote 2.
Groundshine exposure, other times	0.33	See Footnote 2.

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

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

Table 5.17 Summary of environmental impacts and probabilities

Probability of given consequence per reactor-year	Person exposed: >200 rems total bone marrow dose	Person exposed: >25 rems whole-body dose	Early fatalities*	Population exposure, whole body (millions person-rems)**: Within 50 mi	Population exposure: Total	Latent cancers: Within 50 mi	Latent cancers: Total	Cost of offsite mitigating actions, ($millions)***
10^{-4}	0	0	0	0.05	0.07	3	4	2
10^{-5}	0	30,000	0	8	30	700	2,000	900
5×10^{-6}	0	60,000	0	20	50	1,000	3,000	2,000
10^{-6}	1,000	300,000	0	50	100	4,000	10,000	6,000
10^{-7}	10,000	900,000	0	100	400	9,000	20,000	10,000
10^{-8}	30,000	2,000,000	300	200	600	20,000	50,000	30,000
Related figure	5.5	5.5	5.8	5.6	5.6	5.7	5.7	5.9

*Assuming supportive medical treatment of those most exposed.

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

***See Section 5.9.4.5(5) for a listing of costs included.

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

Table 5.18 Estimated values of societal risks from severe accidents, per reactor-year

Consequence type	Estimated risk within the 50-mi region	Estimated total risk
(1) Early fatalities with supportive medical treatment (persons)	9E-6*	9E-6
(2) Early fatalities with minimal medical treatment (persons)	3E-4	3E-4
(3) Early injuries (persons)	0.008	0.008
(4) Latent cancer fatalities (excluding thyroid) (persons)	0.03	0.06
(5) Latent thyroid cancer fatalities (persons)	0.007	0.01
(6) Total person-rems	400	1,000
(7a) Cost of offsite mitigation measures (1980 $)	30,000	40,000
(7b) Regional industrial impact costs (1980 $)	Not calculated	3(4)**
(7c) Plant costs (1980 $)	200,000	200,000
(8) Land area for long-term interdiction (m^2)***	7,000	7,000

*9E-6 = 9×10^{-6} = 0.000009.
**Excludes costs of crop and milk interdiction, which are included in (7a).
***About 2.6 million m^2 equals 1 mi^2.

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

Table 5.19 Regional economic impacts of output and employment

Release specification*	Wind direction	Direct (millions of 1980$) Nonagricultural	Direct (millions of 1980$) Agricultural	Indirect (millions of 1980$)	Total (millions of 1980$)	Loss in employment (annualized jobs)	Expected loss in output per reactor-year (1980$)
			Maximum losses				
1	NNE	11,082	260	1,396	12,744	594,000	1,435
2	NNE	10,911	213	1,368	12,492	546,242	2,110
3	NNE	10,911	213	1,368	12,492	546,242	7,033
4	N	774	186	118	1,078	50,064	494
5	N	774	186	118	1,078	50,064	618
6	N	425	94	64	583	26,943	1,002
7	NW	11	4	2	17	809	38
			Maximum losses				
All	SE	0	0	0	0	0	0
			Expected losses per reactor-year (1980$)				
1	All	3,716	366	502	4,584	<1	**
2	All	3,918	456	538	4,912	<1	
3	All	13,062	1,519	1,794	16,375	<1	
4	All	1,287	813	258	2,358	<1	
5	All	1,606	1,014	323	2,943	<1	
6	All	2,405	659	376	3,440	<1	
7	All	37	145	24	260	0	
All	All	26,031	4,972	4,826	34,818	1.4	

*Release specifications include:
1 - TCγ'
2 - TWγ'
3 - TQUVγ'
4 - TCγ'
5 - TWγ'
6 - TQUVγ'
7 - BWR4

**Not applicable, as the expected loss is already expressed in the "Total" column for this portion of the table.

Source: Bureau of Economic Analysis, U.S. Department of Commerce, with assumptions supplied by the U.S. Nuclear Regulatory Commission.

Table 5.20 (Summary Table S-3) Uranium-fuel-cycle environmental data[1]

[Normalized to model LWR annual fuel requirement [WASH-1248] or reference reactor year [NUREG-0116]]

Environmental considerations	Total	Maximum effect per annual fuel requirement or reference reactor year of model 1,000 MWe LWR
NATURAL RESOURCES USE		
Land (acres):		
Temporarily committed[2]	100	
Undisturbed area	79	
Disturbed area	22	Equivalent to a 110 MWe coal-fired power plant.
Permanently committed	13	
Overburden moved (millions of MT)	2.8	Equivalent to 95 MWe coal-fired power plant
Water (millions of gallons):		
Discharged to air	160	=2 percent of model 1,000 MWe LWR with cooling tower.
Discharged to water bodies	11,090	
Discharged to ground	127	
Total	11,377	<4 percent of model 1,000 MWe LWR with once-through cooling.
Fossil fuel:		
Electrical energy (thousands of MW-hour)	323	<5 percent of model 1,000 MWe LWR output
Equivalent coal (thousands of MT)	118	Equivalent to the consumption of a 45 MWe coal-fired power plant.
Natural gas (millions of scf)	135	<0.4 percent of model 1,000 MWe energy output.
EFFLUENTS—CHEMICAL (MT)		
Gases (including entrainment):[3]		
SO_x	4,400	
NO_x[4]	1,190	Equivalent to emissions from 45 MWe coal-fired plant for a year.
Hydrocarbons	14	
CO	29.6	
Particulates	1,154	
Other gases:		
F	.67	Principally from UF_6 production, enrichment, and reprocessing. Concentration within range of state standards—below level that has effects on human health.
HCl	.014	
Liquids:		
SO_4^-	9.9	From enrichment, fuel fabrication, and reprocessing steps. Components that constitute a potential for adverse environmental effect are present in dilute concentrations and receive additional dilution by receiving bodies of water to levels below permissible standards. The constituents that require dilution and the flow of dilution water are:
NO_3^-	25.8	
Fluoride	12.9	
Ca^{++}	5.4	
Cl^-	8.5	
Na^+	12.1	
NH_3	10.0	
Fe	.4	NH_3—600 cfs.
		NO_3—20 cfs.
		Fluoride—70 cfs.
Tailings solutions (thousands of MT)	240	From mills only—no significant effluents to environment.
Solids	91,000	Principally from mills—no significant effluents to environment.

Table 5.20 (Continued)

[Normalized to model LWR annual fuel requirement (WASH-1248) or reference reactor year (NUREG-0116)]

Environmental considerations	Total	Maximum effect per annual fuel requirement or reference reactor year of model 1,000 MWe LWR
EFFLUENTS—RADIOLOGICAL (CURIES)		
Gases (including entrainment):		
Rn-222		Presently under reconsideration by the Commission.
Ra-226	.02	
Th-230	.02	
Uranium	.034	
Tritium (thousands)	18.1	
C-14	24	
Kr-85 (thousands)	400	Principally from fuel reprocessing plants.
Ru-106	.14	
I-129	1.3	
I-131	.83	
Tc-99		Presently under consideration by the Commission
Fission products and transuranics	.203	
Liquids:		
Uranium and daughters	2.1	Principally from milling—included tailings liquor and returned to ground—no effluents; therefore, no effect on environment.
Ra-226	.0034	From UF_6 production
Th-230	.0015	
Th-234	.01	From fuel fabrication plants—concentration 10 percent of 10 CFR 20 for total processing 26 annual fuel requirements for model LWR
Fission and activation products	5.9×10^{-6}	
Solids (buried on site):		
Other than high level (shallow)	11,300	9,100 Ci comes from low level reactor wastes and 1,500 Ci comes from reactor decontamination and decommissioning—buried at land burial facilities. 600 Ci comes from mills—included in tailings returned to ground. Approximately 60 Ci comes from conversion and spent fuel storage. No significant effluent to the environment.
TRU and HLW (deep)	1.1×10^{7}	Buried at Federal Repository
Effluents—thermal (billions of British thermal units)	4,063	<5 percent of model 1,000 MWe LWR
Transportation (person-rem):		
Exposure of workers and general public	2.5	
Occupational exposure (person-rem)	22.6	From reprocessing and waste management

[1] In some cases where no entry appears it is clear from the background documents that the matter was addressed and that, in effect, the Table should be read as if a specific zero entry had been made. However, there are other areas that are not addressed at all in the Table. Table S-3 does not include health effects from the effluents described in the Table, or estimates of releases of Radon-222 from the uranium fuel cycle or estimates of Technetium-99 released from waste management or reprocessing activities. These issues may be the subject of litigation in the individual licensing proceedings.

Data supporting this table are given in the "Environmental Survey of the Uranium Fuel Cycle," WASH-1248, April 1974; the "Environmental Survey of the Reprocessing and Waste Management Portion of the LWR Fuel Cycle," NUREG-0116 (Supp. 1 to WASH-1248); the "Public Comments and Task Force Responses Regarding the Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," NUREG-0216 (Supp. 2 to WASH-1248); and in the record of the final rulemaking pertaining to Uranium Fuel Cycle Impacts from Spent Fuel Reprocessing and Radioactive Waste Management, Docket RM-50-3. The contributions from reprocessing, waste management and transportation of wastes are maximized for either of the two fuel cycles (uranium only and no recycle). The contribution from transportation excludes transportation of cold fuel to a reactor and of irradiated fuel and radioactive wastes from a reactor which are considered in Table S-4 of § 51.20(g). The contributions from the other steps of the fuel cycle are given in columns A-E of Table S-3A of WASH-1248.

[2] The contributions to temporarily committed land from reprocessing are not prorated over 30 years, since the complete temporary impact accrues regardless of whether the plant services one reactor for one year or 57 reactors for 30 years.

[3] Estimated effluents based upon combustion of equivalent coal for power generation.

[4] 1.2 percent from natural gas use and process.

6 EVALUATION OF THE PROPOSED ACTION

6.1 Unavoidable Adverse Impacts

The staff has reassessed the physical, social, biological, and economic impacts that can be attributed to the operation of the Hope Creek Generating Station. These impacts are summarized in Table 6.1.

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

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

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

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

6.2 Irreversible and Irretrievable Commitment of Resources

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

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

There have been no significant changes in the staff's evaluation for the Hope Creek Generating Station since the construction permit stage environmental review.

6.4 Benefit-Cost Summary

6.4.1 Summary

Sections below describe the economic, environmental, and socieconomic benefits and costs associated with the operation of Hope Creek.

6.4.2 Benefits

A major benefit to be derived from the operation of Hope Creek is the approximately 5.1 billion kWh of baseload electrical energy that will be produced annually (this projection assumes that the unit will operate at an annual average capacity factor of 55%). The addition of the unit will also improve the applicant's ability to supply system load requirements by contributing 1,067 MW of capacity to the Public Service Electric & Gas (PSE&G) and Atlantic City Electric systems (1,014 MW to the PSE&G system and 53 MW to the Atlantic City Electric system).

Another benefit is the overall savings in system production costs that would result from operation of Hope Creek. If it is assumed that the energy available from the unit replaces energy from installed fossil units on the Pennsylvania-New Jersey-Maryland interconnection system, these avoided costs will total approximately $63 million (1987 dollars) per year during the life of the plant.

6.4.3 Economic Costs

The economic costs associated with station operation include fuel costs and operation and maintenance costs, which are expected to average approximately 13.6 mills per kWh and 13.9 mills per kWh, respectively (1987 dollars).

Estimates of decommissioning costs for a boiling-water reactor such as Hope Creek range from $43.6 million to $58.9 million (1978 dollars (NUREG-0586)). Assuming an escalation rate of 10% per year, these costs would range from $103 million to $139 million in 1987 dollars.

6.5 Conclusion

As a result of its analysis and review of potential environmental, technical, and social impacts, the staff concludes that the Hope Creek Generating Station can be operated with minimal environmental impact.

