NUREG-0683
Supplement No. 3
Draft Report

Programmatic
Environmental Impact Statement

related to decontamination and disposal
of radioactive wastes resulting from
March 28, 1979 accident
Three Mile Island Nuclear Station, Unit 2
Docket No. 50-320

Draft Supplement Dealing with
Post-Defueling Monitored Storage
and Subsequent Cleanup

GPU Nuclear, Inc.

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 1988

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ABSTRACT

1. Proposed Action and Location:

   DECONTAMINATION AND DISPOSAL OF RADIOACTIVE WASTES RESULTING FROM THE MARCH 28, 1979, ACCIDENT AT THREE MILE ISLAND NUCLEAR STATION, UNIT 2, LOCATED IN LONDONDERRY TOWNSHIP, DAUPHIN COUNTY, PENNSYLVANIA

2. Comments should be filed no later than 45 days after the date on which the Environmental Protection Agency's notice of availability of this draft supplement to the Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from March 28, 1979 Accident Three Mile Island Nuclear Station, Unit 2 is published in the Federal Register.

3. Further information may be obtained from Dr. Michael T. Masnik, the Project Manager for this draft supplement. He may be contacted at the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, or at (301) 492-1373.

4. In accordance with the National Environmental Policy Act, the Commission's implementing regulations, and its April 27, 1981, Statement of Policy, the Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from March 28, 1979 Accident Three Mile Island Nuclear Station, Unit 2, NUREG-0683 (PEIS) is being supplemented. This draft supplement updates the environmental evaluation of cleanup alternatives published in the PEIS, utilizing more complete and current information. Also, the draft supplement includes a specific environmental evaluation of the licensee's recently submitted proposal for post-defueling monitored storage.

5. The NRC staff has concluded that the licensee's proposal to place the facility in a monitored storage configuration will not significantly affect the quality of the human environment. Further, any impacts from the long-term storage of the facility are outweighed by its benefits.

SUMMARY

The final Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from March 28, 1979 Accident Three Mile Island Nuclear Station, Unit 2 was issued as NUREG-0683 by the U.S. Nuclear Regulatory Commission (NRC) in March 1981. That document (referred to as the PEIS) was intended to provide an overall evaluation of the environmental impacts that could result from cleanup activities at Three Mile Island Unit 2 (TMI-2). Following the publication of the PEIS, the Commission issued a Policy Statement on April 28, 1981, indicating that the NRC staff would evaluate and act on major cleanup proposals as long as the impacts associated with the proposed activities fell within the scope of the impacts already assessed in the PEIS.

The TMI-2 cleanup can be categorized into four fundamental activities: building and equipment decontamination; fuel removal and reactor coolant system decontamination; treatment of radioactive liquids; and packaging, handling, shipment, and disposal of radioactive wastes. Since the 1979 accident, the licensee's (GPU Nuclear's) cleanup program has resulted in substantial cleanup progress in each of these fundamental activities. In addition to having treated all of the water that contained radioactive materials as a result of the accident, facility decontamination efforts have been successful in returning most areas in the auxiliary and fuel handling building to pre-accident radiological conditions, disposal of radioactive wastes has been actively proceeding, and defueling efforts through December 1987 have resulted in removal of more than 60 percent of the damaged core. The licensee projects completion of the current defueling program by January 1989.(a)

The purpose of this supplement to the PEIS is to evaluate the potential environmental impacts of alternative approaches to completing the TMI-2 cleanup. The licensee has submitted a proposal to maintain the TMI-2 facility in a monitored storage mode [referred to by the licensee as "post-defueling monitored storage" (PDMS)] for an unspecified period of time following current efforts to remove the damaged fuel. In addition to removing more than 99 percent of the fuel, major portions of the reactor building and the auxiliary and fuel-handling building (AFHB) would be decontaminated before PDMS, but not to the extent that the cleanup could be considered completed. Following the storage period, the decontamination process would be resumed and completed. This alternative is referred to in this document as "delayed cleanup." Although the duration of the storage period was not specified by the licensee, the NRC staff evaluated delayed cleanup assuming a storage period of 20 years.

(a) During the preparation of this supplement, the staff believed that the licensee would be able to remove essentially all of the fuel from the reactor vessel. However, recent findings indicate that some fuel may remain in the vessel. The final PEIS Supplement 3 will quantify the amount of fuel that will remain in the vessel and identify the measures that will be taken to reduce its environmental impact. The presence of this additional fuel, however, is not expected to change the conclusions given here.

In accordance with the requirements of the National Environmental Policy Act (NEPA) and the Commission's implementing regulations, both the licensee's plan and alternative approaches were examined for their potential environmental impacts. Six alternatives to the licensee's proposal were identified by the NRC staff: (1) immediate cleanup (the continuation and completion of the cleanup at the present level of effort without a storage period), (2) immediate cleanup with a reduced level of effort, (3) additional cleanup (i.e., more extensive cleaning than that proposed by the licensee) before entering PDMS, (4) delayed cleanup with storage less than 20 years, (5) delayed cleanup with storage longer than 20 years, and (6) no further cleanup following defueling, the "no-action" alternative, which is required by NEPA to be considered as part of all environmental impact statements. Delayed cleanup and immediate cleanup were quantitatively evaluated relative to their environmental impacts, including radiation exposure to the offsite population from routine and accidental releases, occupational radiation dose, waste management impacts (including transportation impacts), direct socioeconomic impacts, commitment of resources, and regulatory considerations. The impacts of the latter five alternatives were discussed but not quantitatively evaluated because their impacts would be bounded by the impacts of the licensee's proposal for delayed cleanup and the NRC-identified alternative of immediate cleanup.

The potential environmental impacts associated with delayed cleanup and immediate cleanup are summarized in Table S.1. Estimates of the cancer mortality risks to workers and the general public were based on conservative assumptions (that is, the estimates are probably higher than the actual numbers). Delayed cleanup was estimated to result in a maximum of 0.2 radiation-induced cancer fatalities in the worker population (i.e., approximately 2 chances in 10 that a single cancer death would occur in the total population of occupationally exposed workers as a result of delayed cleanup operations). Immediate cleanup was estimated to result in a maximum of 0.4 radiation-induced cancer fatalities in the worker population (i.e., approximately 4 chances in 10 that a single cancer death would occur in the total population of occupationally exposed workers as a result of immediate cleanup operations). The impact on the work force associated with final cleanup would be comparable to that currently experienced at operating plants. Radiation-induced cancer fatalities in the offsite population residing within 50 miles (80 kilometers) of the site were estimated to be 0.002 for delayed cleanup [i.e., approximately 2 chances in 1000 that a single cancer death would occur in the offsite population (between 2.2 million and 3.2 million people) as a result of delayed cleanup], and 0.0003 for immediate cleanup [i.e., approximately 3 chances out of 10,000 of a single cancer death in the offsite population (2.2 million people) as a result of immediate cleanup]. The estimated number of traffic fatalities during waste shipments is 0.02 to 0.05 for delayed cleanup (i.e., approximately 2 to 5 chances out of 100 that a fatal accident would occur) and 0.1 to 0.2 for immediate cleanup (i.e., approximately 1 to 2 chances out of 10 that a fatal accident would occur).

The NRC staff has concluded, based on this evaluation, that the licensee's proposed plan and the NRC staff-identified alternatives for completion of cleanup are within the applicable regulatory limits and could each be implemented without significant environmental impact. No alternative was found to be clearly preferable from an environmental impact perspective.

TABLE S.1. Impacts from Delayed Cleanup and Immediate Cleanup

Impacts	Delayed Cleanup (24-Year Period)	Immediate Cleanup (4-Year Period)
Occupational dose	48 to 1500 person-rem	300 to 3[illegible]0 person-rem
Estimated number of radiation-induced cancer fatalities in the worker population	0.006 to 0.2	0.04 to 0.4
Estimated number of traffic accidents	0.5 to 1	1 to 3
Estimated number of traffic injuries	0.3 to 0.6	1 to 3
Estimated number of traffic fatalities	0.02 to 0.05	0.1 to 0.2
Bone dose to the offsite population		
Maximally exposed individual	29 mrem	7 mrem
Total population	20 person-rem	6 person-rem
Total body dose to the offsite population within a 50-mile radius of TMI-2		
Maximally exposed individual	4 mrem	0.7 mrem
Total population	11 person-rem	2 person-rem
Estimated number of radiation-induced cancer fatalities in the offsite population	0.002	0.0003
Estimated number of radiation-induced genetic disorders in the offsite population	0.006 to 0.1	0.02 to 0.2
Cost ($ millions)	200 to 320	170 to 240
Radioactive waste burial ground volume	33,000 to 74,000 ft^3	32,000 to 70,000 ft^3

In addition, the staff concluded that the "no further cleanup following defueling" or "no-action" alternative is not acceptable because this course would not result in elimination of the public health and safety risk associated with the damaged facility. Accordingly, the staff concluded that the benefits of cleanup action outweigh the small associated impacts and that the licensee's proposed approach to completing the cleanup will not significantly affect the quality of the human environment.

This draft supplement is circulated to allow public input to the decision-making process. Following consideration of the comments received, a final supplement will be issued.

FOREWORD

This draft supplement to the Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from March 28, 1979, Accident Three Mile Island Nuclear Station, Unit 2 (PEIS) was prepared by the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation (referred to as the staff), pursuant to the Commission's April 27, 1981, Statement of Policy related to the PEIS and the requirements of the National Environmental Policy Act of 1969 (NEPA). Assistance was provided by the Pacific Northwest Laboratory under the direction of the staff; the contributors to the draft supplement are listed in Appendix B. This draft supplement addresses potential environmental impacts associated with the licensee's proposal to place the TMI-2 facility in a post-defueling monitored storage mode followed by cleanup and with alternatives to the licensee's proposal.

Information for the draft supplement was obtained from the licensee's Environmental Report and Final Safety Analysis Report (Metropolitan Edison Co. and Jersey Central Power and Light Co. 1974), from the staff's Final Environmental Statement of the operating license (NRC 1976), from the staff's PEIS of March 1981 (NRC 1981), from Supplement 1 of October 1984 (NRC 1984), from Supplement 2 of June 1987 (NRC 1987), and from new information provided by the licensee or independently developed by the staff. The staff met with the licensee to discuss items of information provided, to seek new information from the licensee that might be needed for an adequate assessment, and to ensure that the staff had a thorough understanding of the proposed action. In addition, the staff sought information from other sources that would assist in the evaluation, and visited and inspected the project site and vicinity.

On the basis of the foregoing, the staff made an independent evaluation of alternatives for completing cleanup of the facility following defueling, including the licensee's proposed alternative, and prepared this draft supplement to the PEIS. This draft supplement is being circulated to Federal, State, and local government agencies and to interested members of the public for comment. A summary notice of the availability of this draft supplement is being published concurrently in the Federal Register. The information on which the supplement is based is being made available to the public.(a) Interested persons are invited to comment on the draft supplement.

The following Federal and State agencies are being asked to comment on this draft supplement to the PEIS:

U.S. Army Corps of Engineers
U.S. Environmental Protection Agency
U.S. Department of Agriculture
U.S. Department of Energy
U.S. Department of Health and Human Services
U.S. Department of Labor

(a) NRC Public Document Room, 1717 H Street, Washington, DC 20555, and the State Library of Pennsylvania, Government Publications Section, Education Building, Commonwealth and Walnut Streets, Harrisburg, PA 17126.

U.S. Department of Interior
U.S. Department of Transportation
U.S. Federal Emergency Management Agency
U.S. Federal Energy Regulatory Commission
U.S. National Oceanic and Atmospheric Administration
U.S. Nuclear Regulatory Commission, Advisory Panel for the Decontamination of TMI Unit 2
Maryland Department of Health and Mental Hygiene
Maryland Department of Natural Resources
Maryland Department of State Planning
New Jersey Department of Environmental Protection
Pennsylvania Department of Environmental Resources
Pennsylvania Department of Health
Pennsylvania Department of Labor and Industry
Pennsylvania Department of Public Welfare
Pennsylvania Intergovernmental Council

After receipt and consideration of comments on the draft supplement, the staff will prepare a final supplement to the PEIS, which will include a discussion of comments on the draft supplement and the responses to them.

Single copies of this supplement may be obtained by writing the Director, Division of Publication Services, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Comments on the supplement should be addressed to:

Dr. Michael T. Masnik
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. Masnik is the Project Manager for this project. He may be reached at the above address or at (301) 492-1373.

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NOMENCLATURE

accident-generated water - On February 27, 1980, an agreement executed among the City of Lancaster, Pennsylvania, Metropolitan Edison Company, and the NRC defined "accident-generated water" as:

"
- Water that existed in the TMI-2 auxiliary, fuel handling, and containment buildings including the primary system as of October 16, 1979, with the exception of water which as a result of decontamination operations becomes commingled with nonaccident-generated water such that the commingled water has a tritium content of 0.025 μCi/mL or less before processing.
- Water that has a total activity of greater than 1 μCi/mL prior to processing except where such water is originally nonaccident water and becomes contaminated by use in cleanup.
- Water that contains greater than 0.025 μCi/mL of tritium before processing."

actinides - the group of radioactive elements with atomic numbers 90 and above, including thorium, protactinium, uranium, neptunium, plutonium, americium, and curium.

activation products - radioactive materials that are created when stable substances are bombarded by neutrons. For example, cobalt-60 is formed from the neutron bombardment of the stable isotope cobalt-59.

Agreement States - States that have agreed to accept the responsibility of enforcing the provisions of Federal legislation for activity within their borders. The Commonwealth of Pennsylvania is an Agreement State with respect to the Clean Water Act, but not the Atomic Energy Act.

ALARA - an acronym for "as low as reasonably achievable." A concept of radiation protection that specifies that radiation exposure of personnel and radioactive discharges from nuclear facilities be kept as far below regulatory limits as reasonably achievable.

alpha radiation - an emission of particles (helium nuclei) from a material undergoing nuclear transformation; the particles have a nuclear mass number of four and a charge of plus two.

ambient radiation - surrounding radiation from multiple or distributed sources.

anadromous fish - fish that ascend freshwater streams from the sea to spawn.

attocurie - 1×10^{-18} curies, a unit for measuring radioactivity.

auxiliary and fuel-handling building (AFHB) - a building located at the TMI-2 facility. It is divided into two sections that are separated by a common wall. The auxiliary section contains tank, pumps, piping, and other equipment to process and store water for the reactor coolant system and to treat radioactive wastes. The fuel-handling section contains large basins, or pools, for the storage of spent fuel.

background radiation - the level of radiation in an area which is produced by sources of radiation (mostly natural) other than the one of specific interest. Examples of such radiation sources are cosmic radiation and radioactive elements in the atmosphere, building materials, the human body, and the crust of the earth. In the Harrisburg area, the background radiation level is about 87 mrem/yr, not including any contribution from medical practice.

BEIR - Biological Effects of Ionizing Radiation. A set of reports by the National Academy of Sciences, Advisory Committee on the Biological Effects of Ionizing Radiation. (See also References, Section 7.0.)

benthic - dwelling on the bottom of a body of water.

beta particles - an electron or a positron (a particle with the same mass as an electron but with a positive charge rather than a negative one). Beta particles are commonly emitted from the nuclei of atoms undergoing nuclear transformation. Also referred to as beta radiation.

beta radiation - radiation consisting of beta particles.

biota - plant and animal life.

CFR - Code of Federal Regulations.

Ci - see curie.

collective 50-year dose commitment - the total radiation received by a population or group of individuals from an initial exposure through the succeeding 50 years. The collective 50-year dose commitment is expressed in person-rem. (See person-rem.)

cumulative occupational dose - the total radiation dose to workers. It is determined by summing the product of the dose rate and the length of time the worker is exposed to the dose rate for all dose rates and all workers. The cumulative occupational dose is expressed in person-rem. (See person-rem.)

curie (Ci) - the special unit of activity. Activity is defined as the number of nuclear transformations occurring in a given quantity of material per unit of time. One curie of activity is 37 billion transformations per second.

decay products - the nuclides formed by the radioactive disintegration of a first nuclide (parent). Also called daughter products.

decommissioning - removing nuclear facilities safely from service and reducing residual radioactivity to a level that permits release of the property for unrestricted use and termination of license.

DECON - the decommissioning alternative in which equipment, structures and portions of a facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after cessation of operations.

defueling - the licensee's term for removal of more than 99 percent of the fuel from the TMI-2 facility.

delayed cleanup - the NRC staff's term for the licensee's (GPU Nuclear's) proposal to maintain the TMI-2 facility in a post-defueling monitored storage (PDMS) mode for an unspecified period of time after more than 99 percent of the fuel has been removed from the facility. After the storage period, the cleanup process would be resumed. (See PDMS.)

demineralizer systems - processing systems in which synthetic ion exchange materials are used to remove impurities from water.

DOE - U.S. Department of Energy.

dose - a general term indicating the amount of energy absorbed from incident radiation by a unit mass of any material.

dose commitment - the integrated dose to an organism that results unavoidably from the intake of radioactive material starting at the time of intake and continuing (at a decreasing dose rate) to a later time (usually specified to be 50 years from intake).

dose rate - the dose (amount of energy absorbed by a unit mass) received per unit of time.

DOT - U.S. Department of Transportation.

emergency allocation - allocation of waste disposal volume by the DOE in commercial LLW burial sites because of unusual circumstances.

ENTOMB - the decommissioning alternative in which radioactive contaminants are encased in a structurally long-lived material, such as concrete. The entombed structure is appropriately maintained and continued surveillance is carried out until the radioactivity decays to a level permitting release for unrestricted use of the property.

EPA - U.S. Environmental Protection Agency.

EPICOR II - a filtration and demineralizer system designed to process some of the liquid radioactive waste resulting from the TMI accident. The system can be used on liquid waste containing up to 100 microcuries of radioactivity per milliliter of water.

ERDA - U.S. Energy Research and Development Administration, predecessor to the DOE.

etiology - the cause of disease or disorder as determined by medical diagnosis.

exposure - the condition of being made subject to the action of radiation; also, a measure of the ionization produced in air by x-ray or gamma radiation.

50-year dose commitment - the total radiation received from initial exposure through the succeeding 50 years.

fission - the spontaneous or induced disintegration of a heavy atom into two or more lighter atoms with an accompanying loss of mass that is converted into energy.

fission products - the nuclides formed by the division of a heavier nucleus, typically in a nuclear reactor. Isotopes of essentially all elements are produced by fission of fissile materials. Fission products are the main radioactive components of high-level radioactive wastes.

gal/min - gallons per minute.

gamma radiation - electromagnetic radiation of high energy (and short wavelength), emitted by nuclei undergoing internal changes. Gamma radiation has the highest energy and shortest wavelength in the electromagnetic spectrum and is capable of penetrating several inches of a solid such as concrete.

genetic effects of radiation - effects of radiation that alter the hereditary material and may therefore affect subsequent unexposed generations.

GPU Nuclear Corporation - the licensee at TMI-2, a subsidiary of General Public Utilities Corporation.

groundwater - water that exists or flows below the ground's surface (within the zone of saturation).

h - hour.

half-life - the time required for half of a given radioactive substance to decay.

Hanford Nuclear Reservation - a nuclear facility near Richland, Washington, that is operated by the DOE.

hectare - a metric unit of measure equal to 2.47 acres.

HEPA filter - high efficiency particulate air filter.

immediate cleanup - the major NRC staff-identified alternative to the licensee's proposal of delayed cleanup. Immediate cleanup involves the continuation and completion of the cleanup at the present level of effort without a storage period.

ion - an atom or molecule from which an electron has been removed (a positively charged ion) or to which an electron has become attached (a negatively charged ion).

ion exchange - in this document, a process for selectively removing a constituent from a waste stream by reversibly transferring ions from a liquid to an insoluble solid (the ion exchange media).

ion exchange media - resins or zeolite materials used in ion exchange processes.

ionization - the process by which a neutral atom or molecule acquires a positive or a negative charge by removal or attachment of an electron.

ionizing radiation - any form of radiation that generates ions in the irradiated material.

isotopes - nuclides with the same atomic number but with different atomic masses, therefore having the same chemical properties but different physical properties.

kg - kilogram.

L - liter.

licensee - the holder of a license issued by the NRC to possess or use radioactive materials. In the case of TMI-2, the license is held by GPU Nuclear Corporation.

LLD - lower limit of detection.

LLW - low-level waste; all radioactive waste materials that are not high-level or transuranic waste. Most TMI-2 wastes are of this type.

L/min - liters per minute.

maximally exposed individual - the hypothetical person who would receive the greatest possible radiation dose from a specific release. For atmospheric releases, this individual is assumed to breathe air at that offsite boundary location with the highest airborne concentration and to consume food products raised exclusively in that offsite boundary location receiving the maximum ground deposition of released radioactive material. For liquid releases, this individual is assumed to consume large quantities of river water and fish and to participate frequently in rivershore activities. In this supplement, the maximally exposed individual is also assumed to eat large quantities of Chesapeake Bay shellfish.

MCi - megacurie (one million curies); a unit for measuring radioactivity.

Memorandum of Understanding - an agreement between the NRC and DOE, whereby the DOE will accept certain categories of waste from the cleanup of TMI-2, for permanent disposal, either with- out cost or on a cost-reimbursement basis. (Memorandum of Understanding Between the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy, Concerning the Removal and Disposal of Solid Nuclear Wastes from Cleanup of the Three Mile Island Unit 2 Nuclear Plant, March 15, 1982.)

μCi - microcurie (1 x 10^{-6} curies, or one-millionth of a curie); a unit for measuring radioactivity.

μg - microgram (1 x 10^{-6} grams, or one-millionth of a gram); a unit for measuring weight.

mg/L - milligrams per liter.

mL - milliliter.

MPC - maximum permissible concentration, the NRC-prescribed concentration limit for radioactive materials (10 CFR 20, Appendix B). The MPCs are expressed as average radionuclide concentrations in air or water. Different MPC values apply to the public and to radiation workers.

mR - milliroentgen [1 x 10^{-3} roentgen, or one-thousandth (1/1000th) of a roentgen]; a unit for measuring radioactivity.

mrem - millirem [1 x 10^{-3} rem or one-thousandth (1/1000th) of a rem]; a unit of measuring radioactivity.

MSL - mean sea level.

MWHT - miscellaneous waste holdup tank, located in the AFHB.

NAS - National Academy of Sciences.

nCi - nanocurie [1 x 10^{-9} curies, or one-billionth (1/1,000,000,000) of a curie]; a unit for measuring radioactivity.

NCRP - National Council on Radiation Protection and Measurement.

NEPA - National Environmental Policy Act of 1969.

neutron - an uncharged elementary particle found in the nucleus of every atom except hydrogen.

neutron capture - the process in which an atomic nucleus absorbs or captures a neutron.

no-action alternative - an alternative to the proposed action, which is required by the National Environmental Policy Act (NEPA) to be considered as part of all environmental impact statements. The no-action alternative for the period addressed by this supplement implies no action to complete the cleanup following the completion of defueling.

NPDES - National Pollutant Discharge Elimination System.

NRC - U.S. Nuclear Regulatory Commission.

nuclide - a species of atom having a specific mass, atomic number, and nuclear energy state.

occupational radiation exposure - the radiation exposure to which workers at a nuclear facility are subjected during the course of their work.

ORNL - Oak Ridge National Laboratory.

PaDER - Commonwealth of Pennsylvania, Department of Environmental Resources.

pCi - picocurie [1 x 10^{-12} curies, or one-trillionth (1/1,000,000,000,000) of a curie]; a unit for measuring radioactivity.

pCi/L - picocuries per liter.

PEIS - Final Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Waste Resulting from March 28, 1979 Accident Three Mile Island Station, Unit 2, NUREG-0683, 1981.

penetration factor - the fraction of the particulates that would pass through a high efficiency particulate air (HEPA) filter.

person-rem - the sum of the individual radiation doses (collective dose) received by members of a certain group or population. It may be calculated by multiplying the average dose per person by the number of persons. For example, a thousand persons, each exposed to one millirem (1/1000 rem), would have a collective dose of 1 person-rem.

photon - a quantity of energy emitted in the form of electromagnetic radiation. Gamma rays and x-rays are examples of photons.

population dose - the summation of individual radiation doses received by all those exposed to the radiation source or event being considered, and expressed as person-rem. The same as collective dose.

post-defueling monitored storage (PDMS) - the licensee's term for monitored storage of the TMI-2 facility following defueling (removal of more than 99 percent of the fuel from the TMI-2 facility). Monitored storage refers to the inspection, surveillance, and maintenance of the facility during the storage period. (See storage.)

ppm - parts per million.

primary system - see reactor coolant system.

PWR - pressurized water reactor. The TMI-2 reactor is of this type.

rad - a unit of absorbed dose of ionizing radiation.

radiation - energy in the form of electromagnetic rays (radiowaves, light, x-rays, gamma rays) or particles (electrons, neutrons, helium nuclei) sent out through space from atoms, molecules, or atomic nuclei as they undergo internal change. It may also result from particle and electromagnetic radiation interactions with matter.

radioactive contamination - radioactive material located in areas where it is not wanted.

radioactive decay - the spontaneous natural process by which an unstable radioactive nucleus releases energy or particles.

radioactivity - product of radioactive decay of an unstable atom.

radioisotopes - radioactive isotopes. (See also radionuclide and isotopes.)

radionuclide - an unstable nuclide that undergoes radioactive decay.

RCRA - Resource Conservation and Recovery Act.

reactor building - a containment building that houses the reactor vessel.

reactor coolant system - consisting of the reactor, the steam generators, the reactor coolant pumps, and the connecting piping. In an operating reactor, the heat produced by the reactor is transferred to the water coolant in the reactor vessel. The hot water is circulated through the steam generator tubes to produce steam. The reactor coolant pump is used to circulate the water coolant. The reactor coolant system is also called the primary coolant system or primary system.

rem - a unit of radiation dose equivalent that is proportional to the risk of biological injury.

resin liners - cylindrical metal containers used for the ion exchange media (resins and/or zeolites) during purification of contaminated water by ion exchange processes.

resins - solid or semisolid products of synthetic origin used in ion exchange processes for purification of liquids.

resuspension factor - the ratio of the amount of radioactive material in the air to the amount of loose radioactive material on a surface.

roentgen (R) - unit of exposure (gamma or x-ray) in air. (One roentgen equals 2.58×10^{-4} coulombs per kilogram of air.)

SAFSTOR - the decommissioning alternative in which the nuclear facility is placed and maintained in such condition that it can be safely stored, monitored, and subsequently decontaminated to levels that permit release for unrestricted use.

scabbling - an aggressive decontamination technique that removes concrete surface coatings with toothed pistons or a rotating drum.

SDS - submerged demineralizer system; a water-treatment system that uses a synthetic zeolite mineral as the ion exchange medium to remove radioactive cesium from the accident-generated water.

shielding - a barrier of solid or liquid material (e.g., lead, concrete, or water) that reduces the intensity of radiation passing through it. Shielding can be used to protect personnel from the damaging effects of ionizing radiation.

somatic effects of radiation - effects of radiation limited to the exposed individual, as distinguished from genetic effects, which may also affect subsequent unexposed generations. Somatic effects include cancers of various types.

source term - the list of radionuclides and the quantity of each radionuclide that is assumed to be present in a given mixture.

specific activity - quantity of radioactivity per unit mass, usually in picocuries per gram.

storage - for the purposes of this supplement, storage is defined as the placement of the TMI-2 facility into a passive monitored state for some unspecified time period before completion of the cleanup.

Supplement 1 - the first supplement to the PEIS [Final Supplement Dealing with Occupational Radiation Dose (NRC 1984)].

Supplement 2 - the second supplement to the PEIS [Final Supplement Dealing with Disposal of Accident-Generated Water (NRC 1987)].

technical specifications - limits and requirements that are set forth in the facility license.