6.6 Reference

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

Table 6.1 Benefit-cost summary for Hope Creek

Primary impact and effect on population or resources	Quantity (section)*	Impact**
BENEFITS		
Direct		
Electrical energy	5.1 billion kWh/year	Large
Additional generating capacity	1,067 MWe	Large
Operating cost avoided	$63 million/year***	Moderate
COSTS		
Economic		
Fuel	13.6 mills/kWh***	Small
Operation and maintenance	13.9 mills/kWh***	Moderate
Decommissioning	$103-139 million***	Small
Environmental		
Damages suffered by other water users		
Surface water consumption	(Sec. 5.3.1)	Small
Surface water contamination	(Sec. 5.3.2)	Small
Ground water consumption	(Sec. 5.3.1.2)	Small
Gound water contamination	(Sec. 5.3.2)	None
Damage to aquatic resources		
Impingement and entrainment	(Sec. 5.5.2)	Small
Thermal effects	(Sec. 5.3.2 and 5.2.2)	Small
Chemical discharges	(Sec. 5.3.2)	Small
Damage to terrestrial resources		
Cooling tower operation	(Sec. 5.5.1.2)	Small
Transmission line maintenance	(Sec. 5.5.1.3)	Small
Adverse socioeconomic impacts		
Loss of historic or archeological resources	(Sec. 5.7)	None
Increased demand on public facilities and services	(Sec. 5.8)	Small
Increased demands on private facilities and services	(Sec. 5.8)	Small
Noise	(Sec. 5.12)	None
Adverse nonradiological health effects		
Water quality changes	(Sec. 5.3.2.1)	None
Air quality changes	(Sec. 5.4.1 and 5.4.2)	None

*See footnotes at end of table.

Table 6.1 (Continued)

Primary impact and effect on population or resources	Quantity (Section)*	Impact**
Adverse radiological health effects		
Routine operation	(Sec. 5.9.3)	Small
Postulated accidents	(Sec. 5.9.4)	****
Uranium fuel cycle	(Sec. 5.10)	Small

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

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

***1987 dollars.

****Impacts of an accident could possibly be large while the risk of an accident is small.

7 LIST OF CONTRIBUTORS

The following personnel of the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, participated in the preparation of this document:

Louis Bykoski	Regional Environmental Economist; Ph.D. (Economics), 1965; 19 years' experience.
Patrick Easley	Nuclear Engineer; M.S. (Chemical Engineering), 1980; 7 years' experience.
James Fairobent	Meteorologist; M.S. (Meteorology), 1972; 11 years' experience.
Charles Ferrell	Site Analyst; B.S. (Physics), 1950; 27 years' experience.
Clarence Hickey, Jr.	Senior Fishery Biologist; M.S. (Marine Science), 1971; 13 years' experience; AFS Certified Fisheries Scientists.
Robert Jachowski	Hydrologist; B.S. (Civil Engineering); 36 years' experience.
Germain LaRoche	Senior Land Use Analyst; Ph.D. (Botany-Ecology), 1969; 26 years' experience.
John Lehr	Senior Environmental Engineer; M.S. (Environmental Engineering), 1972; 12 years' experience.
Maurice Messier	Senior Antitrust Economist; M.S. (Economics), 1965; 19 years' experience.
Tin Mo	Health Physicist; Ph.D. (Nuclear and Radiochemistry), 1971; 12 years' experience.
Jerry Swift	Health Physicist; Ph.D. (Nuclear Engineering), 1971; 18 years' experience.
David Wagner	Project Manager; Bachelor of Nuclear Engineering, 1982; 2 years' experience.

The following contractor personnel participated in the preparation of this document:

S. Marshall Adams	Oak Ridge National Laboratory, Physiological Ecologist; Ph.D. (Marine Biology), 1974; 10 years' experience.
Roger Kroodsma	Oak Ridge National Laboratory, Terrestrial Ecologist; Ph.D. (Zoology), 1970; 14 years' experience.

8 LIST OF AGENCIES, ORGANIZATIONS, AND PERSONS TO WHOM COPIES OF THE DRAFT ENVIRONMENTAL STATEMENT WERE SENT

Advisory Council on Historic Preservation
U.S. Soil Conservation Service
U.S. Department of the Army, Corps of Engineers
U.S. Department of Commerce
U.S. Department of Health and Human Services
U.S. Department of Housing and Urban Development
U.S. Department of the Interior
U.S. Department of Transportation
U.S. Environmental Protection Agency
Federal Energy Regulatory Commission
Attorney General, State of New Jersey
New Jersey Board of Public Utilities
New Jersey Department of Environmental Protection
Attorney General, State of Delaware
Attorney General, State of Pennsylvania
Mayor, Lower Alloways Creek Township, New Jersey
Brookhaven National Laboratory
Wilmington Metropolitan Area Planning Coordinating Council
New Jersey Department of Community Affairs
New Jersey Department of Labor
Salem County Planning Board

9 RESERVED FOR STAFF RESPONSES TO COMMENTS ON THE DRAFT ENVIRONMENTAL STATEMENT

Responses to comments on the Draft Environmental Statement will be included in the Final Environmental Statement to be issued in November 1984.

APPENDIX A

RESERVED FOR COMMENTS ON THE DRAFT ENVIRONMENTAL STATEMENT

APPENDIX B

NEPA POPULATION-DOSE ASSESSMENT

APPENDIX B

NEPA POPULATION-DOSE ASSESSMENT

Population-dose commitments are calculated for all individuals living within 80 km (50 mi) of the Hope Creek facility, employing the same dose calculation models used for individual doses (RG 1.109, Revision 1), for the purpose of meeting the "as low as reasonably achievable" (ALARA) requirements of 10 CFR 50, Appendix I. In addition, dose commitments to the population residing beyond the 80-km region, associated with the export of food crops produced within the 80-km region and with the atmospheric and hydrospheric transport of the more mobile effluent species, such as noble gases, tritium, and carbon-14, are taken into consideration for the purpose of meeting the requirements of the National Environmental Policy Act, 1969 (NEPA). This appendix describes the methods used to make these NEPA population dose estimates.

1. Iodines and Particulates Released to the Atmosphere

Effluent nuclides in this category deposit onto the ground as the effluent moves downwind; thus the concentration of these nuclides remaining in the plume is continuously being reduced. Within 80 km of the facility, the deposition model in RG 1.111, Revision 1, is used in conjunction with the dose models in RG 1.109, Revision 1. Site-specific data concerning production and consumption of foods within 80 km of the reactor are used. For estimates of population doses beyond 80 km, it is assumed that excess food not consumed within the 80-km area would be consumed by the population beyond 80 km. It is further assumed that none, or very few, of the particulates released from the facility will be transported beyond the 80-km distance; thus, they will make no significant contribution to the population dose outside the 80-km region, except by export of food crops.

2. Noble Gases, Carbon-14, and Tritium Released to the Atmosphere

For locations within 80 km of the reactor facility, exposures to these effluents are calculated with a constant mean wind-direction model according to the guidance provided in RG 1.111, Revision 1, and the dose models described in RG 1.109, Revision 1. For estimating the dose commitment from these radionuclides to the U.S. population residing beyond the 80-km region, two dispersion regimes are considered. These are referred to as the first-pass-dispersion regime and the world-wide-dispersion regime. The model for the first-pass-dispersion regime estimates the dose commitment to the population from the radioactive plume as it leaves the facility and drifts across the continental United States toward the northeastern corner of the United States. The model for the world-wide-dispersion regime estimates the dose commitment to the U.S. population after the released radionuclides mix uniformly in the world's atmosphere or oceans.

(a) First-Pass Dispersion

For estimating the dose commitment to the U.S. population residing beyond the 80-km region as a result of the first pass of radioactive pollutants, it is assumed that the pollutants disperse in the lateral and vertical directions along the plume path. The direction of movement of the plume is assumed to be from the facility toward the northeast corner of the United States. The extent of vertical dispersion is assumed to be limited by the ground plane and the stable atmospheric layer aloft, the height of which determines the mixing depth. The shape of such a plume geometry can be visualized as a right cylindrical wedge whose height is equal to the mixing depth. Under the assumption of constant population density, the population dose associated with such a plume geometry is independent of the extent of lateral dispersion, and is only dependent upon the mixing depth and other nongeometrical related factors (NUREG-0597). The mixing depth is estimated to be 1,000 m (0.6 mi), and a uniform population density of 62 persons/km^2 is assumed along the plume path, with an average plume-transport velocity of 2 m/s (7 ft/s).

The total-body population-dose commitment from the first pass of radioactive effluents is due principally to external exposure from gamma-emitting noble gases, and to internal exposure from inhalation of air containing tritium and from ingestion of food containing carbon-14 and tritium.

(b) <u>World-Wide Dispersion</u>

For estimating the dose commitment to the U.S. population after the first-pass, world-wide dispersion is assumed. Nondepositing radionuclides with half-lives greater than 1 year are considered. Noble gases and carbon-14 are assumed to mix uniformly in the world's atmosphere (3.8×10^{18} m^3), and radioactive decay is taken into consideration. The world-wide-dispersion model estimates the activity of each nuclide at the end of a 20-year release period (midpoint of reactor life) and estimates the annual population-dose commitment at that time, taking into consideration radioactive decay and physical removal mechanisms (for example, carbon-14 is gradually removed to the world's oceans). The total-body population-dose commitment from the noble gases is due mainly to external exposure from gamma-emitting nuclides, whereas from carbon-14 it is due mainly to internal exposure from ingestion of food containing carbon-14.

The population-dose commitment as a result of tritium releases is estimated in a manner similar to that for carbon-14, except that after the first pass, all the tritium is assumed to be immediately distributed in the world's circulating water volume (2.7×10^{16} m^3) including the top 75 m (246 ft) of the seas and oceans, as well as the rivers and atmospheric moisture. The concentration of tritium in the world's circulating water is estimated at the time after 20 years of releases have occurred, taking into consideration radioactive decay; the population-dose commitment estimates are based on the incremental concentration at that time. The total-body population-dose commitment from tritium is due mainly to internal exposure from the consumption of food.

3. <u>Liquid Effluents</u>

Population-dose commitments due to effluents in the receiving water within 80 km of the facility are calculated as described in RG 1.109, Revision 1. It is assumed that no depletion by sedimentation of the nuclides present in the receiving water occurs within 80 km. It also is assumed that aquatic biota concentrate radioactivity in the same manner as was assumed for the ALARA evaluation for the maximally exposed individual. However, food-consumption values appropriate for the average, rather than the maximum, individual are used. It is further assumed that all the sport and commercial fish and shellfish caught within the 80-km area are eaten by the U.S. population.

Beyond 80 km, it is assumed that all the liquid-effluent nuclides except tritium have deposited on the sediments so that they make no further contribution to population exposures. The tritium is assumed to mix uniformly in the world's circulating water volume and to result in an exposure to the U.S. population in the same manner as discussed for tritium in gaseous effluents.

4. References

U.S. Nuclear Regulatory Commission, NUREG-0597, K. F. Eckerman, et al., "User's Guide to GASPAR Code," June 1980.

APPENDIX C

IMPACTS OF THE URANIUM FUEL CYCLE

APPENDIX C

IMPACTS OF THE URANIUM FUEL CYCLE

The following assessment of the environmental impacts of the LWR-supporting fuel cycle as related to the operation of the proposed project is based on the values given in Table S-3 of Title 10 of the Code of Federal Regulations, Part 50 (10 CFR 50) (see Section 5.10 of the main body of this report) and the NRC staff's estimates of radon-222 and technetium-99 releases. For the sake of consistency, the analysis of fuel-cycle impacts has been cast in terms of a model 1000-MWe light-water-cooled reactor (LWR) operating at an annual capacity factor of 80%. In the following review and evaluation of the environmental impacts of the fuel cycle, the staff's analysis and conclusions would not be altered if the analysis were to be based on the net electrical power output of the Hope Creek Generating Station.