TMI - Three Mile Island.

TMI-1 - Three Mile Island Unit 1; the NRC-licensed reactor operating on the TMI site.

TMI-2 - Three Mile Island Unit 2; the accident-damaged reactor undergoing cleanup on the TMI site.

total body dose - the radiation dose to the total body, including the bones and all organs, from both external and internal radionuclides.

transuranics - elements having atomic numbers higher than that of uranium (92), including neptunium, plutonium, americium, and curium.

tritiated water - water in which one or both hydrogen atoms have been replaced by a tritium atom.

tritium - a radioactive isotope of hydrogen, containing two neutrons. The nonradioactive form of hydrogen has no neutrons. The half-life of tritium is 12.5 years.

unrestricted use - use of any area or facility without restriction because of prior contamination.

UNSCEAR - United Nations Scientific Committee on the Effects of Atomic Radiation.

U.S. Ecology - the operator of a commercial LLW burial site near Richland, Washington.

volume reduction factor - the ratio of the remaining volume over the initial volume.

water table gradient - the ratio of change in water table elevation over distance.

yr - year.

zeolites - any of various natural or synthesized silicate minerals used to purify water.

1.0 INTRODUCTION

In March 1981, the U.S. Nuclear Regulatory Commission (NRC) published the final Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Waste Resulting from March 28, 1979 Accident Three Mile Island Nuclear Station, Unit 2 (NRC 1981), referred to in this document as the PEIS.

The PEIS was intended to provide an overall evaluation of the environmental impacts that could result from cleanup activities at Three Mile Island Unit 2 (TMI-2), from the stabilization of plant conditions after the accident through the completion of cleanup, based on the information then available. The cleanup plan evaluated in the PEIS called for four fundamental activities: building and equipment decontamination; fuel removal and decontamination of the reactor coolant system; treatment of radioactive liquids; and packaging, handling, shipment, and disposal of radioactive wastes. Following the publication of the PEIS, the Commission issued a Policy Statement on April 28, 1981, indicating that the NRC staff would evaluate and act on major cleanup proposals as long as the impacts associated with the proposed activities fell within the scope of the impacts already assessed in the PEIS. Throughout the cleanup, the NRC staff has reviewed the licensee's proposed major cleanup activities to ensure that the activities are safe and that potential environmental impacts are within the range of impacts given in the PEIS.

The PEIS has been supplemented twice (NRC 1984; NRC 1987) since its publication. Supplement 1 (NRC 1984) reevaluated the occupational dose estimates given in the 1981 PEIS because new information led the NRC staff to conclude that cleanup could result in greater occupational radiation exposure than was originally estimated. Supplement 2 (NRC 1987) updated the information presented in the PEIS regarding options for disposal of the water contaminated as a result of the accident (accident-generated water) and the environmental impacts that could result from disposal.

This document is a draft of the third supplement to the PEIS; its purpose is to address the environmental impacts associated with a proposal from the licensee, GPU Nuclear Corporation (GPU), to maintain the TMI-2 facility in a post-defueling monitored storage (PDMS) mode.

Before entering PDMS, more than 99 percent of the fuel will have been removed from the reactor. This point in time has been designated by the licensee and is referred to in this supplement as the "end of defueling." In addition, limited reactor coolant system decontamination will have been conducted, treatment of radioactive liquids would be completed and disposal would likely be either completed or underway and packaging and shipping much of the radioactive wastes from the site would be completed. If compared to the four fundamental activities identified in the PEIS, only building and equipment decontamination would not be either substantially or actually completed. Of the buildings contaminated by the accident, only the reactor building would have general area radiation levels higher than those of an operating plant nearing the end of life.

After a storage period of an unspecified length of time, the licensee plans to resume the cleanup process. This proposal of a PDMS period followed by cleanup is referred to in this document as "delayed cleanup." The licensee has proposed placing the TMI-2 facility into monitored storage because of the benefits of occupational dose savings resulting from radioactive decay during the storage period as well as anticipated advances in decontamination technology. Further reduction in occupational dose could be achieved through use of advanced robotic technology and waste treatment methods.

In accordance with the National Environmental Policy Act (NEPA), this supplement considers alternative actions to the licensee's proposal. The major alternative evaluated is immediate cleanup, which is similar to the "current cleanup plan" described in Supplement 1. The no-action alternative as required by NEPA is also evaluated for the period addressed by this supplement, from completion of defueling to completion of cleanup. Although other alternatives are considered, delayed cleanup as proposed by the licensee and immediate cleanup serve to bound the environmental impacts that could reasonably be expected from cleanup of the TMI-2 facility.

Although the licensee's proposal is to place the TMI-2 facility in a storage mode, no decisions have been made regarding the final disposition of the TMI-2 facility, whether it is to be decommissioned or refurbished. Therefore, this supplement evaluates the environmental impacts of delayed cleanup and immediate cleanup only to the end of cleanup, as discussed above.

Because this document, like the impact statement it supplements, is programmatic in nature, it is not intended to provide a step-by-step work plan. However, the most probable sequences and methods for cleanup have been assumed in order to predict the resulting environmental impacts. The best available information has been used in this analysis. Where there are uncertainties, conservative assumptions have been made and documented in the text and appendixes as appropriate.

Background information potentially affecting the cleanup is presented in Section 2.0 of this supplement. This information includes cleanup progress and current conditions in the reactor building and the AFHB, radiation source characteristics, and regulatory and administrative considerations. In Section 3.0, the licensee's proposal for delayed cleanup and the alternatives to this proposal are described. Alternatives that were considered but not quantitatively evaluated, including the no-action alternative, are discussed. Delayed cleanup and immediate cleanup are described in detail and the potential environmental impacts of each are quantitatively evaluated. The potential environmental impacts include radiation exposure to the offsite population from routine and accidental releases, occupational radiation dose, waste management impacts, transportation impacts, socioeconomic impacts, commitment of resources, and regulatory considerations. Section 4.0 describes the affected environment. Section 5.0 summarizes and compares the environmental impacts for the evaluated alternatives and discusses the potential for human health effects. The staff's conclusions are presented in Section 6.0, references are listed in Section 7.0, and the index is provided in Section 8.0. Appendix A is reserved for comments on

this draft Supplement 3. Other appendixes list contributors and provide additional details on estimates of impacts.

2.0 BACKGROUND INFORMATION AFFECTING CLEANUP ALTERNATIVES

Section 2.1 summarizes the cleanup progress to the end of 1987, describes the conditions that will exist in the reactor building and the auxiliary and fuel-handling building (AFHB) at the end of defueling, and describes the work required to complete cleanup. Section 2.2 estimates the amount of the radioactive material that will be present in the facility at the end of defueling. The regulatory and administrative considerations affecting the cleanup after defueling is completed are addressed in Section 2.3.

2.1 CLEANUP PROGRESS AND CURRENT CONDITIONS

The 1979 accident at the TMI-2 facility involved a loss of reactor system coolant and resulted in serious damage to the reactor fuel. When coolant was restored, radioactive contamination in the form of fuel debris and fission products was distributed by the cooling water throughout the reactor coolant system. A portion of the water, carrying fuel debris and fission products as dissolved and particulate material, escaped from the reactor coolant system and flowed into the reactor building basement. (A discussion of the radionuclides transported in the water is contained in Section 2.2.) Exposed surfaces in the reactor building and AFHB were contaminated with material in the reactor coolant and from radionuclides that became airborne as steam escaping from the reactor coolant system condensed during and shortly after the accident. After the accident, the water in the basement was heated by residual heat from the reactor vessel, evaporated, condensed on the walls, and drained down onto the floors and back into the basement. This period of evaporation and condensation contributed to the permeation of radionuclides into porous surfaces, such as concrete, and the incorporation of radionuclides into corrosion layers as iron surfaces rusted. A more detailed account of the accident is contained in an NRC report (NRC 1979a), Kemeny et al. (1979), and Rogovin and Frampton (1980).

The PEIS and previous supplements have evaluated the impact of activities necessary to reach the "completion of cleanup." The completion of cleanup will be achieved when four fundamental activities have been completed: (1) building and equipment decontamination to levels typical of an operating reactor nearing the end of its life, (2) fuel removal and decontamination of the reactor coolant system, (3) treatment of radioactive liquids, and (4) packaging, handling, shipment, and disposal of radioactive wastes. After the completion of cleanup, the facility would be decommissioned or refurbished.

The PEIS indicated that the radiation dose rates at the completion of cleanup would approach 10 mrem/h in most areas of the reactor building and AFHB. This is typical of normally occupied areas in a relatively clean reactor facility at the end of its useful life. The primary difference between an undamaged reactor at the end of its useful life and the licensee's PDMS proposal is that during PDMS relatively high levels of contamination

would remain in the reactor building basement and a small amount of residual fuel would remain in the reactor coolant system during storage.

Within certain cubicles, shielded areas, and other infrequently occupied areas, radiation dose rates may be considerably higher both in undamaged facilities and in the TMI-2 facility. Radiation levels may be lowered in one of two ways: radiation sources may be shielded, or they may be removed. Both portable shielding and radionuclide removal have been used in TMI-2 cleanup. Shielding, however, is a temporary measure to minimize dose to the workers. The radiation sources must ultimately be removed. In assessing the measures necessary to complete cleanup, the NRC staff has assumed that dose rates, in the absence of portable shielding, would need to be comparable to those of an undamaged reactor nearing the end of its useful life.

Although radiation levels at the completion of cleanup would be comparable to those of an undamaged reactor, the mix of radionuclides that contributes to the radiation levels in TMI-2 will differ substantially from the mix in an undamaged reactor. In most reactors, radiation levels are primarily due to cobalt-60 and other activation products. The radiation levels in the TMI-2 reactor are primarily due to cesium-137, a fission product.

A description of the cleanup progress to date, the conditions that will exist at the end of defueling, and the work required to complete cleanup is given for four major areas: (1) the reactor building, (2) the reactor vessel, (3) the reactor coolant system, and (4) the AFHB.

2.1.1 Reactor Building Cleanup

The reactor containment building is uniquely designed and constructed to maintain its structural integrity (with almost no leakage) during a wide variety of accidents. The entire building is constructed of reinforced concrete lined with welded steel. The liner is painted with a corrosion-resistant paint to the level of the basement floor. The bottom of the building is covered with approximately 2 feet (0.6 meters) of poured concrete to form the floor of the reactor building basement. Piping and electrical system penetrations that enter the building are sealed to maintain their integrity through a variety of accident conditions.

The building is equipped with a two-train ventilation system, both trains having double-stage high-efficiency particulate air (HEPA) filters. These filters remove particulate material but allow gases to pass through.

A plan view of the reactor building is given in Figure 2.1. The three levels within the building are referred to by elevation above sea level: the 305-foot elevation (entry level), the 347-foot elevation (operating floor), and 282-foot elevation (referred to as the basement). Decontamination work to date has significantly reduced radiation fields in the reactor building. The emphasis during cleanup has been on removing debris, decontaminating, and shielding frequently traveled and frequently occupied areas. The specific conditions at each elevation are discussed separately in the following paragraphs.