1. Land Use

The total annual land requirement for the fuel cycle supporting a model 1,000-MWe LWR is about 460,000 m^2 (113 acres). Approximately 53,000 m^2 (13 acres) per year are permanently committed land, and 405,000 m^2 (100 acres) per year are temporarily committed. (A "temporary" land commitment is a commitment for the life of the specific fuel-cycle plant, such as a mill, enrichment plant, or succeeding plants. On abandonment or decommissioning, such land can be used for any purpose. "Permanent" commitments represent land that may not be released for use after plant shutdown and/or decommissioning.) Of the 405,000 m^2 per year of temporarily committed land, 320,000 m^2 are undisturbed and 90,000 m^2 are disturbed. Considering common classes of land use in the United States,* fuel-cycle land-use requirements to support the model 1000-MWe LWR do not represent a significant impact.

2. Water Use

The principal water-use requirement for the fuel cycle supporting a model 1000-MWe LWR is that required to remove waste heat from the power stations supplying electrical energy to the enrichment step of this cycle. Of the total annual requirement of 43×10^6 m^3 (11.4×10^9 gal), about 42×10^6 m^3 are required for this purpose, assuming that these plants use once-through cooling. Other water uses involve the discharge to air (for example, evaporation losses in process cooling) of about 0.6×10^6 m^3 (16×10^7 gal) per year and water discharged to the ground (for example, mine drainage) of about 0.5×10^6 m^3 per year.

On a thermal effluent basis, annual discharges from the nuclear fuel cycle are about 4% of those from the model 1000-MWe LWR using once-through cooling. The

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*A coal-fired plant of 1000-MWe capacity using strip-mined coal requires the disturbance of about 810,000 m^2 (200 acres) per year for fuel alone.

consumptive water use of 0.6×10^6 m^3 per year is about 2% of that from the model 1000-MWe LWR using cooling towers. The maximum consumptive water use (assuming that all plants supplying electrical energy to the nuclear fuel cycle used cooling towers) would be about 6% of the model 1000-MWe LWR using cooling towers. Under this condition, thermal effluents would be negligible. The staff finds that these combinations of thermal loadings and water consumption are acceptable relative to the water use and thermal discharges of the proposed project.

3. Fossil Fuel Consumption

Electrical energy and process heat are required during various phases of the fuel-cycle process. The electrical energy is usually produced by the combustion of fossil fuel at conventional power plants. Electrical energy associated with the fuel cycle represents about 5% of the annual electrical power production of the model 1000-MWe LWR. Process heat is primarily generated by the combustion of natural gas. This gas consumption, if used to generate electricity, would be less than 0.3% of the electrical output from the model plant. The staff finds that the direct and indirect consumptions of electrical energy for fuel-cycle operations are small and acceptable relative to the net power production of the proposed project.

4. Chemical Effluents

The quantities of chemical, gaseous, and particulate effluents associated with fuel-cycle processes are given in Table S-3. The principal species are sulfur oxides, nitrogen oxides, and particulates. On the basis of data in a Council on Environmental Quality report (CEQ, 1976), the staff finds that these emissions constitute an extremely small additional atmospheric loading in comparison with the same emissions from the stationary fuel-combustion and transportation sectors in the U.S.; that is, about 0.02% of the annual national releases for each of these species. The staff believes that such small increases in releases of these pollutants are acceptable.

Liquid chemical effluents produced in fuel cycle processes are related to fuel-enrichment, -fabrication, and -reprocessing operations and may be released to receiving waters. These effluents are usually present in dilute concentrations such that only small amounts of dilution water are required to reach levels of concentration that are within established standards. The flow of dilution water required for specific constituents is specified in Table S-3. Additionally, all liquid discharges into the navigable waters of the U.S. from plants associated with the fuel-cycle operations will be subject to requirements and limitations set forth in the NPDES permit.

Tailings solutions and solids are generated during the milling process. These solutions and solids are not released in quantities sufficient to have a significant impact on the environment.

5. Radioactive Effluents

Radioactive effluents estimated to be released to the environment from reprocessing and waste-management activities and certain other phases of the fuel-cycle process are set forth in Table S-3. Using these data, the staff has calculated for 1 year of operation of the model 1000-MWe LWR, the 100-year environmental dose commitment* to the U.S. population from the LWR-supporting fuel cycle. Dose commitments are provided in this section for exposure to four categories of radioactive releases: (1) airborne effluents that are quantified in Table S-3 (that is, all radionuclides except radon-222 and technetium-99), (2) liquid effluents that are quantified in Table S-3 (that is, all radionuclides except technetium-99); (3) the staff's estimates of radon-222 releases; and (4) the staff's estimate of technetium-99 releases. Dose commitments from the first two categories are also described in a proposed explanatory narrative for Table S-3, which was published in the Federal Register on March 4, 1981 (46 FR 15154-15175).

Airborne Effluents

Population dose estimates for exposure to airborne effluents are based on the annual releases listed in Table S-3, using an environmental dose commitment (EDC) time of 100 years.* The computational code used for these estimates is the RABGAD code originally developed for use in the "Generic Environmental Impact Statement on the Use of Mixed Oxide Fuel in Light-Water-Cooled Nuclear Power Plants," GESMO (NUREG-0002, Chapter IV, Section J, Appendix A). Two generic sites are postulated for the points of release of the airborne effluents: (1) a site in the midwestern United States for releases from a fuel reprocessing plant and other facilities, and (2) a site in the western United States for releases from milling and a geological repository.

The following environmental pathways were considered in estimating doses: (1) inhalation and submersion in the plume during its initial passage; (2) ingestion of food; (3) external exposure from radionuclides deposited on soil; and (4) atmospheric resuspension of radionuclides deposited on soil. Radionuclides released to the atmosphere from the midwestern site are assumed to be transported with a mean wind speed of 2 m/sec over a 2413-km (1500-mile)** pathway from the midwestern United States to the northeast corner of the United States, and deposited on vegetation (deposition velocity of 1.0 cm/sec) with subsequent uptake by milk- and meat-producing animals. No removal mechanisms are assumed during the first 100 years, except normal weathering from crops to soil (weathering half-life of 13 days). Doses from exposure to carbon-14 were estimated using the GESMO model to estimate the dose to U.S. population from the initial passage of carbon-14 before it mixed in the world's carbon pool. The model developed by Killough (1977) was used to estimate doses from exposure to carbon-14 after it mixed in the world's carbon pool.

In a similar manner, radionuclides released from the western site were assumed to be transported over a 3218-km (2000-mile) pathway to the northeast corner of the United States. The agricultural characteristics that were used in computing doses from exposure to airborne effluents from the two generic sites are described in GESMO (NUREG-0002, page IV J(A)-19). To allow for an increase in population, the population densities used in this analysis were 50% greater than the values used in GESMO (NUREG-0002, page IV J(A)-19).

*The 100-year environmental dose commitment is the integrated population dose for 100 years; that is, it represents the sum of the annual population doses for a total of 100 years.

**Here and elsewhere in this narrative, insignificant digits are retained for purposes of internal consistency in the model.

Liquid Effluents

Population dose estimates for exposure to liquid effluents are based on the annual releases listed in Table S-3 and the hydrological model described in GESMO (NUREG-0002, pages IV J(A)-20, -21, and -22). The following environmental pathways were considered in estimating doses: (1) ingestion of water and fish; (2) ingestion of food (vegetation, milk, and beef) that had been produced through irrigation; and (3) exposure from shoreline, swimming, and boating activities.

It is estimated from these calculations that the overall total-body dose commitment to the U.S. population from exposure to gaseous releases from the fuel cycle (excluding reactor releases and the dose commitment due to radon-222 and technetium-99) would be approximately 450 person-rems to the total body for each year of operation of the model 1000-MWe LWR (reference reactor year, or RRY). Based on Table S-3 values, the additional total-body dose commitments to the U.S. population from radioactive liquid effluents (excluding technetium-99) as a result of all fuel-cycle operations other than reactor operation would be about 100 person-rems per year of operation. Thus, the estimated 100-year environmental dose commitment to the U.S. population from radioactive gaseous and liquid releases due to these portions of the fuel cycle is about 550 person-rems to the total body (whole body) per RRY.

Because there are higher dose commitments to certain organs (for example, lung, bone, and thyroid) than to the total body, the total risk of radiogenic cancer is not addressed by the total body dose commitment alone. Using risk estimators of 135, 6.9, 22, and 13.4 cancer deaths per million person-rems for total-body, bone, lung, and thyroid exposures, respectively, it is possible to estimate the total body risk equivalent dose for certain organs (NUREG-0002, Chapter IV, Section J, Appendix B). The sum of the total body risk equivalent dose from those organs was estimated to be about 100 person-rems. When added to the above value, the total 100-year environmental dose commitment would be about 650 person-rems (total body risk equivalent dose) per RRY (Section 5.9.3.1.1 describes the health effects models in more detail).

Radon-222

At this time the quantitites of radon-222 and technetium-99 releases are not listed in Table S-3. Principal radon releases occur during mining and milling operations and as emissions from mill tailings, whereas principal technetium-99 releases occur from gaseous diffusion enrichment facilities. The staff has determined that radon-222 releases per RRY from these operations are as given in Table C-1. The staff has calculated population-dose commitments for these sources of radon-222 using the RABGAD computer code described in Volume 3 of NUREG-0002 (Appendix A, Chapter IV, Section J). The results of these calculations for mining and milling activities prior to tailings stabilization are listed in Table C-2.

The staff has considered the health effects associated with the releases of radon-222, including both the short-term effects of mining and milling and active tailings, and the potential long-term effects from unreclaimed open-pit mines and stabilized tailings. The staff has assumed that after completion of active mining, underground mines will be sealed, returning releases of radon-222 to background levels. For purposes of providing an upper bound impact assessment, the staff has assumed that open-pit mines will be unreclaimed and has calculated that if all ore were produced from open-pit mines, releases from them would be 110 Ci per RRY. However, because the distribution of uranium-ore reserves available by conventional mining methods is 66% underground and 34% open pit (Department of Energy, 1978), the staff has further assumed that uranium to fuel LWRs will be produced by conventional mining methods in these proportions. This means that long-term releases from unreclaimed open-pit mines will be 0.34 x 110 or 37 Ci per year per RRY.

Based on a value of 37 Ci per year per RRY for long-term releases from unreclaimed open-pit mines, the radon released from unreclaimed open-pit mines over 100- and 1000-year periods would be about 3700 Ci and 37,000 Ci per RRY, respectively. The environmental dose commitments for a 100- to 1000-year period would be as shown in Table C-3.

These commitments represent a worst case situation in that no mitigating circumstances are assumed. However, state and Federal laws currently require reclamation of strip and open-pit coal mines, and it is very probable that similar reclamation will be required for open-pit uranium mines. If so, long-term releases from such mines should approach background levels.

For long-term radon releases from stabilized tailings piles, the staff has assumed that these tailings would emit, per RRY, 1 Ci per year for 100 years, 10 Ci per year for the next 400 years, and 100 Ci per year for periods beyond 500 years. With these assumptions, the cumulative radon-222 release from stabilized-tailings piles per RRY would be 100 Ci in 100 years, 4090 Ci in 500 years, and 53,800 Ci in 1000 years (Gotchy, 1978). The total-body, bone, and bronchial epithelium dose commitments for these periods are as shown in Table C-4.

Using risk estimators of 135, 6.9, and 22 cancer deaths per million person-rems for total-body, bone, and lung exposures, respectively, the estimated risk of cancer mortality resulting from mining, milling, and active-tailings emissions of radon-222 (that is, Table C-2) is about 0.11 cancer fatality per RRY. When the risks from radon-222 emissions from stabilized tailings and from reclaimed and unreclaimed open-pit mines are added to the value of 0.11 cancer fatality, the overall risks of radon-induced cancer fatalities per RRY are as follows:

0.19 fatality for a 100-year period
2.0 fatalities for a 1000-year period

These doses and predicted health effects have been compared with those that can be expected from natural-background emissions of radon-222. Using data from the National Council on Radiation Protection (NCRP, 1975), the staff calculates the average radon-222 concentration in air in the contiguous United States to be about 150 pCi/m^3, which the NCRP estimates will result in an annual dose to the bronchial epithelium of 450 millirems. For a stabilized future U.S. population of 300 million, this represents a total lung-dose commitment of 135 million person-rems per year. Using the same risk estimator of 22 lung-cancer fatalities per million person-lung-rems used to predict cancer fatalities for the model 1000-MWe LWR, the staff estimates that lung-cancer fatalities alone from background radon-222 in the air can be calculated to be about 3000 per year, or

300,000 to 3,000,000 lung-cancer deaths over periods of 100 to 1000 years, respectively.

Current NRC regulations (10 CFR 40, Appendix A) require that an earth cover not less than 3 meters (10 ft) in depth be placed over tailings to reduce the Rn-222 emanation from the disposed tailings to less than 2 pCi/m^2-sec, on a calculated basis above background. In October 1983, the U.S. Environmental Protection Agency (EPA) published environmental standards for the disposal of uranium and thorium mill tailings at licensed commercial processing sites (EPA 1983). The EPA regulations (40 CFR 192) require that disposal be designed to limit Rn-222 emanation to less than 20 pCi/m^2-sec, averaged over the surface of the disposed tailings. The NRC Office of Nuclear Material Safety and Safeguards is reviewing its regulations for tailings disposal to ensure that they conform with the EPA regulations. Although a few of the dose estimates in this appendix would change if NRC adopts EPA's higher Rn-222 flux limit for disposal of tailings, the basic conclusion of this appendix should still be valid. That conclusion is: "The staff concludes that both the dose commitments and health effects of the LWR-supporting uranium fuel cycle are very small when compared with dose commitments and potential health effects to the U.S. population resulting from all natural-background sources."

Technetium-99

The staff has calculated the potential 100-year environmental dose commitment to the U.S. population from the release of technetium-99. These calculations are based on the gaseous and the hydrological pathway model systems described in Volume 3 of NUREG-0002 (Chapter IV, Section J, Appendix A) and are described in more detail in the staff's testimony at the operating license hearing for the Susquehanna Station (Branagan and Struckmeyer, 1981). The gastrointestinal tract and the kidney are the body organs that receive the highest doses from exposure to technetium-99. The total body dose is estimated at less than 1 person-rem per RRY and the total body risk equivalent dose is estimated at less than 10 person-rems per RRY.

Summary of Impacts

The potential radiological impacts of the supporting fuel cycle are summarized in Table C-5 for an environmental dose commitment time of 100 years. For an environmental dose commitment time of 100 years, the total body dose to the U.S. population is about 790 person-rems per RRY, and the corresponding total body risk equivalent dose is about 2000 person-rems per RRY. In a similar manner, the total body dose to the U.S. population is about 3000 person-rems per RRY, and the corresponding total body risk equivalent dose is about 15,000 person-rems per RRY using a 1000-year environmental dose commitment time.

Multiplying the total body risk equivalent dose of 2000 person-rems per RRY by the preceding risk estimator of 135 potential cancer deaths per million person-rems, the staff estimates that about 0.27 cancer death per RRY may occur in the U.S. population as a result of exposure to effluents from the fuel cycle. Multiplying the total body dose of 790 person-rems per RRY by the genetic risk estimator of 258 potential cases of all forms of genetic disorders per million person-rems, the staff estimates that about 0.20 potential genetic disorder per RRY may occur in all future generations of the population exposed during the 100-year environmental dose commitment time. In a similar manner, the staff estimates that about 2 potential cancer deaths per RRY and about 0.8 potential genetic disorder per RRY may occur using a 1000-year environmental dose commitment time.

Some perspective can be gained by comparing the preceding estimates with those from naturally occurring terrestrial and cosmic-ray sources. These average about 100 millirems. Therefore, for a stable future population of 300 million persons, the whole-body dose commitment would be about 30 million person-rems per year, or 3 billion person-rems and 30 billion person-rems for periods of 100 and 1000 years, respectively. These natural-background dose commitments could produce about 400,000 and 4,000,000 cancer deaths and about 770,000 and 7,700,000 genetic disorders, during the same time periods. From the above analysis, the staff concludes that both the dose commitments and health effects of the LWR-supporting uranium fuel cycle are very small when compared with dose commitments and potential health effects to the U.S. population resulting from all natural-background sources.

6. Radioactive Wastes

The quantities of buried radioactive waste material (low-level, high-level, and transuranic wastes) associated with the uranium fuel cycle are specified in Table S-3. For low-level waste disposal at land-burial facilities, the Commission notes in Table S-3 that there will be no significant radioactive releases to the environment. The Commission notes that high-level and transuranic wastes are to be buried at a Federal repository and that no release to the environment is associated with such disposal. NUREG-0116, which provides background and context for the high-level and transuranic waste values in Table S-3 established by the Commission, indicates that these high-level and transuranic wastes will be buried and will not be released to the biosphere. No radiological environmental impact is anticipated from such disposal.

7. Occupational Dose

The annual occupational dose attributable to all phases of the fuel cycle for the model 1000-MWe LWR is about 200 person-rems. The staff concludes that this occupational dose will have a small environmental impact.

8. Transportation

The transportation dose to workers and the public is specified in Table S-3. This dose is small in comparison with the natural-background dose.

9. Fuel Cycle

The staff's analysis of the uranium fuel cycle did not depend on the selected fuel cycle (no recycle or uranium-only recycle), because the data provided in Table S-3 include maximum recycle-option impact for each element of the fuel cycle. Thus the staff's conclusions as to acceptability of the environmental impacts of the fuel cycle are not affected by the specific fuel cycle selected.

10. References

Branagan, E., and R. Struckmeyer, testimony from "In the Matter of Pennsylvania Power & Light Company, Allegheny Electric Cooperatives, Inc. (Susquehanna Steam Electric Station, Units 1 and 2)," U.S. Nuclear Regulatory Commission, Docket Nos. 50-387 and 50-388, presented on October 14, 1981, in the transcript following page 1894.

Council on Environmental Quality, "The Seventh Annual Report of the Council on Environmental Quality," Figs. 11-27 and 11-28, pp. 238-239, September 1976.

Gotchy, R., testimony from "In the Matter of Duke Power Company (Perkins Nuclear Station)," U.S. Nuclear Regulatory Commission, Docket No. 50-488, filed April 17, 1978.

Killough, G. G., "A Diffusion-Type Model of the Global Carbon Cycle for the Estimation of Dose to the World Population from Releases of Carbon-14 to the Atmosphere," ORNL-5269, May 1977.

National Council on Radiation Protection and Measurements, NCRP, "Natural Background Radiation in the United States," NCRP Report No. 45, November 1975.

U.S. Environmental Protection Agency, "Environmental Standards for Uranium and Thorium Mill Tailings at Licensed Commercial Processing Sites (40 CFR 192)," Federal Register, Vol 48, No. 196, pp. 45926-45947, October 7, 1983.

U.S. Department of Energy, "Statistical Data of the Uranium Industry," GJO-100(8-78), January 1978.

U.S. Nuclear Regulatory Commission, NUREG-0002, "Final Generic Environmental Statement on the Use of Recycled Plutonium in Mixed Oxide Fuel in Light-Water-Cooled Reactors," August 1976.

---, NUREG-0116, "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle" (Supplement 1 to WASH-1248), October 1976.

Table C-1 Radon releases from mining and milling operations and mill tailings for each year of operation of the model 1000-MWe LWR*

Radon source	Quantity released
Mining**	4060 Ci
Milling and tailings*** (during active mining)	780 Ci
Inactive tailings*** (before stabilization)	350 Ci
Stabilized tailings*** (several hundred years)	1 to 10 Ci/year
Stabilized tailings*** (after several hundred years)	110 Ci/year

*After 3 days of hearings before the Atomic Safety and Licensing Appeal Board (ASLAB) using the Perkins record in a "lead case" approach, the ASLAB issued a decision on May 13, 1981 (ALAB-640) on the radon-222 release source term for the uranium fuel cycle. The decision, among other matters, produced new source term numbers based on the record developed at the hearings. These new numbers did not differ significantly from those in the Perkins record, which are the values set forth in this table. Any health effects relative to radon-222 are still under consideration before the ASLAB. Because the source term numbers in ALAB-640 do not differ significantly from those in the Perkins record, the staff continues to conclude that both the dose commitments and health effects of the uranium fuel cycle are insignificant when compared to dose commitments and potential health effects to the U.S. population resulting from all natural background sources. Subsequent to ALAB-640, a second ASLAB decision (ALAB-654, issued September 11, 1981) permits intervenors a 60-day period to challenge the Perkins record on the potential health effects of radon-222 emissions

**R. Wilde, NRC transcript of direct testimony given "In the Matter of Duke Power Company (Perkins Nuclear Station)," Docket No. 50-488, April 17, 1978.

***P. Magno, NRC transcript of direct testimony given "In the Matter of Duke Power Company (Perkins Nuclear Station)," Docket No. 50-488, April 17, 1978.

Table C-2 Estimated 100-year environmental dose commitment per year of operation of the model 1000-MWe LWR

		Environmental dose commitments			
Radon source	Radon-222 releases (Ci)	Total body (person rems)	Bone (person rems)	Lung (bronchial epithelium) (person rems)	Total body risk equivalent dose (person-rems)
Mining	4100	110	2800	2300	630
Milling and active tailings	1100	29	750	620	170
Total	5200	140	3600	2900	800

Table C-3 Estimated 100-year environmental dose commitments from unreclaimed open-pit mines for each year of operation of the model 1000-MWe LWR

		Environmental dose commitments			
Time span (years)	Radon-222 releases (Ci)	Total body (person-rems)	Bone (person rems)	Lung (bronchial epithelium) (person-rems)	Total body risk equivalent dose (person-rems)
100	3,700	96	2,500	2,000	550
500	19,000	480	13,000	11,000	3,000
1,000	37,000	960	25,000	20,000	5,500

Table C-4 Estimated 100-year environmental dose commitments from stabilized-tailings piles for each year of operation of the model 1000-MWe LWR

		Environmental dose commitments			
Time span (year)	Radon-222 releases (Ci)	Total body (person-rems)	Bone (person-rems)	Lung (bronchial epithelium) (person-rems)	Total body risk equivalent dose (person-rems)
100	100	2.6	68	56	15
500	4,090	110	2,800	2,300	630
1,000	53,800	1,400	37,000	30,000	8,200

Table C-5 Summary of 100-year environmental dose commitments per year of operation of the model 1000-MWe light-water reactor

Source	Total body (person-rems)	Total body risk equivalent (person-rems)
All nuclides in Table S-3 except radon-222 and technetium-99	550	650
Radon-222		
Mining, milling, and active tailings, 5200 Ci	140	800
Unreclaimed open-pit mines, 3700 Ci	96	550
Stabilized tailings, 100 Ci	3	15
Technetium-99, 1.3 Ci*	<1	<10
Total	790	2000

*Dose commitments are based on the "prompt" release of 1.3 Ci/RRY. Additional releases of technetium-99 are estimated to occur at a rate of 0.0039 Ci/yr/RRY after 2000 years of placing wastes in a high-level-waste repository.

APPENDIX D

EXAMPLES OF SITE-SPECIFIC DOSE ASSESSMENT CALCULATIONS

APPENDIX D

EXAMPLES OF SITE-SPECIFIC DOSE ASSESSMENT CALCULATIONS

1. Calculational Approach

As mentioned in the main body of this report, the quantities of radioactive material that may be released annually from the Hope Creek facility are estimated on the basis of the description of the design and operation of the radwaste systems as contained in the applicant's FSAR and by using the calculative models and parameters described in NUREG-0016. These estimated effluent release values for normal operation, including anticipated operational occurrences, along with the applicant's site and environmental data in the ER and in subsequent responses to NRC staff questions, are used in the calculation of radiation doses and dose commitments.