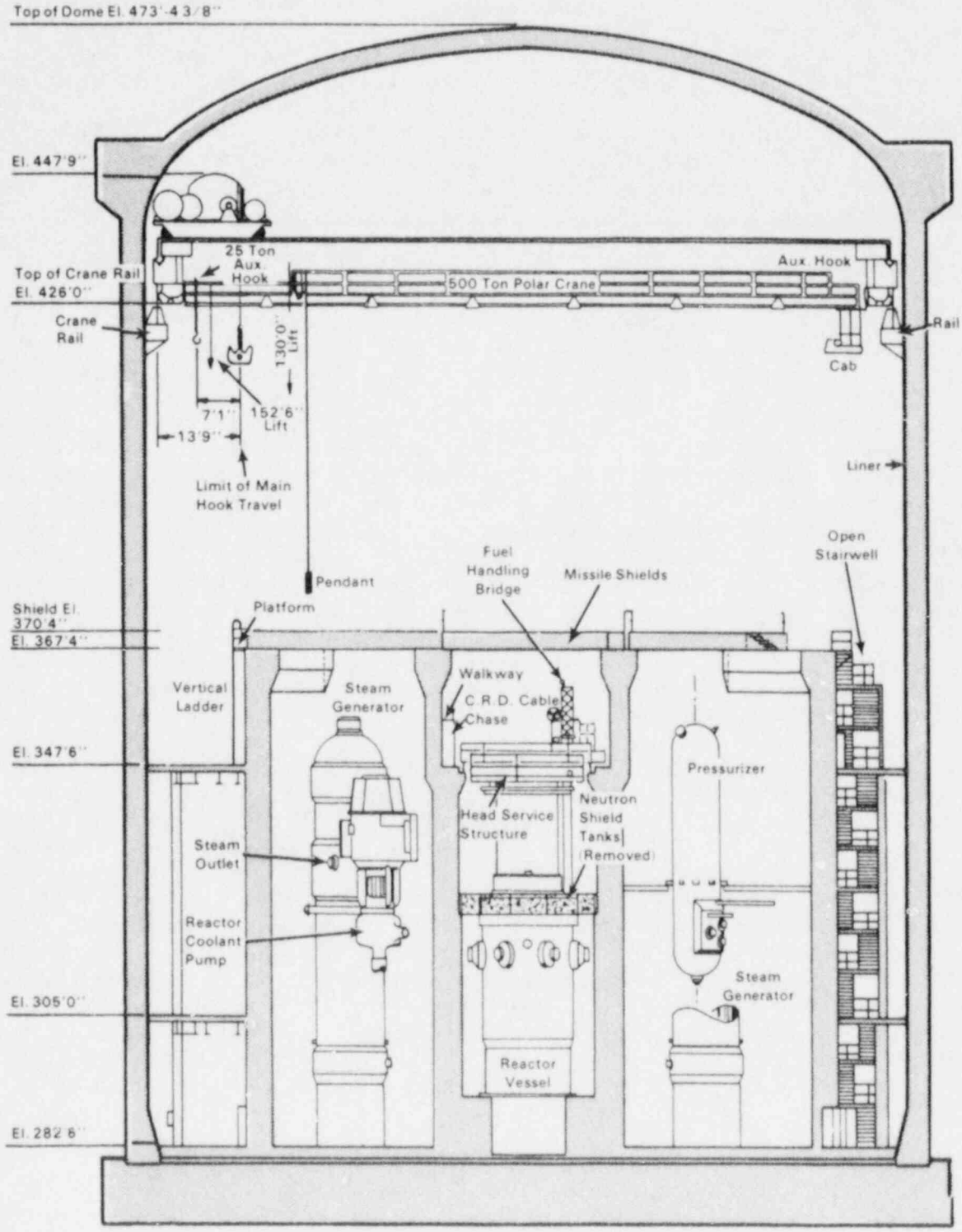

FIGURE 2.1. Reactor Building Cross Section

The building is entered at the 305-foot elevation (Figure 2.2). When the building was first entered after the accident, the radiation dose rates at this elevation averaged 430 mrem/man-h. Removal of debris, decontamination, placement of shielding, and the removal of the surface layer from floors and walls (scabbling) had reduced the general area exposure rates at this level to an average of about 60 to 70 mR/h by the end of 1987. Decontamination using high- and low-pressure sprays of borated water appears to have reduced the amount of contamination on equipment and building surfaces. Effective, but temporary, dose rate reductions also have been achieved by placing shielding around some sources of high-level radiation, including the air coolers, elevator shaft, both stairwells, and some floor drains. Scabbling, an aggressive decontamination technique that removes concrete surface coatings with toothed pistons or a rotating drum, has removed additional contamination and reduced the general area dose rates. A large portion of the 305-foot elevation has been scabbled and the remaining rough surfaces sealed by applying an epoxy sealant to prevent recontamination of the concrete. Figure 2.3 shows the general area exposure rates (gamma radiation) at the end of 1987. Reducing the dose rates below the current level is expected to require greater effort than that expended so far because the remaining radiation sources are difficult to remove and/or are in relatively inaccessible locations. Permanent dose-reduction techniques will be necessary for the shielded structures (such as the air coolers and floor drains) before the completion of cleanup. Electrical cables and trays, piping supports, and overheads will also need to be decontaminated or removed to complete cleanup.

The 347-foot elevation (Figure 2.4) is the operating floor, formerly reached by an open stairway, an enclosed stairwell, and an elevator. Radiation dose rates resulting from the accident have prevented the refurbishment of the elevator and minimized use of the enclosed stairwell. A temporary stairway allows access to a portion of the enclosed stairwell. Shielding has been placed within the stairwell, reducing the dose rates. The reactor vessel defueling platform is accessed from the 347-foot elevation. Dose rates at the 347-foot elevation averaged 240 mrem/man-h following the accident. Essentially all the concrete floors at the 347-foot elevation have been scabbled and sealed. Shielding, removing debris, decontaminating, and scabbling reduced the general area exposure rates to approximately 25 mR/h by the end of 1987, with less than 20 mR/h for most well-traveled areas and approximately 10 mR/h on the defueling platform. A map of the general area exposure rates (gamma radiation) at the end of 1987 is shown in Figure 2.5. To complete the cleanup, permanent dose-reduction techniques (such as decontamination or removal) are expected to be necessary for currently shielded structures, electrical cable trays, piping supports, and other overhead components.

The polar crane located at the 426-foot elevation is reached by ladder or hoist from the 347-foot elevation. The elevation of the crane's cab is 418 feet, 6 inches. The polar crane, which is shown in Figure 2.1, was used to prepare for defueling and continues to be used to transport decontamination equipment, radioactive waste, and shielding materials within the reactor building. Dose rates at initial access to the polar crane after the accident

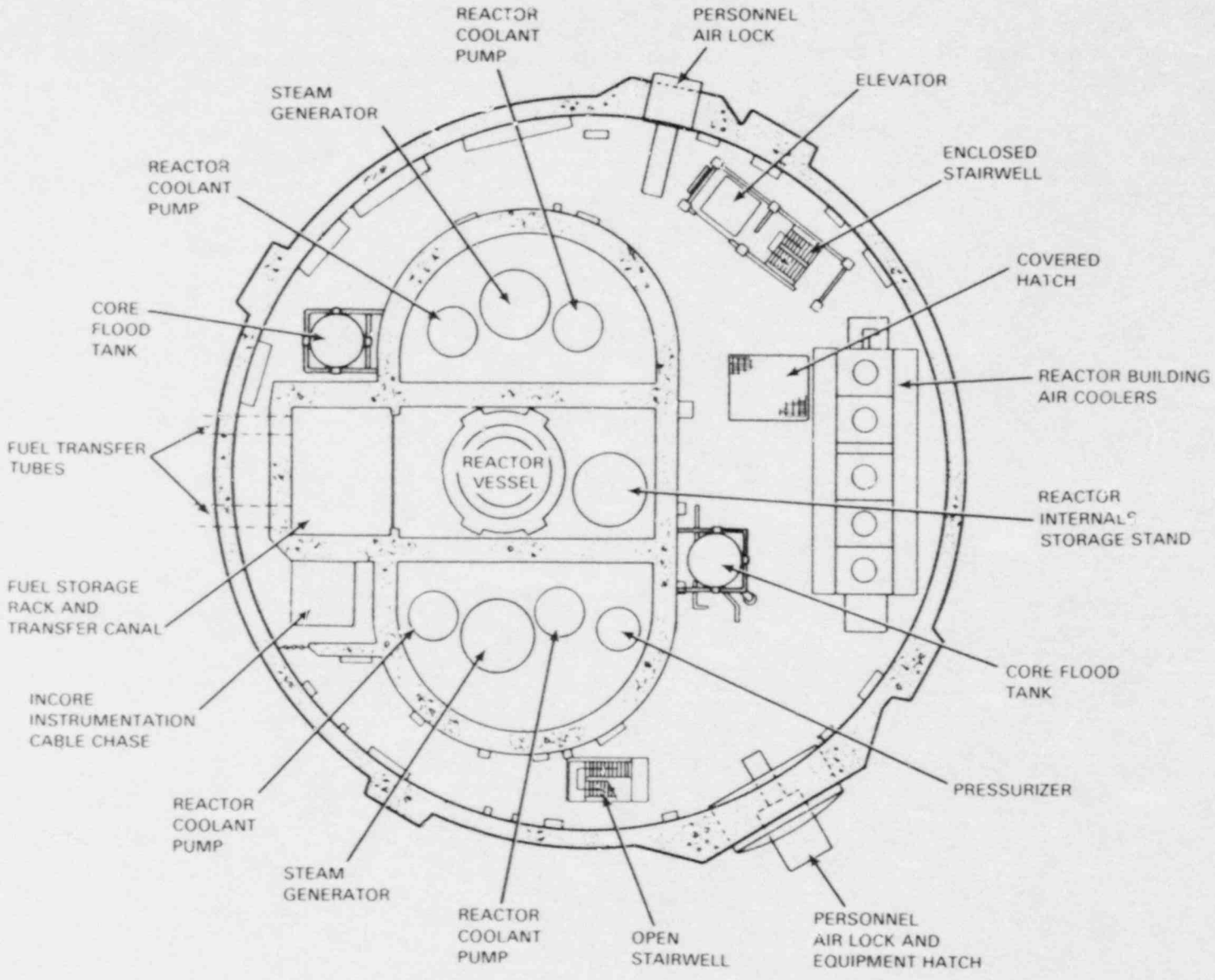

FIGURE 2.2. 305-Foot Elevation of Reactor Building

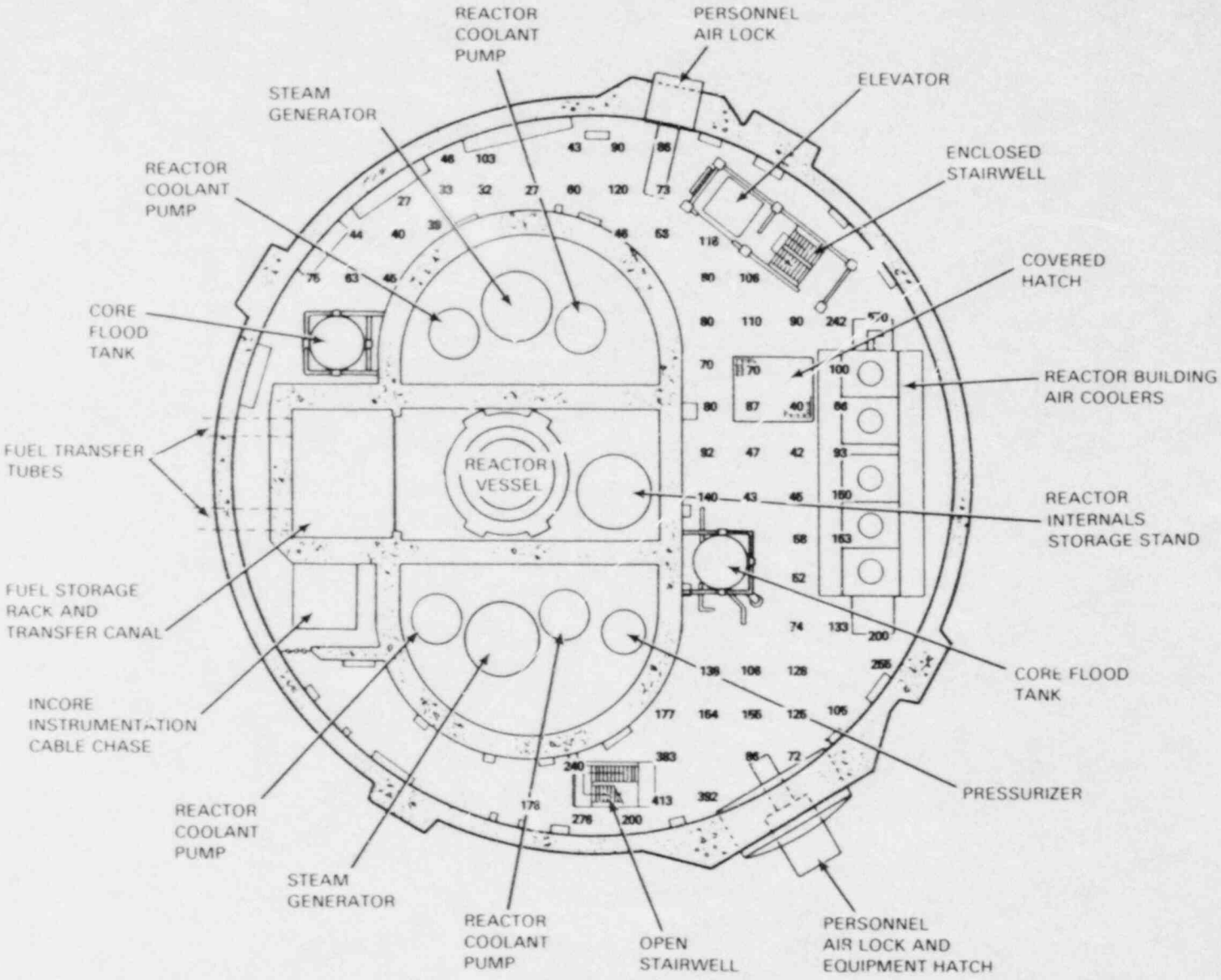

FIGURE 2.3. Map of the General Area Exposure Rates on the 305-Foot Elevation at the End of 1987, in mR/h for Gamma Radiation