The models and considerations for environmental pathways that lead to estimates of radiation doses and dose commitments to individual members of the public near the plant and of cumulative doses and dose commitments to the entire population within an 80-km (50-mi) radius of the plant as a result of plant operations are discussed in detail in RG 1.109, Revision 1. Use of these models with additional assumptions for environmental pathways that lead to exposure to the general population outside the 80-km radius is described in Appendix B of this statement.

The calculations performed by the staff for the releases to the atmosphere and hydrosphere provide total integrated dose commitments to the entire population within 80 km of this facility based on the projected population distribution in the year 2000. The dose commitments represent the total dose that would be received over a 50-year period, following the intake of radioactivity for 1 year under the conditions existing 20 years after the station begins operation (that is, the mid-point of station operation). For younger persons, changes in organ mass and metabolic parameters with age after the initial intake of radioactivity are accounted for.

2. Dose Commitments From Radioactive Effluent Releases

The NRC staff's estimates of the expected gaseous and particulate releases (listed in Table D-1) along with the site meteorological considerations (summarized in Table D-2) were used to estimate radiation doses and dose commitments for airborne effluents. Individual receptor locations and pathway locations considered for the maximally exposed individual in these calculations are listed in Table D-3.

Annual average relative concentration (χ/Q) and relative deposition (D/Q) were calculated using the straight-line Gaussian atmospheric dispersion model described in RG 1.111, modified to reflect spatial and temporal variations in airflow using the correction factors contained in NUREG/CR-2919. All releases were assumed to be at ground level with mixing in the turbulent wake of plant structures.

A 5-year period of record (January 1977-December 1981) of onsite meteorological data was used for this evaluation. Wind-speed and -direction data were based on measurements made at the 10-m (33-ft) level, and atmospheric stability was defined by the vertical temperature gradient measured between the 10-m and 45.7-m (150-ft) levels.

In addition, the NRC staff estimates of the expected liquid releases (listed in Table D-4), along with the site hydrological considerations (summarized in Table D-5), were used to estimate radiation doses and dose commitments from liquid releases.

(a) Radiation Dose Commitments to Individual Members of the Public

As explained in the text, calculations are made for a hypothetical individual member of the public (that is, the maximally exposed individual) who would be expected to receive the highest radiation dose from all pathways that contribute. This method tends to overestimate the doses because assumptions are made that would be difficult for a real individual to fulfill.

The estimated dose commitments to the individual who is subject to maximum exposure at selected offsite locations from airborne releases of radioiodine and particulates, and waterborne releases are listed in Tables D-6 and D-7. The maximum annual total body and skin dose to a hypothetical individual and the maximum beta and gamma air dose at the site boundary are presented in Tables D-6 and D-7.

The maximally exposed individual is assumed to consume well above average quantities of the potentially affected foods and to spend more time at potentially affected locations than the average person as indicated in Tables E-4 and E-5 of Revision 1 of RG 1.109.

(b) Cumulative Dose Commitments to the General Population

Annual radiation dose commitments from airborne and waterborne radioactive releases from the Hope Creek facility are estimated for two populations in the year 2010: (1) all members of the general public within 80 km (50 mi) of the station (Table D-7) and (2) the entire U.S. population (Table D-8). Dose commitments beyond 80 km are based on the assumptions discussed in Appendix B. For perspective, annual background radiation doses are given in the tables for both populations.

3. References

U.S. Nuclear Regulatory Commission, NUREG-0016, F. P. Cardile and R. R. Bellamy (editors), "Calculation of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors," Revision 1, January 1979.

---, NUREG/CR-2919, "User Guide for XOQ DOQ: Evaluating Routine Effluent Releases at Commercial Nuclear Power Stations," J. F. Sagendorf, S. T. Goll, and W. F. Sandusky, September 1982.

Table D-1 Calculated releases of radioactive materials in gaseous effluents from Hope Creek Unit 1 (Ci/yr)

Nuclide	South plant vent (SPV)*	North plant vent*	Total for vents*	SPV**	SPV***	Total
Noble Gases						
Ar-41	a	a	a	1.5(+1)[b]	a	1.5(+1)
Kr-83	a	a	a	a	a	a
Kr-85m	2.9(+1)	a	2.9(+1)	a	a	2.9(+1)
Kr-85	a	2.2(+2)	2.2(+2)	a	a	2.2(+2)
Kr-87	6.3(+1)	a	6.3(+1)	a	a	6.3(+1)
Kr-88	9.5(+1)	a	9.5(+1)	a	a	9.5(+1)
Kr-89	6.1(+2)	a	6.1(+2)	a	a	6.1(+2)
Xe-131m	a	7.0(+0)[b]	7.0(+0)	a	a	7.0(+0)
Xe-133m	a	a	a	a	a	a
Xe-133	4.7(+2)	6.2(+1)	5.3(+2)	1.1(+1)	1.3(+3)	1.8(+3)
Xe-135m	9.9(+2)	a	9.9(+2)	a	a	9.9(+2)
Xe-135	7.4(+2)	a	7.4(+2)	3.0(+0)	5.0(+2)	1.2(+3)
Xe-137	1.3(+3)	a	1.3(+3)	a	a	1.3(+3)
Xe-138	1.0(+3)	a	1.0(+3)	a	a	1.0(+3)
Particulates						
P-32	3.0(-5)	c	3.0(-5)	c	c	3.0(-5)
Cr-51	1.8(-3)	c	1.8(-3)	c	1.0(-6)	1.8(-3)
Mn-54	6.6(-4)	c	6.6(-4)	c	c	6.6(-4)
Fe-55	1.5(-4)	c	1.5(-4)	c	c	1.5(-4)
Co-58	1.0(-3)	c	1.0(-3)	c	c	1.0(-3)
Fe-59	1.1(-4)	c	1.1(-4)	c	c	1.1(-4)
Co-60	1.2(-3)	c	1.2(-3)	c	5.6(-7)	1.2(-3)
Zn-65	6.1(-3)	c	6.1(-3)	c	3.4(-7)	6.1(-3)
Sr-89	6.0(-3)	c	6.0(-3)	c	c	6.0(-3)
Sr-90	2.0(-5)	c	2.0(-5)	c	c	2.0(-5)
Y-91	1.0(-5)	c	1.0(-5)	c	c	1.0(-5)
Nb-95	1.1(-4)	c	1.1(-4)	c	c	1.1(-4)
Zr-95	5.8(-5)	c	5.8(-5)	c	c	5.8(-5)
Mo-99	2.9(-3)	c	2.9(-3)	c	c	2.9(-3)
Ru-103	9.2(-5)	c	9.2(-5)	c	c	9.2(-5)
Ag-110m	2.4(-8)	c	2.4(-8)	c	c	2.4(-8)
Sb-124	1.0(-4)	c	1.0(-4)	c	c	1.0(-4)
Te-129m	1.0(-5)	c	1.0(-5)	c	c	1.0(-5)
Cs-134	2.7(-4)	c	2.7(-4)	c	3.2(-6)	2.7(-4)
Cs-136	1.1(-4)	c	1.1(-4)	c	1.9(-6)	1.1(-4)
Cs-137	1.1(-3)	c	1.1(-3)	c	8.9(-6)	1.1(-3)
Ba-140	1.0(-2)	c	1.0(-2)	c	1.1(-5)	1.0(-2)
Ce-141	1.0(-2)	c	1.0(-2)	c	c	1.0(-2)
Pr-143	1.0(-5)	c	1.0(-5)	c	c	1.0(-5)
Others						
C-14	a	9.5(+0)	9.5(+0)	a	a	9.5(+0)
H-3	5.2(+1)	-	5.2(+1)	-	-	5.2(+1)[d]
I-131	1.6(-1)	a	1.6(-1)	2.4(-4)	8.6(-2)	2.5(-1)
I-133	2.3(+0)	a	2.3(+0)	4.1(-4)	9.7(-1)	3.3(+0)

See footnotes on next page.

Table D-1 (Continued)

*1) Continuous releases via south plant vent for containment, auxiliary building including radwaste area. and turbine building exhausts. Releases from the radwaste area of the auxiliary building include the releases to the atmosphere as vapor after processing of the chemical wastes by the decontamination solution evaporator. These are given under Column 2 of the table.

2) Air ejector, that is, offgas system releases are continuous via the north plant vent and are given under Column 3.

3) Total, that is, north and south plant vents releases that are continuous are given under Column 4.

**Intermittent drywell purge releases. Release duration of 400 hours is assumed. These intermittent drywell purge releases are through the south plant vent. These are given under Column 5.

***Intermittent mechanical vacuum pump releases via the south plant vent are given under Column 6. Release duration of 400 hours is assumed.

a - Less than 1 Ci/yr for noble gases and C-14. Less than 1.0×10^{-4} Ci/yr for iodines.

b - Exponential notation: $1.5(+1) = 1.5 \times 10^{1} = 15.0$
$7.0(+0) = 7.0 \times 10^{0} = 7.0$

c - Neligible fraction of the total release for the isotope.

d - Total tritium gaseous effluent release is assumed to be from containment and turbine building per NUREG-0016, Rev. 1.

Table D-2 Summary of atmospheric dispersion factors (χ/Q) and relative deposition values for maximum site boundary and receptor locations near the Hope Creek Generating Station*

Location**	Source***	χ/Q (sec/m^3)	Relative deposition (m^{-2})
Nearest effluent-control boundary (0.59 km N)	A or B C	1.4×10^{-5} 2.4×10^{-5}	8.0×10^{-8} 1.4×10^{-7}
Nearest residence, garden, milk cow, and meat animal (5.6 km NW)	A or B C	2.1×10^{-7} 5.1×10^{-7}	8.0×10^{-10} 2.0×10^{-9}

*The values presented in this table are calculated in accordance with Regulatory Guide 1.111, Rev. 1, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light Water Reactors," July 1977.

**"Nearest" refers to that type of location where the highest radiation dose is expected to occur from all appropriate pathways.

***Sources:

A - South plant vent (on turbine building): continuous ground level releases from the containment, auxiliary building, and turbine building.

B - North plant vent (on turbine building): continuous ground level releases from the steam jet air ejector.

C - South plant vent: intermittent ground level releases from drywell purge and from mechanical vacuum pumping, assumed to be 400 hours each per year.

Table D-3 Nearest pathway locations used for maximally exposed individual dose commitments for the Hope Creek Generating Station

Location	Sector	Distance (km)
Nearest effluent-control boundary*	N	0.59
Residence, garden,** milk cow,*** and meat animal	NW	5.6
Milk goat†	-	-

*Beta and gamma air doses, total body doses, and skin doses from noble gases are determined at the effluent-control boundaries in the sector where the maximum potential value is likely to occur.

**Dose pathways including inhalation of atmospheric radioactivity, exposure to deposited radionuclides, and submersion in gaseous radioactivity are evaluated at residences. This particular location includes doses from vegetable consumption as well.

***It was conservatively assumed that a milk cow exists at this residence.

†None identified.

Table D-4 Calculated release of radioactive materials in liquid effluent from Hope Creek Unit 1

Nuclide	Ci/yr	Nuclide	Ci/yr
Corrosion and Activation Products		Fission Products (continued)	
Na-24	9.2(-3)*	Ru-103	3.2(-4)
P-32	4.5(-4)	Tc-104	3.0(-5)
Cr-51	1.3(-2)	Ru-105	8.4(-4)
Mn-54	3.9(-3)	Ru-106	8.9(-3)
Mn-56	1.1(-2)	Ag-110m	1.2(-3)
Fe-55	8.6(-3)	Te-129m	6.0(-5)
Fe-59	2.2(-3)	Te-131m	1.1(-4)
Co-58	8.2(-3)	I-131	1.3(-2)
Co-60	1.5(-2)	Te-132	1.0(-5)
Ni-63	1.7(-3)	I-132	7.5(-3)
Ni-65	6.0(-5)	I-133	4.0(-2)
Cu-64	2.6(-2)	I-134	2.0(-3)
Zn-65	2.9(-4)	Cs-134	1.1(-2)
Zn-69m	1.8(-3)	I-135	2.0(-2)
W-187	3.2(-4)	Cs-136	6.9(-4)
Np-239	9.8(-3)	Cs-137	1.7(-2)
		Cs-138	5.9(-4)
Fission Products		Ba-139	7.3(-4)
		Ba-140	1.5(-3)
Br-83	8.0(-4)	Ce-141	2.8(-4)
Br-84	3.0(-5)	La-142	5.3(-4)
Sr-89	2.3(-4)	Ce-143	3.0(-5)
Sr-90	2.0(-5)	Pr-143	6.0(-5)
Sr-91	3.0(-3)	Ce-144	3.9(-3)
Y-91	1.6(-4)		
Sr-92	2.3(-3)	All Others**	6.6(-3)
Y-92	5.1(-3)		
Y-93	3.1(-3)	Total (except H-3)	2.8(-1)
Zr-95	1.1(-3)	H-3	5.1(+1)
Nb-95	1.9(-3)		
Nb-98	9.0(-5)		
Mo-99	2.6(-3)		
Tc-99m	1.2(-2)		

*Exponential notation: $9.2(-3) = 9.2 \times 10^{-3}$.