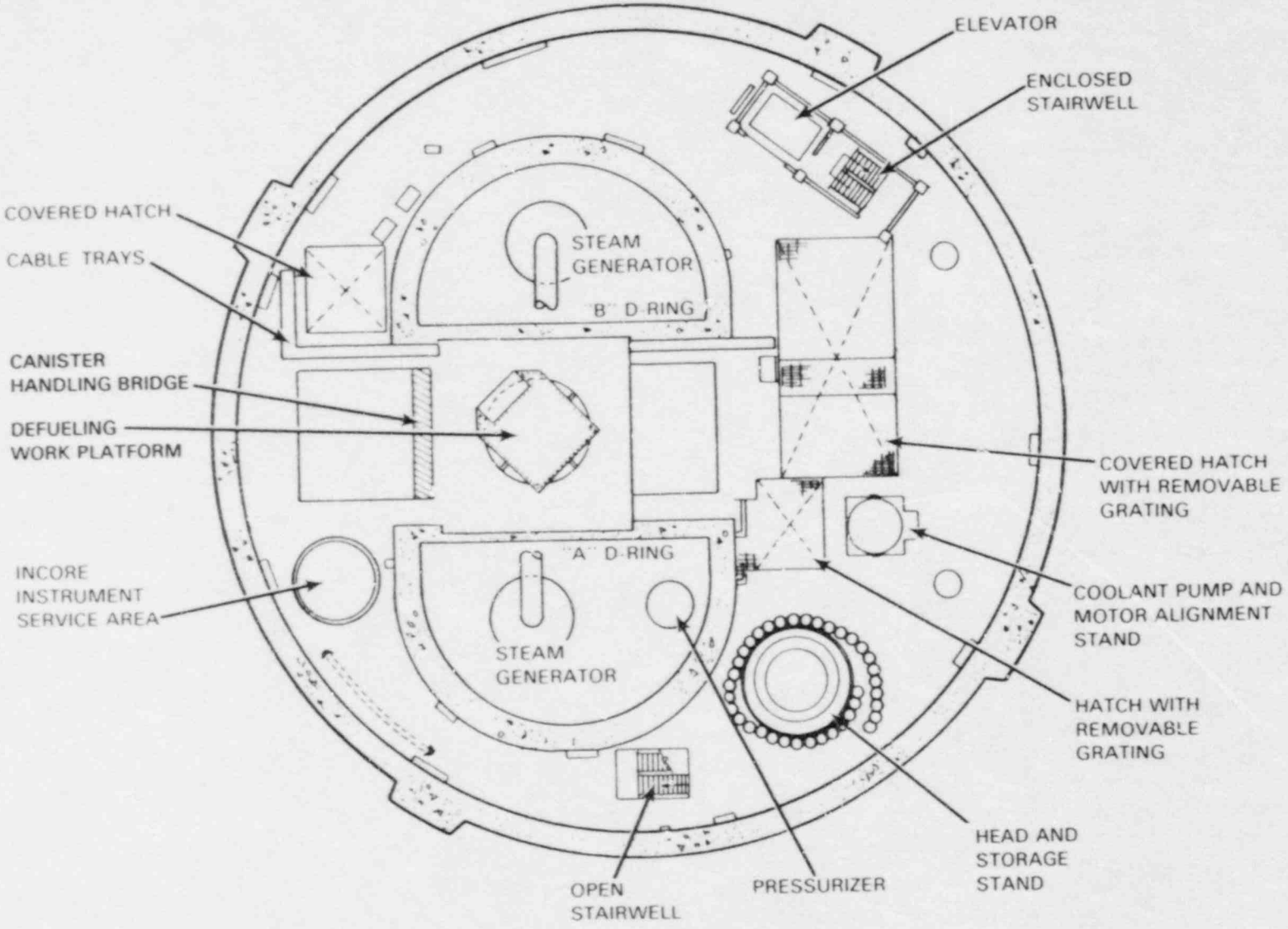

FIGURE 2.4. 347-Foot Elevation of Reactor Building

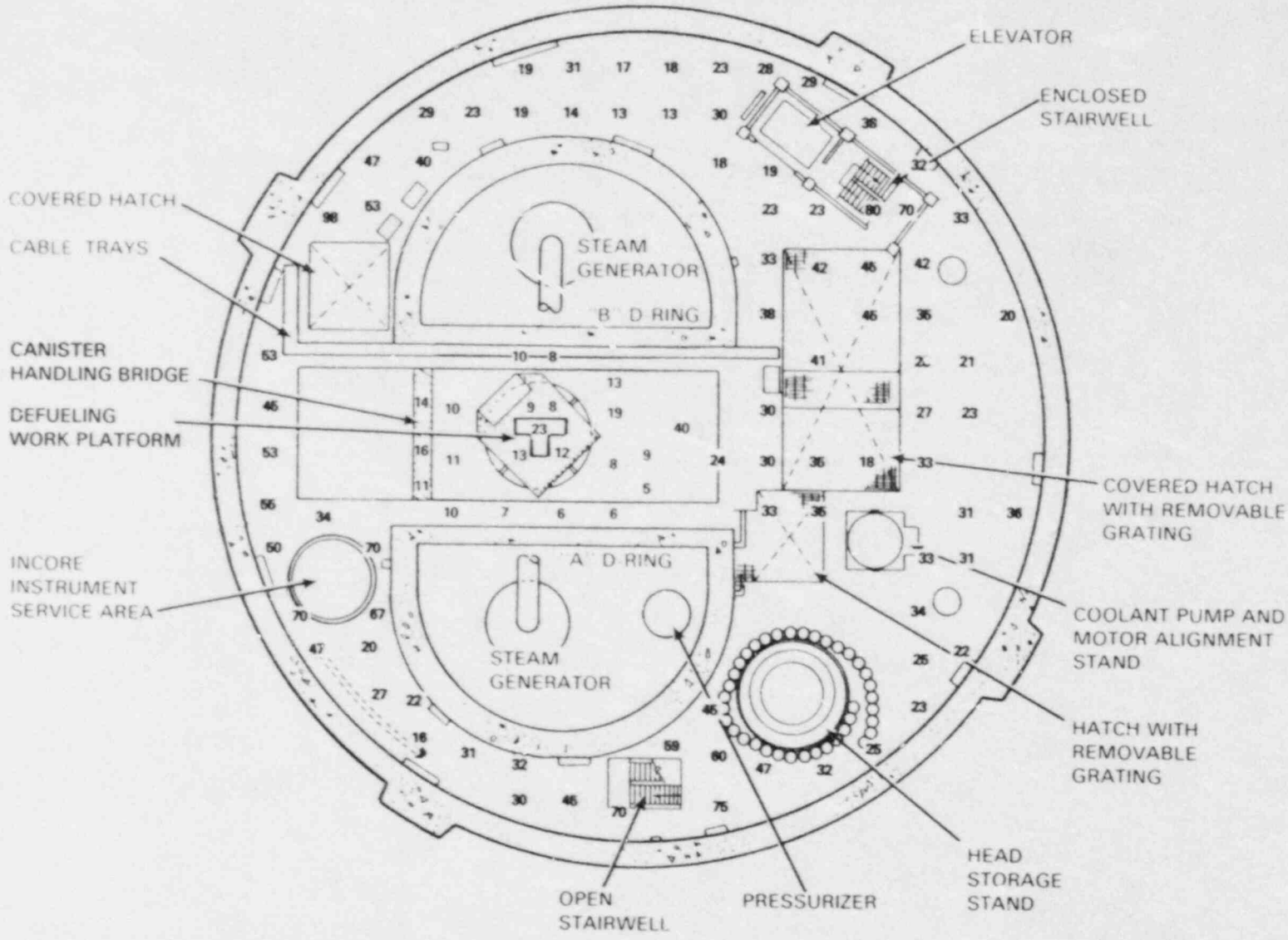

FIGURE 2.5. Map of the General Area Exposure Rates on the 347-Foot Elevation at the End of 1987, in mR/h for Gamma Radiation

averaged 120 mrem/man-h, but had been reduced to an exposure rate of about 80 to 90 mR/h by the end of 1987. Additional decontamination may be necessary before cleanup of the polar crane is complete.

The 282-foot elevation is the reactor building basement (Figure 2.6). The basement is divided into two distinct areas that are separated by D-ring-shaped shield walls. Within the D-ring walls, which extend to the 367-foot, 4-inch elevation, are the two steam generators, the pressurizer, the four reactor coolant pumps, and some structural components. Outside the D-ring walls, at the basement elevation, are large numbers of reactor control cables, various pumps and piping systems, the stairways, the reactor coolant drain tank (located in a shielded cubicle), and other equipment. During the accident, the major water flow path out of the reactor core was from the reactor coolant system, through the pressurizer relief valve, into the reactor coolant drain tank, and out the tank's vent line (through a ruptured blow-out disk) into the reactor building basement. This flow resulted in about 260,000 gallons (1,000,000 liters) of water covering the reactor basement to a depth of slightly more than 3.5 feet (1.1 meters). Water from the reactor building sprays, from additional reactor coolant, and from river-water inleakage through the building air coolers contributed approximately 360,000 gallons (1,400,000 liters) to the water level in the reactor building basement, raising it to a depth of approximately 8 feet (2.4 meters) (Munson and Harty 1985). Because the accident-generated water remained in the basement for several years, radionuclides concentrated on vertical surfaces at the water surface level and below, and were absorbed into the basement's concrete floors and walls. In addition, a layer of sludge was deposited on the basement floor, primarily from the river-water inleakage through the building air coolers.

Since the accident, the water has been drained, extensively processed, and recycled for use in decontamination. Water used during decontamination procedures on the upper levels has flowed into the basement, dissolving additional contamination in the basement, which has been removed as the water was pumped out and processed. Disposal of the accident-generated water was the subject of Supplement 2 to the PEIS (NRC 1987) and is not discussed further here.

Cleanup activities that have been conducted in the basement to date include: radiation monitoring with instrumentation mounted on robots and with strings of dosimeters suspended from the 305-foot elevation, video inspections using robots and cameras lowered on cables from the 305-foot elevation, concrete cores collected using robots, flushing and pumping of the elevator shaft, and high- and low-pressure flushing by robots and flushing from upper elevations.

Approximately 22,000 pounds (9900 kilograms) of wet sludge has been removed from approximately half of the basement floor, pumped into a tank located in the auxiliary building, and solidified for burial at a low-level waste (LLW) disposal site. Part of the liquid was returned to the basement, with a limited amount, approximately 1000 gallons (3800 liters), processed. A small quantity of fuel fragments, estimated to be between 3.7 and

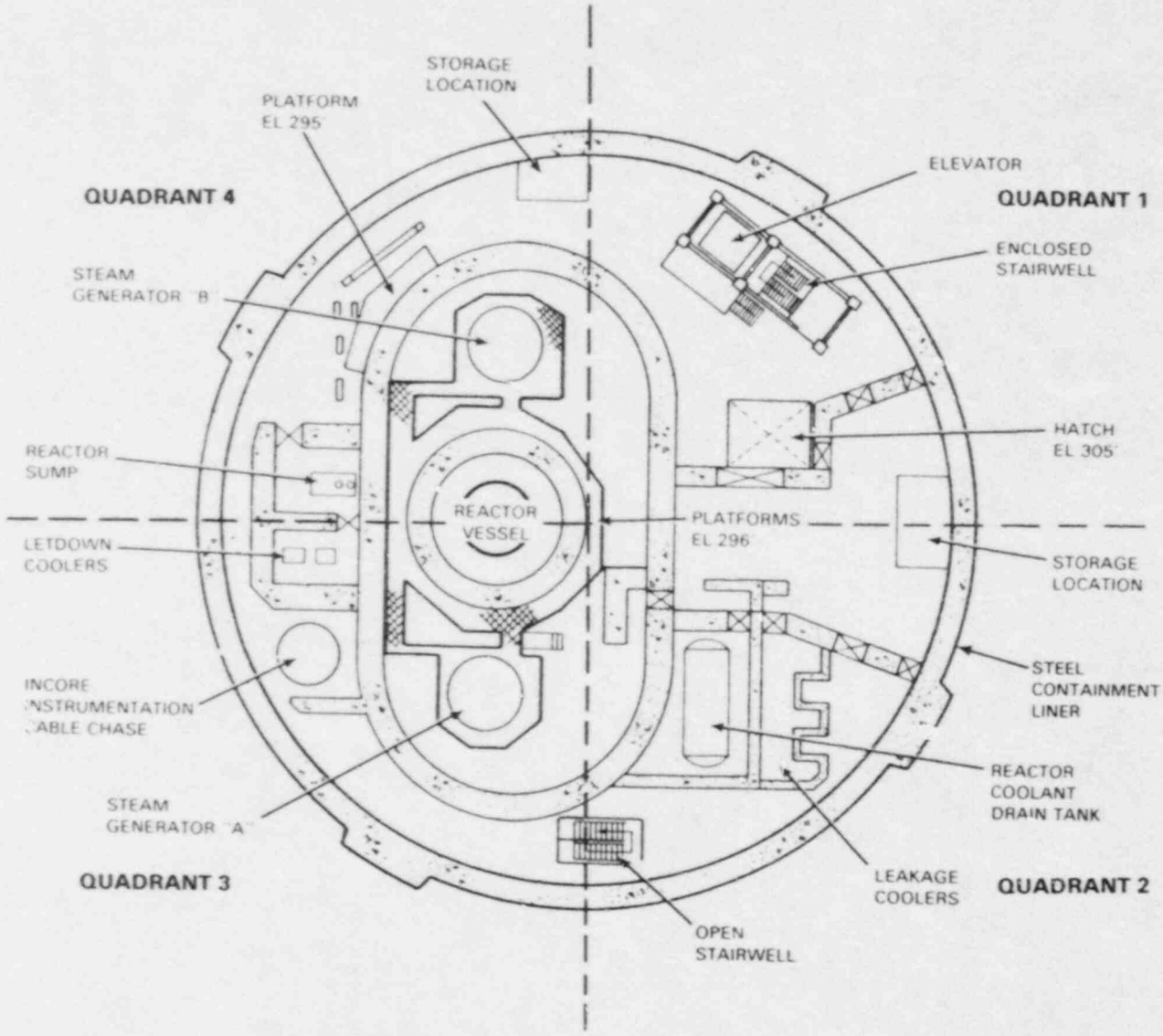

FIGURE 2.6 282-Foot Elevation of Reactor Building

7.1 pounds (1.7 and 3.2 kilograms) was deposited in the basement and has since mixed with solid materials in the sediment in the reactor building. Some of this material was most likely removed during sludge removal; however, because the amount removed cannot be accurately determined, it is conservatively assumed that 7.1 pounds (3.2 kilograms) of fuel remain dispersed in the basement.

A map of the radiation exposure rates in the basement at the end of 1987 is shown in Figure 2.7. Radiation levels vary somewhat with elevation. This map represents conditions 4 to 6 feet (1.2 to 1.8 meters) above the floor of the basement. General area radiation levels are around 35 R/h. The highest measured radiation levels (400 R/h to 1100 R/h before decontamination) in the reactor building basement were in the vicinity of the elevator shaft and enclosed stairwell. These structures, which are made of hollow concrete blocks, became saturated with the accident-generated water and absorbed radionuclides from the water. Analyses of core samples of the concrete block indicate that the contamination (primarily cesium-137) has completely penetrated the concrete block. Analyses of core samples from the concrete walls indicate that approximately 90 percent of the radioactivity (primarily cesium-137) in the concrete walls and the D-ring walls is within the first 1/8 inch (0.3 centimeter) to 1/4 inch (0.6 centimeter) of concrete.

Currently, the concrete walls in sections of quadrants 2 and 3 (see Figure 2.6) are being scabbled using robots in an effort to remove the surface layer of contaminated concrete. Only the wall area from 4 to 7 feet (1.2 to 2.1 meters) above the basement floor is being scabbled.

Projected work to be performed before the completion of defueling includes removing debris from sections of quadrants 2 and 3 that was generated during the scabbling process, constructing a manifold for waterflow to leach activity from the concrete-block wall of the stairway and elevator shaft structure, pumping the water from the basement and processing it through the submerged demineralizer system (SDS) and EPICOR II system, and a final flushing and removal of debris from the basement floor (using robots). The licensee has estimated that a maximum of 8600 pounds (3900 kilograms) of wet sludge [600 pounds (270 kilograms) of dry material] would remain after final removal of debris.