**Nuclides whose annual releases are less than 10^{-5} Ci/yr are not listed individually, but are included in the category "All Others."

Table D-5 Summary of hydrologic transport and dispersion for liquid releases from the Hope Creek Generating Station*

Location	Transit time (hours)	Dilution factor
ALARA Dose Calculations		
Nearest sport-fishing location (discharge area)**	0	20
Nearest shoreline (bank of Delaware Estuary near discharge area)	0	20
Populations Dose Calculations		
Sport fishing (including invertebrates) shoreline usage, swimming and boating along the 80-km stretch of the Delaware Estuary downstream from the site	168	300
Commercial fishing (including invertebrates) along the 80-km stretch of the Delaware Estuary downstream from the site	248	300
Drinking water at the following distances downstream from the plant (km)***		
8	41	3.5
23	71	6.0
39	125	74.0
55	188	2,300
71	257	1,200

*See Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

**Assumed for purposes of an upper-limit estimate; detailed information not available.

***The transit times and dilution factors for drinking were from ER-OL, Table 5.2-1.

Table D-6 Annual dose commitments to a maximally exposed individual near the Hope Creek Generating Station

Location	Pathway	Doses (mrems/yr per unit, except as noted)			
		Noble gases in gaseous effluents			
		Total body	Skin	Gamma air dose (mrads/yr/unit)	Beta air dose (mrads/yr/unit)
Nearest* site boundary (0.59 km N)	Direct radiation from plume	3.9	8.7	5.9	5.7
		Iodine and particulates in gaseous effluents**			
		Total body	Organ		
Nearest*** site boundary (0.59 km N)	Ground deposition	0.15	0.17		(skin)
	Inhalation	a	8.6	(C)	(thyroid)
Nearest residence garden, milk cow, and meat animal (5.6 km NW)	Ground deposition	a	a	(I)	(thyroid)
	Inhalation	a	0.13	(I)	(thyroid)
	Vegetable consumption	a	a	(I)	(thyroid)
	Cow milk consumption	a	2.97	(I)	(thyroid)
	Meat deposition	a	a	(I)	(thyroid)
Nearest milk goat (none identified)	Ground deposition	-	-		
	Inhalation	-	-		
	Vegetable consumption	-	-		
	Goat milk consumption	-	-		
		Liquid effluents**			
		Total body	Organ		
Nearest drinking water	Water ingestion	b	b		
Nearest fish at plant-discharge area	Fish consumption	a (A)	a (A)		(bone)
Nearest shore access near plant-discharge area	Shoreline recreation	a (A or T)	a (A or T)		(skin)

*"Nearest" refers to that site boundary location where the highest radiation doses as a result of gaseous effluents have been estimated to occur.

**Doses are for the age group and organ that results in the highest cumulative dose for the location: A=adult, T=teen, C=child, I=infant. Calculations were made for these age groups and for the following organs: gastrointestinal tract, bone, liver, kidney, thyroid, lung, and skin.

***"Nearest" refers to the location where the highest radiation dose to an individual from all applicable pathways has been estimated.

[a]Less than 0.1 mrem/year.

[b]A dose from drinking water is unlikely. Site is in estuary portion of Delaware River; water is brackish. Also, no private wells within 2 mi.

Table D-7 Calculated 10 CFR 50 Appendix I dose commitments to a maximally exposed individual and to the population from operation of Hope Creek Generating Station

	Annual dose per reactor unit	
	Individual	
	Appendix I design objectives*	Calculated doses**
Liquid effluents		
Dose to total body from all pathways	3 mrems	a
Dose to any organ from all pathways	10 mrems	0.12 (GI-LLI)
Noble-gas effluents (at site boundary)		
Gamma dose in air	10 mrads	5.9 mrads
Beta dose in air	20 mrads	5.7 mrads
Dose to total body of an individual	5 mrems	3.9 mrems
Dose to skin of an individual	15 mrems	8.7 mrems
Radioiodines and particulates***		
Dose to any organ from all pathways	15 mrems	3.1 mrems (thyroid)

	Population dose within 80 km, person-rems	
	Total body	Thyroid
Natural-background radiation†	580,000	-
Liquid effluents	1.4	1.4
Noble-gas effluents	0.45	0.44
Radioiodine and particulates	1.2	12.0

*Design objectives from Sections II.A, II.B, II.C, and II.D of Appendix I, 10 CFR 50 consider doses to maximally exposed individual and to population per reactor unit.

**Numerical values in this column were obtained by summing appropriate values in Table D-6. Locations resulting in maximum doses are represented here.

***Carbon-14 and tritium have been added to this category.

†"Natural Radiation Exposure in the United States," U.S. Environmental Protection Agency, ORP-SID-72-1, June 1972; using the average background dose for New Jersey of 105 mrems/year, and year 2010 projected population of 5,400,000.

[a]Less than 0.1 mrem/year.

[b]Less than 0.1 mrad/year.

[c]Less than 0.1 person-rem.

Table D-8 Annual total-body population dose commitments, year 2010

Category	U.S. population dose commitment, person-rems/yr
Natural background radiation*	28,000,000*
Hope Creek Generating Station operation	
Plant workers	920
General public:	
Liquid effluents**	1.4
Gaseous effluents	22.0
Transportation of fuel and waste	3.0

*Using the average U.S. background dose (100 mrems/year) and year 2010 projected U.S. population from "Projections of the Population of the U.S. 1982-2050," Advance Report, U.S. Bureau of the Census, Department of Commerce, Current Population Reports, Series P-25, No. 922, October 1982.

**80-km (50-mi) population dose.

APPENDIX E

HISTORIC AND ARCHEOLOGICAL SITES

PSEG Public Service Electric and Gas Company

80 Park Plaza, Newark, NJ 07101 / 201 430 8217 MAILING ADDRESS / P.O. Box 570, Newark, NJ 07101

Robert L. Mittl General Manager
Nuclear Assurance and Regulation

March 27, 1984

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20014

Attention: Mr. Albert Schwencer, Chief
Licensing Branch 2
Division of Licensing

Gentlemen:

HOPE CREEK GENERATING STATION
DOCKET NO. 50-354
CULTURAL RESOURCES

Enclosed is a letter dated March 6, 1984 from Mr. Russell W. Myers, Deputy State Historic Preservation Officer, New Jersey Department of Environmental Protection, addressing the effects of Hope Creek Generating Station on cultural resources listed on or eligible to be listed on the National Register of Historic Places.

This letter is being submitted in accordance with Regulatory Guide 4.2, Section 2.6, which states: "The environmental report should contain evidence of contact with the Historic Preservation Officer for the state involved, including a copy of his comments concerning the effect of the undertaking on historic, archeological and cultural resources".

Should you have any questions in this regard, do not hesitate to contact Mr. D. E. Cooley at 201-430-8143.

Very truly yours,

R. L. Mittl

Enclosure

C D. H. Wagner
USNRC Licensing Project Manager

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The Energy People

State of New Jersey
DEPARTMENT OF ENVIRONMENTAL PROTECTION

ONJH-C84-30

DIVISION OF PARKS AND FORESTRY
OFFICE OF THE DIRECTOR

March 6, 1984

PLEASE ADDRESS REPLY TO
CN 404
TRENTON, N J 08625

Mr. James A. Shissias
General Manager, Environmental
Affairs
PSE & G
Post Office Box 570
Newark, NJ 07101

Re: Hope Creek Generating Station
Salem County

Dear Mr. Shissias:

The Office of New Jersey Heritage reviews federally funded or approved actions for their potential effects upon significant cultural resources. This letter serves as formal consultation comments as per 36 CFR Part 800: the Protection of Historic and Cultural Properties.

Review of maps and information on file and the materials submitted for review indicates that the project will have no effect upon cultural resources listed on or eligible for the National Register of Historic Places. In addition, since all facilities are already constructed, it is unlikely that any cultural resources will be affected by licensing this facility. Thank you for your cooperation.

If you have any questions, please feel free to contact Mr. John McCarthy of my staff at the Office of New Jersey Heritage (609) 292-2028.

Sincerely,

R.W. Myers

Russell W. Myers, Deputy
State Historic Preservation
Officer

RWM:JPM:ih

APPENDIX F

RSS METHODOLOGY APPLIED TO HOPE CREEK RELEASE CATEGORIES

APPENDIX F

RSS METHODOLOGY APPLIED TO HOPE CREEK RELEASE CATEGORIES

Neither the staff nor the licensee did a comprehensive probabilistic risk assessment (PRA) for Hope Creek. Instead, the staff used the best available results from in-depth PRAs for plants of similar design. The staff chose the six release categories previously determined for the rebaselined Peach Bottom analysis (NUREG-0773), and a release category from WASH-1400 (BWR 4), which accounts for accidents that do not catastrophically fail containment. The rebaselining effort and its results are described below, followed by a description of the BWR 4 release category. Some of the probabilities used in the release category were adjusted, on the basis of insights gained from in-depth analyses of other plants, including Limerick. These probabilities are discussed below. All the parameters used in the consequence analysis for each release category, including the probabilities, are listed in Table 5.14 of the main text.

The update of the results of the Reactor Safety Study (RSS) (WASH-1400) was done largely to incorporate results of research and development conducted after the October 1975 publication of the RSS (NUREG-75/014) and to provide a baseline against which the risk associated with various LWRs could be consistently compared.

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

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

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*It should be noted that the MARCH code was used on a number of scenarios in connection with the TMI-2 recovery efforts and for post-TMI-2 investigations to explore possible alternative scenarios that TMI-2 could have experienced.

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

For rebaselining of the RSS BWR design (Peach Bottom), the sequence TCγ' (described later) was explicitly included into the rebaselining results. The accident processes associated with the TC sequence had been erroneously calculated in WASH-1400. In general, the rebaselined results led to slightly increased health impacts being predicted for the RSS BWR design. This is believed to be largely attributable to the inclusion of TCγ'.

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

The accident sequences identified in the rebaselining effort which are expected to dominate risk of the RSS BWR design are briefly described below. These sequences are assumed to represent the approximate accident risks from the Hope Creek BWR design.

Each of the accident sequences is designated by a string of identification characters in the same manner as in the RSS (see page 6 of this appendix for key to sequence symbols). Each character represents a failure in one or more of the important plant systems or features. For example, in sequences having a γ' at the end of the string, the γ' indicates a particular failure mode (overpressure) of the containment structure (and a rupture location) where a release of radioactivity takes place directly to the atmosphere from the primary containment. In the sequence having a γ at the end of the string, the containment failure mode is again by overpressure, but this time the rupture location is such that the release takes place into the reactor building (secondary containment) before discharging to the environment. In this latter (γ) case, the overall magnitude of radioactivity release is somewhat diminished by the deposition and plateout processes that take place within the reactor building.