To complete the cleanup process following completion of defueling, the following tasks would be performed: further decontamination or removal of the concrete-block stairwell and elevator structure; removal of debris; removal of remaining sludge; removal of insulation, equipment, and electrical boxes; scabbling and sealing of remaining walls and floors; and removal of contamination from remaining structures. Although methods of leaching contamination from the concrete-block walls are planned, it is likely that removal of the structure will be necessary even following leaching. The decision on the method to be used to remove the contamination will be based on leaching tests, chemical and engineering analysis of the structure, and the capabilities of the robots in use at the facility.

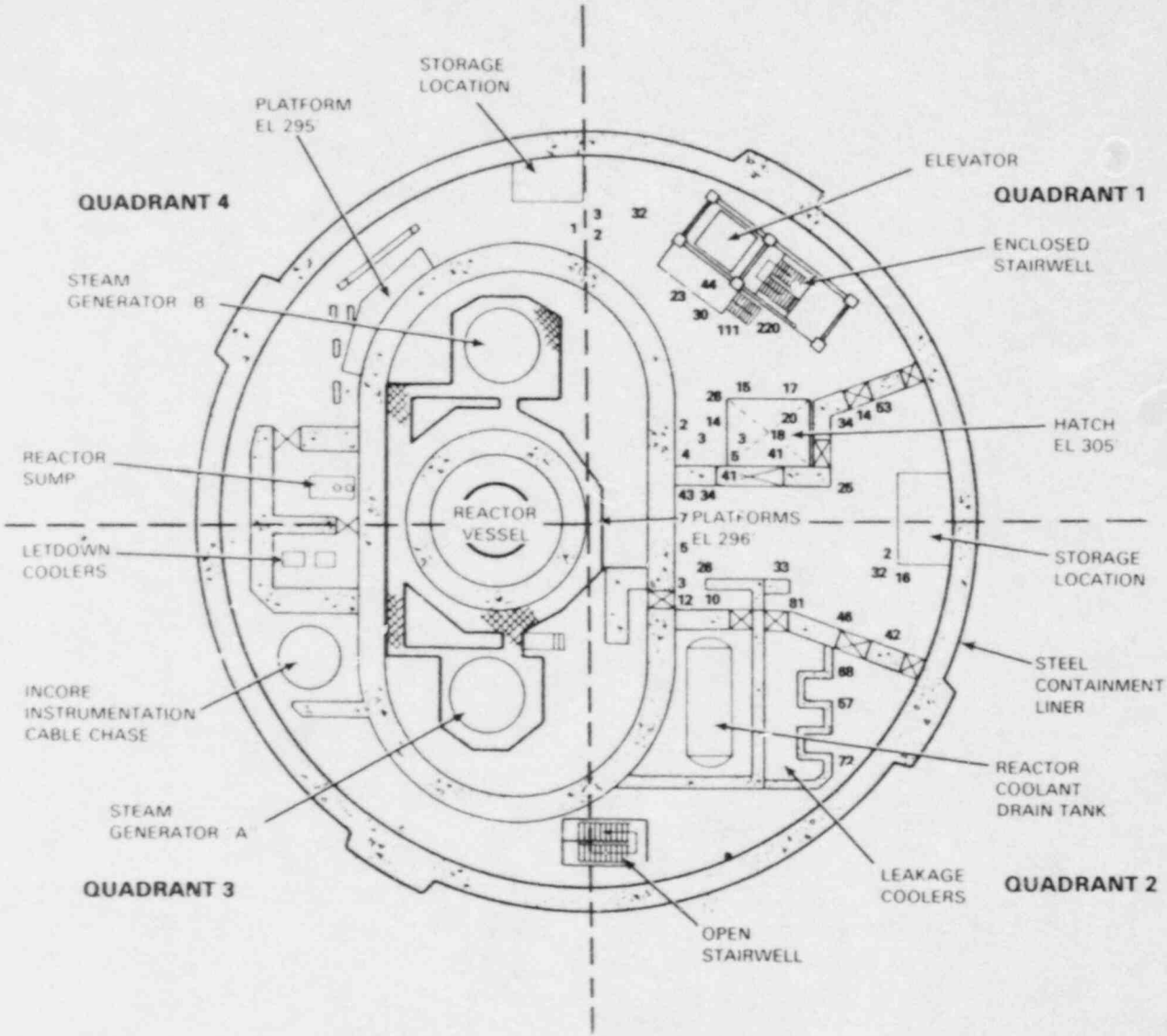

FIGURE 2.7 Map of the General Area Exposure Rates 4 to 7 Feet (1.2 to 2.1 Meters) Above the 282-Foot Elevation at the End of 1987, in R/h for Gamma Radiation

2.1.2 Reactor Vessel Defueling and Disassembly

A comparison between Figures 2.8 and 2.9 illustrates the progress of defueling and disassembly to the end of 1987. Figure 2.8 is a cutaway view of the TMI-2 vessel showing the status of the disassembly and defueling process in October 1984.(a) This figure is explained on page 2.8 of Supplement 1 to the PEIS (NRC 1984). Figure 2.9 is a cutaway view of the TMI-2 reactor vessel as it looked at the end of 1987. Progress in reactor vessel defueling and disassembly through 1987 has included removing the reactor vessel head, the upper plenum assembly (the device that positions the control rods), and most of the fuel. The head was placed on a storage stand at a shielded location on the 347-foot level. The internals indexing fixture was installed after the reactor vessel head was removed. It remains on the reactor vessel, flooded to about 15.5 feet (4.7 meters) above the top of the core region. The defueling platform is located on top of the internals indexing fixture. A dam was installed across the fuel transfer canal to create a storage pool for the plenum assembly and the fuel canisters. The plenum assembly was removed intact and stored in the deep end of the fuel transfer canal under 5 feet (1.5 meters) of water. [Total depth of the water in this end of the fuel transfer canal is 20 feet (6.1 meters).] A water cleanup system was installed to clarify and decontaminate the water used for defueling operations.

A total of 195,000 pounds (88,000 kilograms) of core material (fuel, structural material, and absorber material) had been removed from the reactor vessel as of December 31, 1987. This amount constitutes 67 percent of the total estimated postaccident core materials inventory, an estimated 293,000 pounds (133,00 kilograms). As of December 31, 1987, 154 canisters of damaged core material [156,000 pounds (71,000 kilograms)] had been shipped from TMI and 55 canisters were awaiting shipment. The amount shipped constitutes approximately 53 percent of the estimated core materials inventory. Table 2.1 shows the estimated distribution on December 31, 1978, of core material remaining in the reactor vessel. The latter stages of defueling may require cutting through the lower grid plates and flow distributor forging in the lower core support assembly and removing the fuel that is located in the bottom of the vessel. A portion of the core baffle plates will be removed to permit access for defueling the region between the baffle plates and the core barrel. Fuel particles that were swept into the outlet nozzles of the reactor vessel may be removed as part of defueling. Defueling will continue until all the fuel accessible throughout the reactor vessel has been removed. After defueling, reactor internals may be stored under shielding in the defueling canal or returned to the vessel. Sectioning and disposal of the reactor internals and reactor vessel are not considered part of cleanup because radiation levels expected from these components would be no higher than in a normal reactor nearing the end of its life.

(a) A cutaway view of a typical, undamaged pressurized water reactor (PWR) vessel was shown in Figure 6.1 of the PEIS (NRC 1981).

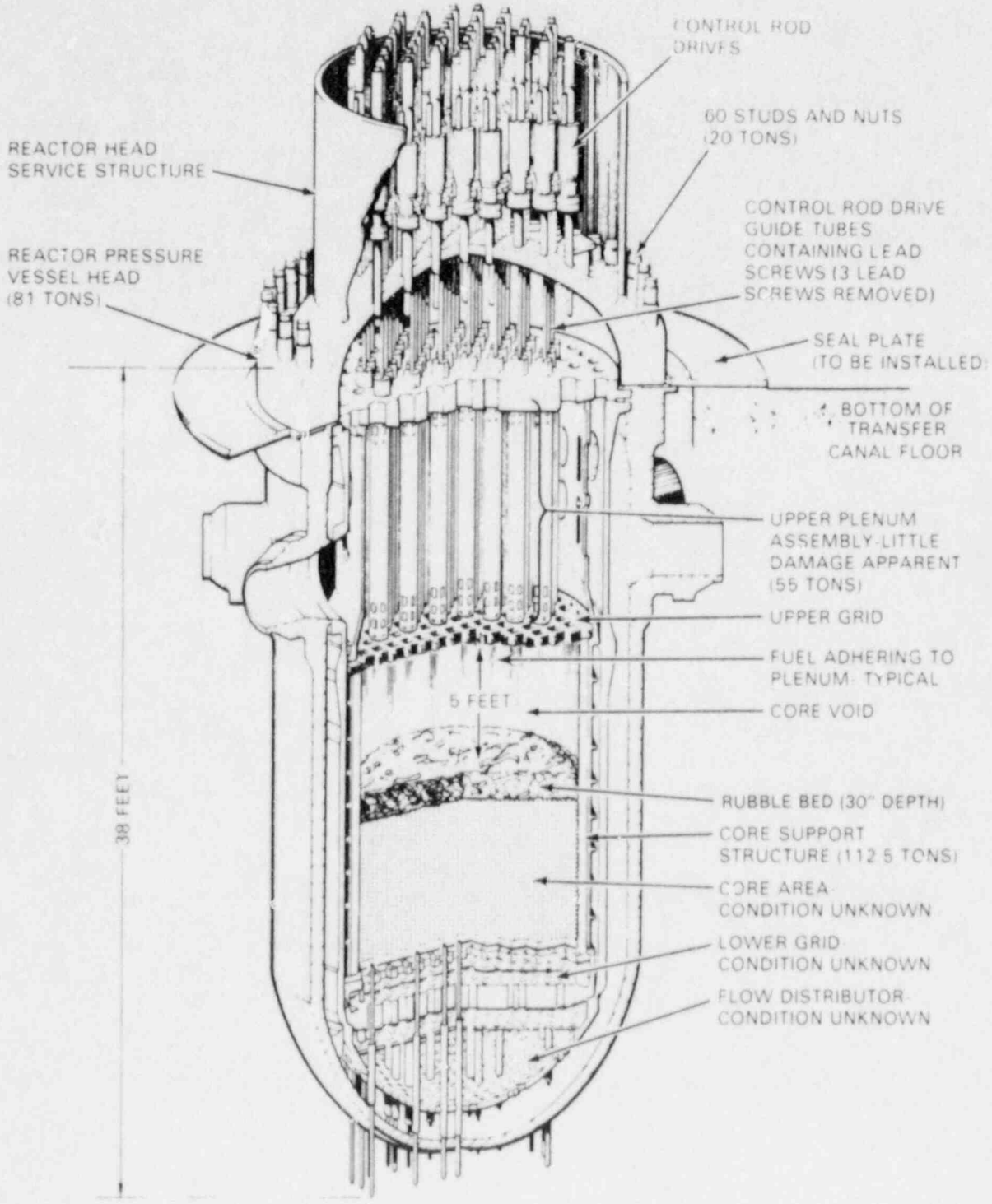

FIGURE 2.8. Cutaway View of TMI-2 Vessel Showing Status in October 1984

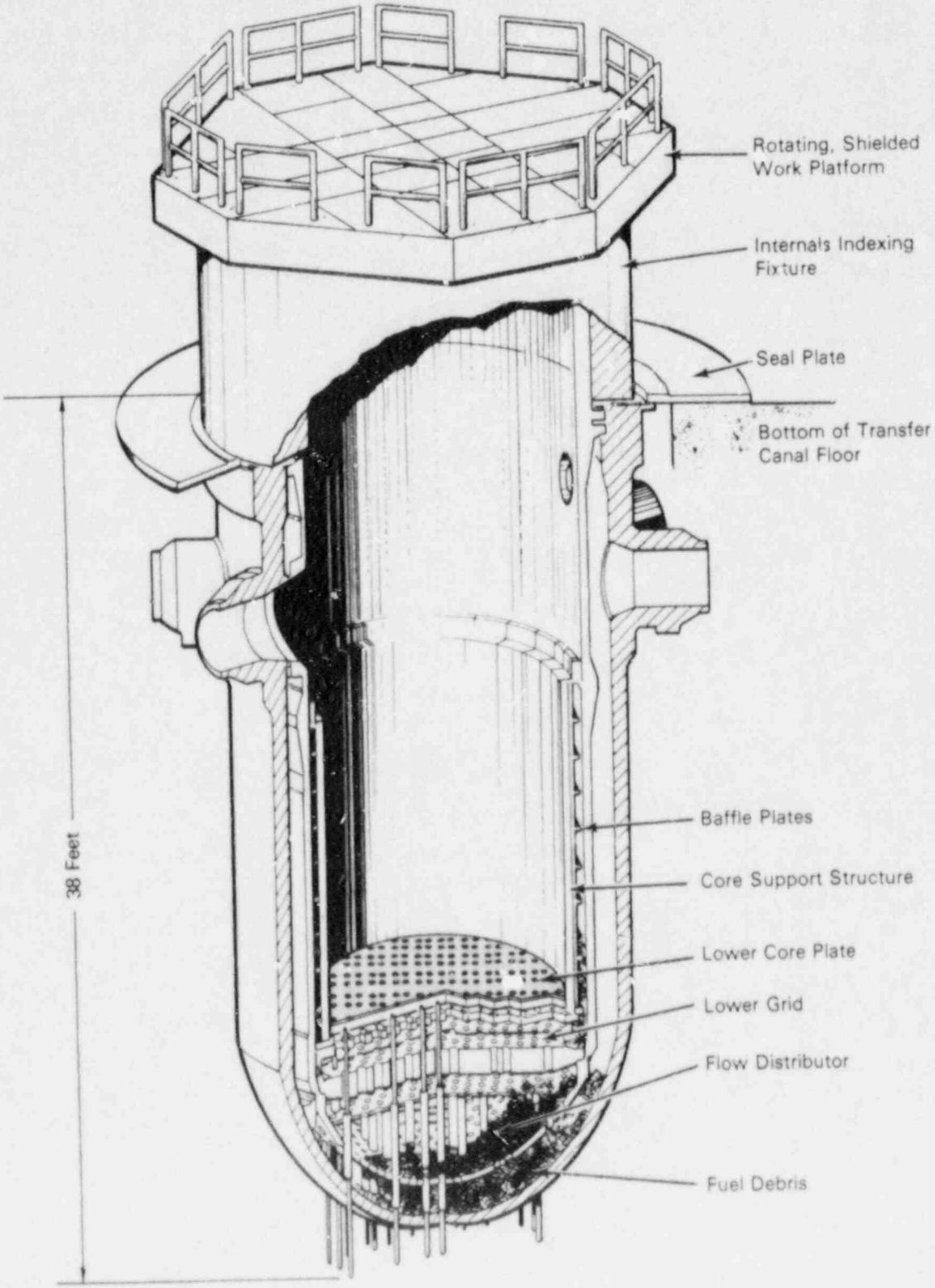

FIGURE 2.9. Cutaway View of TMI-2 Vessel at the End of 1987

TABLE 2.1. Estimated Core Material Distribution in the Reactor Vessel (Source: NRC 1988)