<u>TCγ' and TCγ</u>

These sequences involve a transient event requiring shutdown of the reactor while at full power, followed by a failure to make the reactor subcritical (that is, terminate power generation by the core). The containment is assumed to be isolated by these events; then, one or the other of the following chain of events is assumed to happen:

(1) The high pressure coolant injection (HPCI) system would succeed for some time in providing makeup water to the core in sufficient quantity to cope with the rate of coolant loss through relief and safety valves to the suppression pool of the containment. During this time, the core power level varies, but causes substantial energy to be directed into the suppression

pool; this energy is in excess of what the containment and containment heat removal systems are designed to cope with. Ultimately, in about 1-1/3 hours, the containment is estimated to fail by overpressure and it is assumed that this rather severe structural failure of the containment would disable the high pressure coolant makeup system. Over a period of roughly 1-1/2 hours after breach of containment, it is assumed the core would melt. This has been estimated to be one of the more dominant sequences in terms of accident risks to the public.

(2) A variant to the above sequence is one where the high pressure coolant injection system fails somewhat earlier and before containment over-pressure failure. In this case, the earlier melt could result in a reduced magnitude of release because some of the fission products discharged to the suppression pool via the safety and relief valves could be more effectively retained if the pool remained subcooled. The overall accident consequences would be somewhat reduced in this earlier melt sequence, but ultimately the processes accompanying melt (for example, noncondensibles, steam, and steam pressure pulses during reactor vessel melt-through) could cause overpressure failure (γ or γ') of the containment.

The probabilities assigned to these sequences for the rebaselined Peach Bottom study were 2×10^{-6} per reactor-year for $TC\gamma'$ and 8×10^{-6} per reactor-year for $TC\gamma$. In the absence of a detailed PRA, the staff determined that the same probabilities per reactor-year, and the identical release categories, would be used for Hope Creek.

Since the rebaselining effort, anticipated transients without scram (ATWS) generic review activities were completed, including a discussion of ATWS probability. According to A. C. Thadani's memo to L. G. Hulman, "In light of the results in the ATWS rule, until the ATWS modifications are known to be implemented at RBS [and, by later reference, at Hope Creek], we suggest that a frequency of 5×10^{-5} per reactor year be used in accident evaluation. If the modifications are in place, the frequency would be reduced to approximately 1×10^{-5}/RY." Because the Hope Creek applicant has implemented the modifications, including an improved standby liquid control system (which will shut the reactor down if the control rods fail to insert), a total probability of 1×10^{-5} per reactor year was used for $TC\gamma'$ and $TC\gamma$. Keeping the same ratio between the two probabilities as before, the probabilities of 2×10^{-6} for $TC\gamma'$ and 8×10^{-6} per reactor-year were obtained.

<u>$TW\gamma'$ and $TW\gamma$</u>

The TW sequence involves a transient where the reactor has been shut down and containment has been isolated from its normal heat sink (that is, the power conversion system). In this sequence, the failure to transfer decay heat from the core and containment to an ultimate sink could ultimately cause overpressure failure of containment. Overpressure failure of containment would take many, many hours, allowing for repair or other emergency actions to be accomplished, but should this sequence occur, it is assumed that the rather severe structural failure of containment would disable the systems (for example, HPCI, reactor core isolation cooling (RCIC)) providing coolant makeup to the reactor core. (In the RSS design, the service water system which conveys heat from the containment via the residual heat removal system to the ultimate sink was found to be the dominant failure contribution in the TW sequence.) After breach of containment, the core is assumed to melt.

The probabilities determined for Peach Bottom were 3 x 10^{-6} per reactor-year for TWγ' and 1 x 10^{-5} for TWγ. These probabilities were also used for Hope Creek.

<u>[TQUVγ', AEγ', S_γEγ', S_2Eγ'] and [TQUVγ, AEγ, S_γEγ, S_2Eγ]</u>

Each of the accident sequences shown grouped into the two bracketed categories above is estimated to have quite similar consequence outcomes, and these would be somewhat smaller than the TCγ', TCγ and TWγ' sequences described above. In essence, these sequences, which are characterized as in the RSS, involve failure to deliver makeup coolant to the core after a LOCA or a shutdown transient event requiring such coolant makeup. The core is assumed to melt down and the melt processes ultimately cause overpressure failure of containment (either γ' or γ). The overall risk from these sequences is expected to be dominated by the higher frequency initiating events (that is, the small LOCA (S_2) and shutdown transients (T)).

On the basis of insights gained during the analysis of initiating events at several plants (including the plants examined in the Reactor Safety Study Methodology Applications Program; Accident Sequence Evaluation Program Studies, sponsored by the NRC Office of Nuclear Regulatory Research; and review of the Limerick PRA), the probabilities per reactor-year of these release categories were revised upward. The total probability per reactor-year of core melts other than TCγ, TCγ', TWγ, and TWγ' (discussed above) was estimated to be 8 x 10^{-5} per reactor-year. These other core melts result in releases via three pathways: directly from the containment to the environment, to the environment via the reactor building, and to the environment via the filtration, recirculation, and ventilation system (FRVS). The latter release pathway depends on the FRVS performing its intended function of preventing overpressure of the reactor building and filtering the flow from the reactor building to the environment. Sequences for which this is true were assigned to the release category BWR 4, described below. The staff determined that the conditional probability of the FRVS preventing a major failure was 0.5, meaning that the probability per reactor-year of BWR 4 is estimated to be 4 x 10^{-5}, and the sum of the probabilities per reactor-year of the other two groups of sequences is also 4 x 10^{-5}. For Peach Bottom, the releases that were postulated would go directly to the atmosphere (TQUVγ', AEγ', $S_1E^{1'}$, and S_2Eγ') were estimated to have about one-fourth of the total probability. That is, the conditional probability of containment failure from overpressurization and release directly to the environment (that is, the γ' failure as opposed to γ) was estimated to be about 0.25. This conditional probability was estimated to be about the same for Hope Creek, since both Peach Bottom and Hope Creek have Mark I containments with similar relative structural strengths and weaknesses. For Hope Creek, therefore, the following probabilities per reactor-year were used: for the group (TQUVγ', AEγ', S_1Eγ', and S_2Eγ'), probability = 0.25 x 4 x 10^{-5} = 10^{-5}; for the group (TQUVγ, AEγ, S_1Eγ, and S_2Eγ), probability = (1 - 0.25) x 4 x 10^{-5} = 3 x 10^{-5}. (Note that (1 - 0.25) is the conditional probability of γ, a release through the reactor building.)

BWR 4

The sequences that do not lead to overpressurization of containment include TQUV, AE, S_1E, S_2E, and similar sequences. These sequences were assigned to the release category BWR 4, described in WASH-1400, Appendix VI, p. 2-4, with a probability per reactor-year of 4×10^{-5}. The release category description is reproduced below for completeness.

> This release category is representative of a core meltdown with enough containment leakage to the reactor building to prevent containment failure by overpressure. The quantity of radioactivity released to the atmosphere would be significantly reduced by normal ventilation paths in the reactor building and potential mitigation by the secondary containment filter systems. Condensation in the containment and the action of the standby gas treatment system on the releases would also lead to a low rate of energy release. The radioactive material would be released from the reactor building or the stack at an elevated level. Most of the release would occur over a 2-hour period and would involve approximately 0.08% of the iodines and 0.5% of the alkali metals.

References

Thadani, A. C., memorandum to L. G. Hulman, March 16, 1984, which was referenced (for Hope Creek) by A. C. Thadani and B. W. Sheron memorandum to L. G. Hulman, March 22, 1984.

U.S. Atomic Energy Commission, WASH-1400, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," The Rasmussen Report, August 1974.

U.S. Nuclear Regulatory Commission, NUREG-75/014 (formerly WASH-1400), "Reactor Safety Study - An Assessment," October 1975.

---, NUREG-0773, "The Development of Severe Reactor Accident Source Terms: 1957-1981," R. Blond et al., 1982.

---, NUREG/CR-0400, "Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission, H. W. Lewis et al., September 1978.

KEY TO BWR ACCIDENT SEQUENCE SYMBOLS

A	-	Rupture of reactor coolant boundary with an equivalent diameter greater than 6 in.
B	-	Failure of electric power to engineered safety features
C	-	Failure of the reactor protection system
D	-	Failure of vapor suppression
E	-	Failure of emergency core cooling injection
F	-	Failure of emergency core cooling functionability
G	-	Failure of containment isolation to limit leakage to less than 100 volume percent per day
H	-	Failure of core spray recirculation system
I	-	Failure of low pressure service water system
J	-	Failure of high pressure service water system
M	-	Failure of safety/relief valves to open
P	-	Failure of safety/relief valves to reclose after opening
Q	-	Failure of normal feedwater system to provide core makeup water
S_1	-	Small pipe break with an equivalent diameter of about 2 in. - 6 in.
S_2	-	Small pipe break with an equivalent diameter of about 1/2 in.- 2 in.
T	-	Transient event
U	-	Failure of HPCI or RCIC to provide core makeup water
V	-	Failure of low pressure emergency core cooling system to provide core makeup water
W	-	Failure to remove residual core heat
α	-	Containment failure from steam explosion in vessel
β	-	Containment failure from steam explosion in containment
γ	-	Containment failure from overpressure - release through reactor building
γ'	-	Containment failure from overpressure - release direct to atmosphere
Δ	-	Containment isolation failure in drywell
ε	-	Containment isolation failure in wetwell
ζ	-	Containment leakage greater than 2,400 volume percent per day
η	-	Reactor building isolation failure
θ	-	Standby gas treatment system failure

APPENDIX G

CONSEQUENCE MODELING CONSIDERATIONS

APPENDIX G

CONSEQUENCE MODELING CONSIDERATIONS

G.1 Evacuation Model

"Evacuation," used in the context of offsite emergency response in the event of a substantial amount of radioactivity release to the atmosphere in a reactor accident, denotes an early and expeditious movement of people to avoid exposure to the passing radioactive cloud and/or to acute ground contamination in the wake of the cloud passage. It should be distinguished from "relocation," which denotes a postaccident response to reduce exposure from long-term ground contamination after plume passage. The Reactor Safety Study (RSS) (NUREG-75/014, originally WASH-1400) consequence model contains provision for incorporating radiological consequence reduction benefits of public evacuation. The benefits of a properly planned and expeditiously carried out public evacuation would be well manifested in a reduction of early health effects associated with early exposure; namely, in the number of cases of early fatality (see Section G.2) and acute radiation sickness which would require hospitalization. The evacuation model originally used in the RSS consequence model is described in WASH-1400 as well as in NUREG-0340 and NUREG/CR-2300. The evacuation model which has been used herein is a modified version of the RSS model (Sandia, 1978) and is, to a certain extent, site emergency planning oriented. The modified version is briefly outlined below.

The model uses a circular area with a specified radius (the 16-km (10-mi) plume exposure pathway emergency planning zone (EPZ)), with the reactor at the center. It is assumed that people living within portions of this area would evacuate if an accident should occur involving imminent or actual release of significant quantities of radioactivity to the atmosphere.

Significant atmospheric releases of radioactivity would in general be preceded by one or more hours of warning time (postulated as the time interval between the awareness of impending core melt and the beginning of the release of radioactivity from the containment building). This warning time is given for each release category in Table 5.14. For the purpose of calculation of radiological exposure, the model assumes that all people who live in a fan-shaped area (fanning out from the reactor) within the circular zone with the downwind direction as its median - that is, those people who would potentially be under the radioactive cloud that would develop following the release - would leave their residences after lapse of a specified amount of delay time* and then evacuate. The delay time is reckoned from the beginning of the warning time and is recognized as the sum of: the time required by the reactor operators to notify the responsible authorities; the time required by the authorities to interpret the data, decide to evacuate, and direct the people to evacuate; and the time required for the people to mobilize and get under way.

*Assumed to be a constant value, 1 hour, that would be the same for all evacuees.