Location	Estimated Quantity, pounds
Lower Core Region	200
Lower Core Support Assembly	
Resolidified material	10,000
Loose material (vacuumable)	9,500
Loose material (rods and rocks)	7,500
Lower Head	
Monolith or fused material	11,000
Postaccident loose material (nonvacuumable)	20,000
Postaccident loose material (vacuumable)	10,000
Newly relocated loose material (vacuumable)	17,000
Newly relocated rods and rocks	2,500
Core Former Region	9,300

2.1.3 Reactor Coolant System Decontamination

A diagram of the reactor coolant system is shown in Figure 2.10. Directional radiation surveys performed by the licensee confirm that reactor fuel and fission products were dispersed throughout the reactor coolant piping system as finely divided particles and/or as plating on surfaces. During the accident, a small quantity of finely fragmented fuel was also released into the basement by reactor coolant escaping through the pressurizer relief valve to the reactor coolant drain tank and into the basement through a disk, which ruptured to relieve pressure in the reactor coolant drain tank. Directional surveys of the reactor coolant system components have permitted preliminary estimates of fuel present in these locations. Fuel has been removed and is currently being removed from some portions of the system, such as the steam generators. By the end of defueling, an estimated 55 to 320 pounds (25 to 150 kilograms) of fuel will remain outside the reactor vessel. Possible residual fuel locations outside the reactor vessel and current licensee estimates of the fuel quantities remaining after defueling are listed in Table 2.2. The quantity at each of these locations was determined using a variety of methods, including gamma spectroscopy and path flow modeling.

After defueling, some additional decontamination of the reactor coolant system would probably be required before the completion of cleanup. Decontamination of the reactor coolant system is expected to involve additional mechanical decontamination techniques, such as vacuuming fuel debris and water flushing of the system, and may involve some chemical decontamination techniques as well. The reactor coolant system may be decontaminated all at once or section by section.

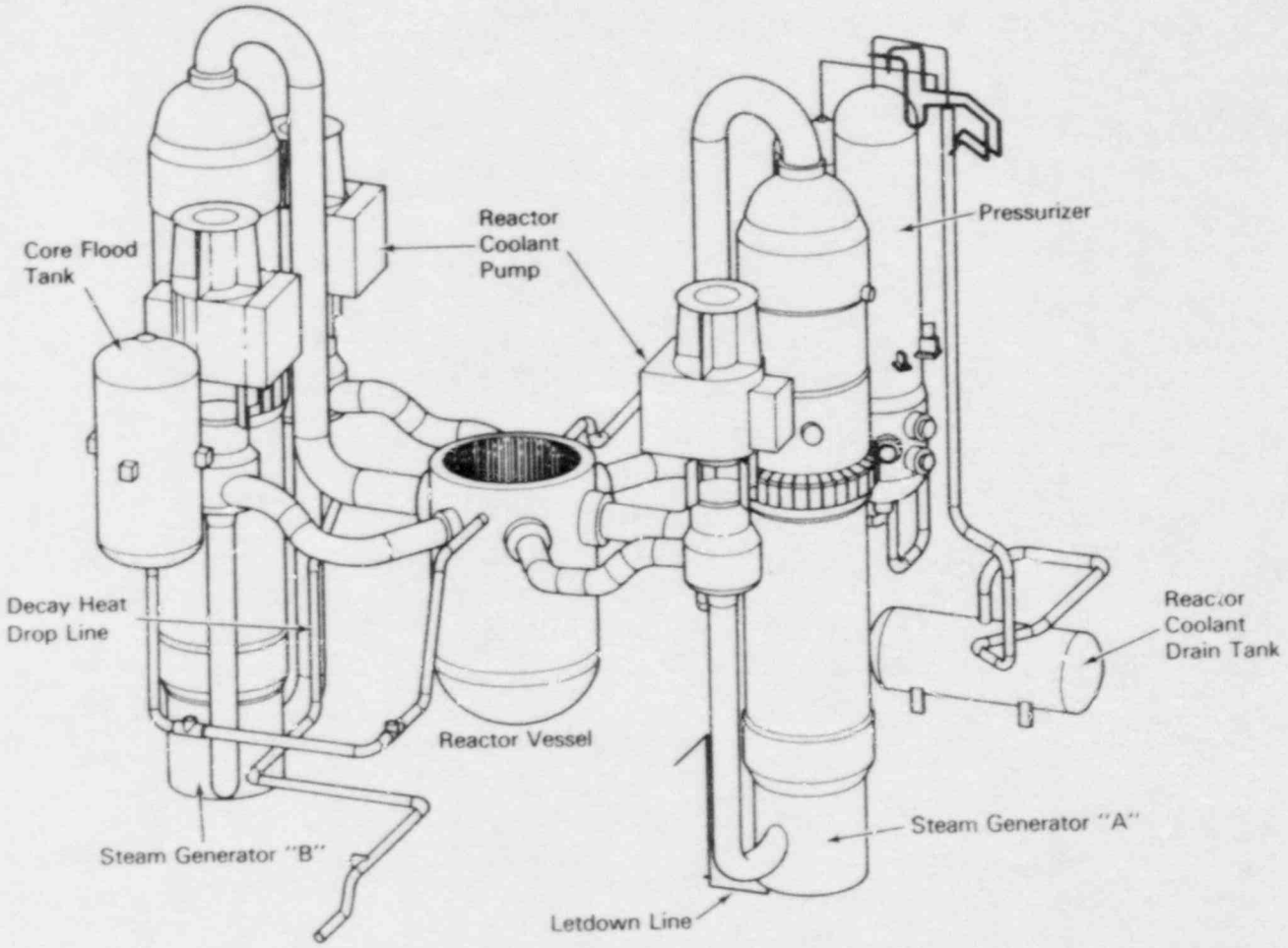

FIGURE 2.10. Reactor Coolant System Components (Source: GPU 1987b)

TABLE 2.2. Estimated Quantity of Fuel Located Outside of the Reactor Vessel at the End of Defueling (Source: GPU 1987b)

Location	Quantity of Fuel	
	lb	kg
Reactor Building		
Reactor coolant system Reactor coolant pipes Pressurizer Steam generators Reactor coolant pumps	33-276	15-125
Outside the reactor coolant system Upper plenum assembly Reactor building basement Core flood tanks Makeup and purification demineralizers Letdown line and coolers	11-33	5-15
Auxiliary and Fuel Handling Building		
Pipe systems, drains, floors, and sumps	<11	<5
TOTAL	55-320	25-150

2.1.4 Auxiliary and Fuel-Handling Building Cleanup

The auxiliary and fuel-handling building (AFHB) was also designed and constructed to maintain its structural integrity during a variety of accidents. However, unlike the reactor building, the AFHB was not designed to be leak-free during such conditions.

The AFHB is composed of two sections that are separated by a common wall. The auxiliary section contains tanks, pumps, piping, and other equipment to process and store water for the reactor coolant system and to treat radioactive wastes. The fuel-handling section contains large basins, or pools, for the storage of spent fuel. Equipment, such as the cranes used to remotely handle the spent fuel, is also present. The general layout of the AFHB is shown in Figures 2.11 and 2.12. The truck bay area within the AFHB is shared with TMI-1.

The interior of the AFHB and piping systems were also contaminated as a consequence of the accident although less severely than the reactor building. There are 26 piping systems in the AFHB that were contaminated as a result of the accident. Cleanup of the AFHB started shortly after the accident and is still under way. So far, considerable amounts of debris and contaminated equipment have been removed, contaminated systems have been flushed, and the

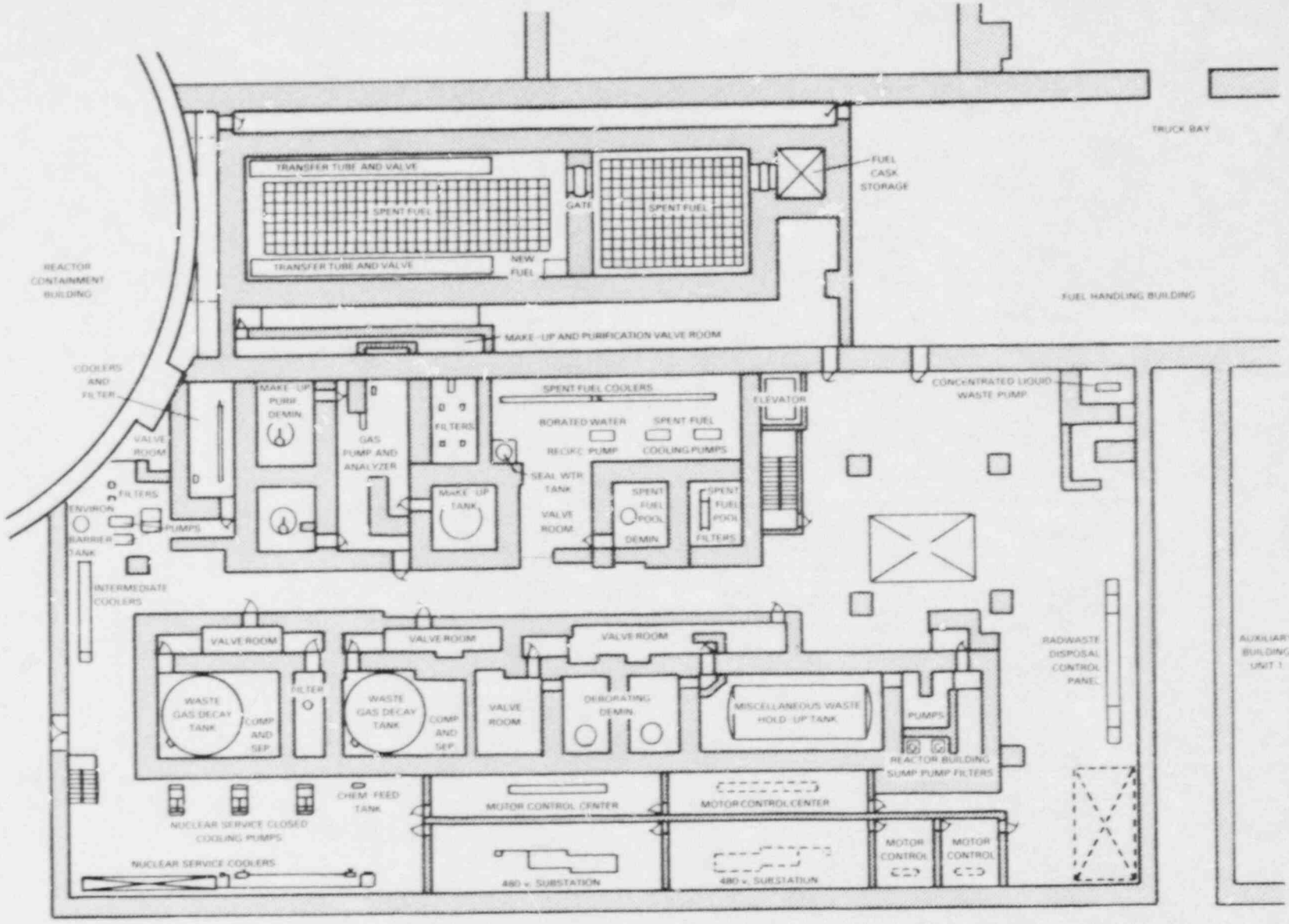

FIGURE 2.11. Plan View of Auxiliary and Fuel-Handling Building (305-foot elevation)