The model assumes that each evacuee would move radially outward* away from the reactor with an average effective speed** (obtained by dividing the zone radius by the average time taken to clear the zone after the delay time) over a fixed distance from the evacuee's starting point. This distance is selected to be 24 km (15 mi) (which is 8 km (5 mi) more than the 16-km (10-mi) plume exposure pathway EPZ radius). After reaching the end of the travel distance, the evacuee is assumed to receive no further radiation exposure.

The model incorporates a finite length of the radioactive cloud in the downwind direction that would be determined by the product of the duration over which the atmospheric release would take place and the average wind speed during the release. It is assumed that the front and the back of the cloud would move with an equal speed that would be the same as the prevailing wind speed; therefore, its length would remain constant at its initial value. At any time after the release, the concentration of radioactivity is assumed to be uniform over the length of the cloud. If the delay time were less than the warning time, then all evacuees would have a head start; that is, the cloud would be trailing behind the evacuees initially. On the other hand, if the delay time were more than the warning time, then depending on initial locations of the evacuees there are possibilities that (1) an evacuee would still have a head start, or (2) the cloud would be already overhead when an evacuee starts to leave, or (3) an evacuee would be initially trailing behind the cloud. However, this initial picture of cloud/people disposition would change as the evacuees travel, depending on the relative speed and positions between the cloud and people. The cloud and an evacuee might overtake one another one or more times before the evacuee would reach his/her destination. In the model, the radial position of an evacuating person, either stationary or in transit, is compared to the front and the back of the cloud as a function of time to determine a realistic period of exposure to airborne radionuclides. The model calculates the time periods during which people are exposed to radionuclides on the ground while they are stationary and while they are evacuating. Because radionuclides would be deposited continually from the cloud as it passed a given location, a person who is under the cloud would be exposed to ground contamination less concentrated than if the cloud had completely passed. To account for this, at least in part, the revised model assumes that persons are: (1) exposed to the total ground contamination concentration that is calculated to exist after complete passage of the cloud, after they are completely passed by the cloud; (2) exposed to one-half the calculated concentration when anywhere under the cloud; and (3) not exposed when they are in front of the cloud. Different values of the shielding protection factors for exposures from airborne radioactivity and ground contamination have been used.

Results shown in Section 5.9.4.5 of the main body of this environmental statement for accidents involving significant release of radioactivity to the atmosphere were based on the assumption that all people within the 16-km (10-mi) plume exposure pathway EPZ would evacuate according to the evacuation scenario described above. Because sheltering can be a mitigative feature, it is not expected that detailed inclusion of any facility (see Section 5.9.4.5(2)) near a specific plant site, where not all persons would be quickly evacuated, would

*In the RSS consequence model, the radioactive cloud is assumed to travel radially outward only, spreading out as it moves away.

**Assumed to be a constant value, 5.5 km (3.4 mi) per hour, that would be the same for all evacuees.

significantly alter the conclusions. For the delay time before evacuation, a value of 1 hour was used. The staff believes that such a value appropriately reflects the Commission's emergency planning requirements. The applicant has provided estimates of the time required to clear the 16-km (10-mi) zone (see Parsons, et al., 1981).

From these estimates, the staff has conservatively estimated the effective evacuation speed to be 1.5 meters per second (3.4 mph). It is realistic to expect that the authorities would aid and encourage evacuation at distances from the site where exposures above the threshold for causing early fatalities could be reached regardless of the EPZ distance. The sensitivity of the early fatalities to evacuation distance was calculated by assuming the longer evacuation distance of 24 km (15 mi) from Hope Creek. As an additional emergency measure for the Hope Creek site, it was also assumed that all people beyond the evacuation distance who would be exposed to the contaminated ground would be relocated after passage of the plume. A modification of the RSS consequence model was used, which incorporates the assumption that if the calculated ground dose to the total marrow over a 7-day period would exceed 200 rems, then this high dose rate would be detected by actual field measurements following plume passage, and people from these regions would be relocated immediately. For this situation the model limits the period of ground dose calculation to 12 hours; otherwise, the period of ground exposure is limited to 7 days for calculation of early dose.

Figure G.1 shows the early fatalities for (1) evacuation distances of 24 km (15 mi) followed by relocation as described above, (2) a pessimistic case for which no early evacuation is assumed and all persons are assumed to be exposed for the first 24 hours following an accident and are then relocated, (3) a case of evacuation to 16 km (10 mi) followed by relocation of those outside 16 km as described above. (This case is judged most realistic and was the case used for the calculation in Section 5.9.4.)

The model has the same provision for calculation of the economic cost associated with implementation of evacuation as the original RSS model. For this purpose, the model assumes that for atmospheric releases of durations of 3 hours or less, all people living within a circular area of 8-km (5-mi) radius centered at the reactor plus all people within a 45° angular sector within the plume exposure pathway EPZ and centered on the downwind direction will be evacuated and temporarily relocated. However, if the duration of release were to exceed 3 hours, the cost of evacuation is based on the assumption that all people within the entire plume exposure pathway EPZ would be evacuated and temporarily relocated. For either of these situations, the cost of evacuation and relocation is assumed to be $225 (1980 dollars) per person, which includes cost of food and temporary sheltering for a period of 1 week.

G.2 Early Health Effects Model

The medical advisors to the RSS (WASH-1400, Appendix IV, Section 9.2.2, and Appendix F) proposed three alternative dose-mortality relationships that can be used to estimate the number of early fatalities that might result in an exposed population. These alternatives characterize different degrees of post-exposure medical treatment from "minimal," to "supportive," to "heroic"; they are more fully described in NUREG-0340. There is uncertainty associated with the mortality relationships (NUREG/CR-3185) and the availability and effectiveness of different classes of medical treatment (Elliot, 1982).

The calculative estimates of the early fatality risks presented in the text of Section 5.9.4.5(3) of the main body of this report and in Section G.1 of this appendix used the dose-mortality relationship that is based on the supportive treatment alternative. This implies the availability of medical care facilities and services that are designed for radiation victims, for those exposed in excess of 170 rems, the approximate level above which the medical advisors to the RSS recommended more than minimum medical care to reduce early fatality risks. At the extreme low probability end of the spectrum (that is, at the one chance in four million per reactor-year level), the number of persons involved might exceed the capacity of facilities that provide the best such services, in which case the number of early fatalities might have been underestimated. To gain perspective on this element of uncertainty, the staff has also performed calculations using the most pessimistic dose-mortality relationship based on the RSS medical expert's estimated dose-mortality relationship for minimal medical treatment and using identical assumptions regarding early evacuation and early relocation as made in Section 5.9.4.5(3). This shows an overall 30-fold increase in annual risk of early fatalities (see Table 5.17). The major fraction of the increased risk of early fatality in the absence of supportive medical treatment would occur within 28 km (17.5 mi), and virtually all would be within 64 km (40 mi) of the Hope Creek site. However, the hospitals now in the United States are likely to be able to supply considerably better care to radiation victims than the medical care on which the minimal medical treatment relationship is based. Further, a major reactor accident at Hope Creek would certainly cause a mobilization of such medical services with a high national priority to save the lives of radiation victims. Therefore, the staff expects that the mortality risks would be less than those indicated by the RSS description of minimal treatment (and much less, of course, for those who will be given the type of treatment defined as "supportive"). For these reasons, the staff has concluded that the early fatality risk estimates are bounded by the range of uncertainties discussed in Section 5.9.4.5(7).

G.3 References

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

Parsons, Brinckerhoff, Quade and Douglas, for Public Service Electric & Gas Company, "Evacuation Time Estimates for Areas Near the Site of Salem and Hope Creek Nuclear Generating Stations," February 1981.

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

U.S. Nuclear Regulatory Commission, NUREG-75/014 (formerly WASH-1400), "Reactor Safety Study," October 1975.

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

---, NUREG/CR-2300, "PRA Procedures Guide," January 1983.

---, NUREG/CR-3185, "Critical Review of the Reactor Safety Study Radiological Health Effects Model," March 1983.

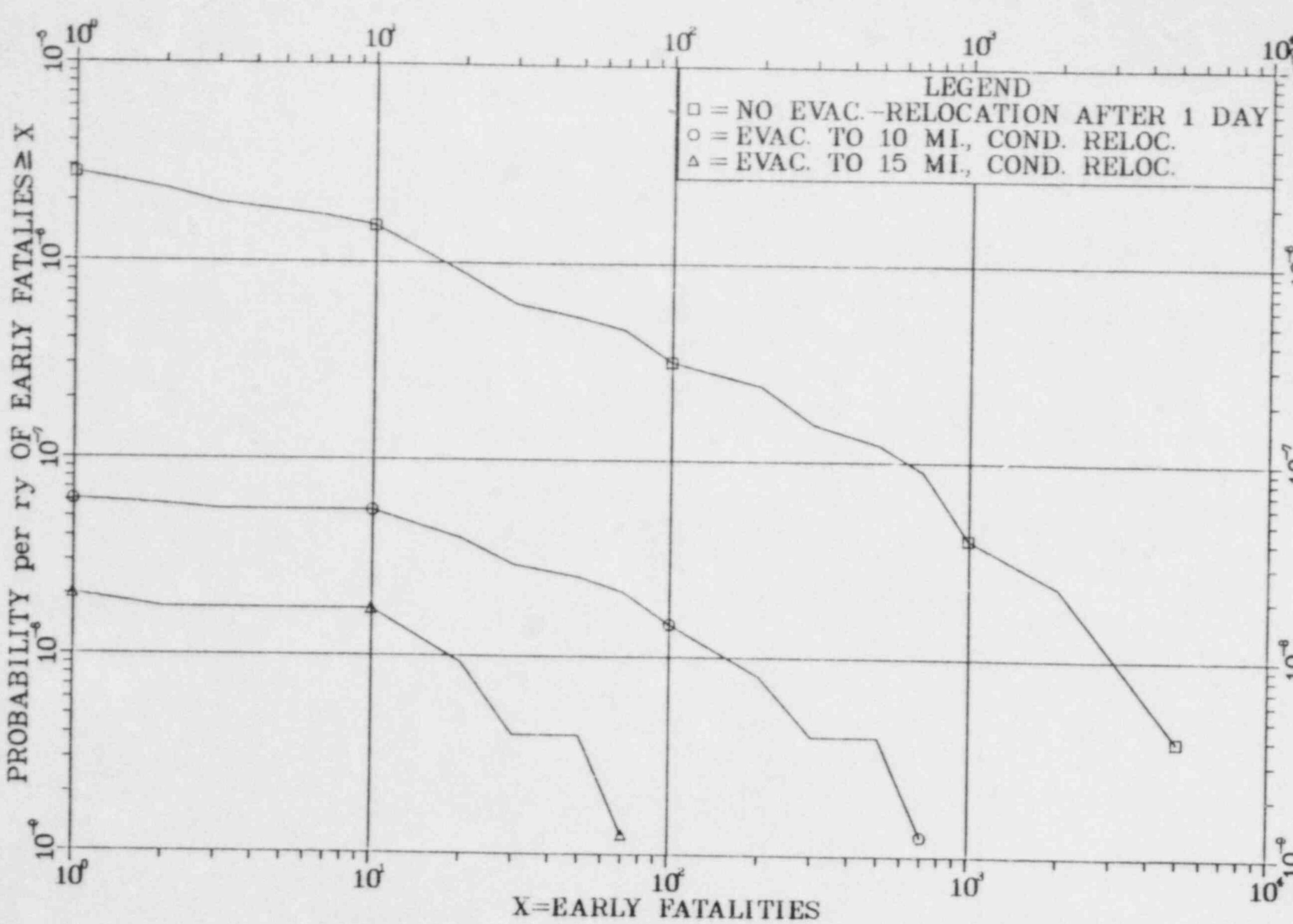

Figure G.1 Sensitivity of early fatalities to evacuation characteristics
NOTE: See Section 5.9.4.5(7) for a discussion of uncertainties.

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The Draft Environmental Statement related to the operation of Hope Creek Generating Station, located in Salem County, New Jersey, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The statement reports on staff's review of the environmental and socio-economic impacts of plant operation. Comments received on this document will be included and addressed in the Final Environmental Statement.

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