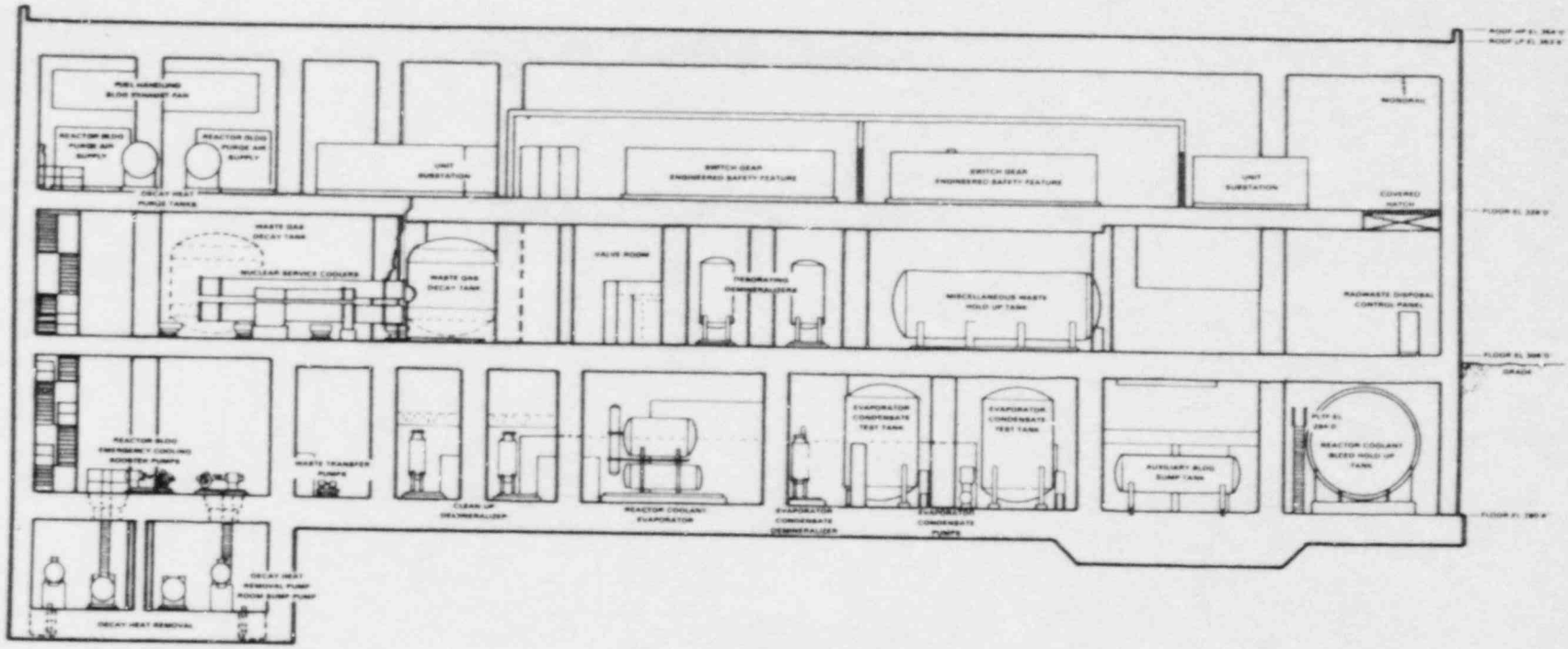

FIGURE 2.12. Cutaway View of Auxiliary and Fuel-Handling Building

building and remaining equipment are in the process of being decontaminated. Because most of the interior surfaces of the building (walls, floors, etc.) are constructed of uncoated concrete, radioactive materials have penetrated into the surfaces to varying depths. High- and low-pressure water sprays, wet vacuuming, scabbling (usually followed by an application of sealant), and manual wiping have reduced both the level of smearable contamination on building surfaces and the dose rates. Some temporary dose rate reduction has also been achieved by shielding radiation sources, such as floor drains, the elevator shaft, and various valves, piping, and pipe dead legs. Cleanup, as previously defined, has been completed in most halls and normally occupied areas. The cubicle areas have proven to be the most difficult to decontaminate because of the concentration of equipment (tanks, filters, piping, etc.), the crowded work space, and the high contamination and high radiation levels. Some of the more highly contaminated components have been removed, however, and the radiation levels in most cubicles have been substantially reduced. By the end of 1987, 118 of the 143 contaminated cubicles (136 of which are located in the AFHB) were decontaminated to radiation exposure rates that are generally less than 15 mR/h. The remaining cubicles are scheduled to be decontaminated before the end of defueling. By the end of defueling, the exposure rates in the remaining cubicles will generally approach 15 mR/h.

The fuel-handling section of the AFHB has undergone extensive decontamination and refurbishment to prepare for defueling. All the contaminated temporary water-storage tanks have been removed from the "A" fuel pool, the pool liner cleaned, and new fuel canister racks and a canister dewatering system installed. However, contamination has been reintroduced to the fuel pool as a result of defueling operations. At the present time, dose rates throughout the fuel-handling section are generally less than 15 mR/h.

After defueling has been completed and the fuel has been shipped offsite, the fuel pools will be drained and decontaminated. It is expected that the dose levels present in the AFHB at the end of defueling will be similar to those found in an operating reactor at the end of its life, and no further work will be required to complete cleanup.

2.2 SOURCE CHARACTERISTICS

The potential environmental impacts of cleanup activities at TMI-2 depend in part on the quantity and distribution of radionuclides present in the facility. Several methods have been used to determine the quantity and distribution of radionuclides, including direct measurements, sample analysis, and reactor operation and accident data analysis. Identifying all the radionuclides present in the facility is difficult using measurement or sample-analysis techniques because (1) there are a large number of radionuclides associated with the fuel and (2) the relatively large quantities of cesium-137 and strontium-90 make detection of other radionuclides difficult. The estimates of the amount of cesium-137 and strontium-90 present are based on measurements. However, the number and the quantity of the remaining radionuclides are estimated from the amount present at the time of the accident. The amount present at the time of the accident is, in turn,

based on the original composition of the fuel and the operating history of the TMI-2 reactor.

The estimated inventory of radionuclides has been calculated using the ORIGEN-2 computer code. Table 2.3 provides the inventory of the longer-lived radionuclides estimated to be present at the time of the reactor shutdown on March 28, 1979 (GPU 1987a; Cunnane and Nicolosi 1982). Table 2.3 also provides the estimated inventory, decay-corrected to January 1, 1989 (the licensee's current estimate for the end of defueling), that would have been present in the facility if no defueling or cleanup had taken place. The expected inventory of the decay products is also included.(a) Any isotope that would have been present in a quantity of less than 1 curie before January 1, 1989 (in the absence of defueling or cleanup), was not included.

The amount of radioactive material in TMI-2 at the completion of defueling will be considerably less than that shown in Table 2.3 because of defueling and cleanup. The majority of the radioactive material is being removed as the reactor vessel is defueled. The gaseous fission products that were released from the fuel to the containment atmosphere during the accident were later purged to the environment. Also, water-soluble fission products have been removed from the accident-generated water and shipped from the site in the resin liners.

Table 2.4 indicates the estimated maximum quantity of each radionuclide assumed to be present after defueling (with the exception of the fraction of activation products assumed to be incorporated into metal material). Table 2.4 also includes a brief description of the assumed location of each radionuclide after defueling. Although predictions have been made regarding the transport and deposition of materials released as vapors and/or aerosols during core heatup, refined modeling methods are not available for accurately analyzing the transport and deposition of the fragmentation debris, or the leaching of soluble materials from the damaged core (Cunnane and Nicolosi 1982). However, because significant amounts of core materials were released either as particulate debris or as soluble material, it is apparent that mechanisms were involved which distributed contamination through the facility. Therefore, conservative assumptions are made in this report to estimate the quantity and distribution of radioactive material expected to remain in the facility at the end of defueling. The radionuclides remaining after defueling can be grouped into three major categories: activation products, fission products, and actinides.

2.2.1 Activation Products

Activation products, including manganese-54, iron-55, cobalt-60, and nickel-63, were formed in the reactor core region, but outside of the fuel by

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(a) Those radionuclides with short-lived decay products, which have reached equilibrium or are approaching equilibrium, are listed on the same line in Table 2.3. Radionuclides with extremely short-lived decay products, such as strontium-90/yttrium-90 or cesium-137/barium-137m, are referred to in the text by using the designation for the parent isotope.

TABLE 2.3. Inventory of Isotopes in the TMI-2 Facility Following the Accident, Decay-Corrected to January 1, 1989 (Assuming No Cleanup)

Radionuclide	Halflife[a]	Calculated Activity, Ci	
		March 28, 1979	January 1, 1989[b]
Tritium	12.3 y	8,800	5,100
Carbon-14	5,726 y	16	16
Manganese-54	312 d	26,000	12
Iron-55	2.68 y	103,000	8,300
Cobalt-60	5.27 y	98,000	27,000
Nickel-63	100 y	6,000	5,600
Selenium-79	65,000 y	3.3	3.3
Krypton-85	10.7 y	94,000	50,000
Strontium-90/Yttrium-90	28.8 y/2.7 d	720,000/890,000	570,000
Zirconium-93/Niobium-93m	1,500,000 y/13.6 y	16/0.15	16
Technetium-99	214,000 y	110	110
Ruthenium-106/Rhodium-106	1 y/30 s	53,000,000/5,400,000	62,000
Cadmium-113m	14 y	3.2	2.0
Antimony-125/Tellerium-125m	2.7 y/ 58 d	150,000/2,100	12,000/3,000
Tin-126/Antimony-126m/Antimony-126	100,000 y/19 m/12.4 d	2.3/96/1,200	2.3
Cesium-134	2.06 y	160,000	6,000

(a) s = seconds; m = minute; d = day; y = year.
(b) The values represent decay-corrected activities on January 1, 1989, assuming no defueling or cleanup effort had taken place.

TABLE 2.3. (contd)

Radionuclide	Halflife(a)	Calculated Activity, Ci	
		March 28, 1979	January 1, 1989(b)
Cesium-135	3,000,000 y	2.2	2.2
Cesium-137/Barium-137m	30.2 y/2.5 m	820,000/760,000	660,000
Cerium-144/Praseodymium-144	284.5 d/17.3 m	23,000,000/ 24,000,000	3,900/3,800
Promethium-147	2.62 y	2,500,000	190,000
Samarium-151	90 y	18,000	17,000
Europium-152	13 y	44	26
Europium-154	8.8 y	7,600	3,400
Europium-155	4.9 y	47,000	12,000
Uranium-234	245,000 y	120	120
Uranium-235/Thorium-231	704,000,000 y/25.5 h	4/4	4
Uranium-236	23,400,000 y	3.2	3.2
Uranium-238/Thorium-234/ Protactinium-234m	4.47×10^9 y/24 d/1.17 m	27/27/27	27
Plutonium-238	87.7 y	760	710
Plutonium-239	24,100 y	8,600	8,600
Plutonium-240	6,570 y	2,400	2,400
Plutonium-241/Americium-241/ Uranium-237	14.4 y/432 y/6.75 d	160,000/19/ 13,500,000	100,000/1,900/ 2.3

(a) s = seconds; m = minute; d = day; y = year.
(b) The values represent decay-corrected activities on January 1, 1989, assuming no defueling of cleanup effort had taken place.

TABLE 2.4. Maximum Anticipated Inventory and General Location of Radionuclides at the End of Defueling(a)

Radionuclide	Activity at End of Defueling, Ci	Location
Tritium	<1	Moisture in piping, etc.
Carbon-14	0.51	Dispersed
	0.026	Fuel debris
Manganese-54	1.2	Activated metals in fuel debris or corrosion film on piping
Iron-55	830	Activated metals in fuel debris or corrosion film on piping
Cobalt-60	2,700	Activated metals in fuel debris or corrosion film on piping
Nickel-63	560	Activated metals in fuel debris or corrosion film on piping
Selenium-79	0.11	Dispersed
	0.0053	Fuel debris
Krypton-85	<1	Fuel debris
Strontium-90/	1,400	Dispersed
Yttrium-90	870	Fuel debris
Zirconium-93	0.026	Fuel debris
Niobium-93m	0.21	Dispersed
	0.026	Fuel debris
Technetium-99	3.5	Dispersed
	0.18	Fuel debris
Ruthenium-106/	2,000	Dispersed
Rhodium-106	60	Fuel debris
Cadmium-113m	0.064	Dispersed
	0.0032	Fuel debris
Antimony-125	380	Dispersed
	4.8	Fuel debris
Tellurium-125m	96	Dispersed
	4.8	Fuel debris

(a) The end of defueling (removal of more than 99 percent of the fuel) was assumed to occur January 1, 1989, for the purpose of estimating radioactive decay.

TABLE 2.4. (contd)

Radionuclide	Activity at End of Defueling, Ci	Location
Tin-126/Antimony-126m/Antimony-126	0.074	Dispersed
	0.0037	Fuel debris
Cesium-134	190	Dispersed
	2.4	Fuel debris
Cesium-135	0.07	Dispersed
	0.00088	Fuel debris
Cesium-137/Barium-137m	21,000	Dispersed
	260	Fuel debris
Cerium-144	6.2	Fuel debris
Praseodymium-144	6.1	Fuel debris
Promethium-147	300	Fuel debris
Samarium-151	540	Dispersed
	27	Fuel debris
Europium-152	0.042	Fuel debris
Europium-154	5.6	Fuel debris
Europium-155	19	Fuel debris
Uranium-234	0.19	Fuel debris
Uranium-235/Thorium-231	0.0064	Fuel debris
Uranium-236	0.0051	Fuel debris
Uranium-237	0.0037	Fuel debris
Uranium-238/Thorium-234/Protactinium-234m	0.043	Fuel debris
Plutonium-238	1.1	Fuel debris
Plutonium-239	14	Fuel debris
Plutonium-240	3.8	Fuel debris
Plutonium-241	160	Fuel debris
Americium-241	3.0	Fuel debris

activation of stainless steel and other metal components. In operating reactors, small amounts of these activation products form in a corrosion film on the reactor piping. Additional amounts of these activation products are associated with the metal portions of the core and the reactor internals. It is assumed that most of the activation products in the TMI-2 reactor are present as solid material removed with the fuel or incorporated into the stainless steel of the reactor vessel, plenum assembly, and remaining internals. However, a small amount may remain associated with fuel debris that was dispersed through the reactor coolant system or as a corrosion film in the reactor coolant system piping and on the inside of the reactor vessel. For this analysis it is conservatively estimated that 10 percent of the activity for each activation product will remain in the reactor building at the end of defueling with particulates located in the reactor coolant system or as a removable corrosion film in the piping or vessel internals. The other 90 percent is assumed to have been removed during the defueling process or to be incorporated in the stainless steel composing the reactor vessel, plenum assembly, and internals; it is therefore inaccessible.

2.2.2 Fission Products

Fission products were formed within the fuel elements by the nuclear fission of uranium-235 as the reactor operated. The transport and deposition of the fission products were dependent on the chemical and physical state of the radionuclide (e.g., soluble or insoluble material, or gas). Fission products were considered in groups based on their chemical and physical properties. Where definitive information on the chemical state of a fission product was lacking, assumptions were made regarding the transport and deposition of the fission product. These assumptions were based on the information available from fuel measurements and contamination measurements throughout the reactor building, as well as on the physical state of the radionuclide. Fission products are discussed in the following order: (1) gaseous fission products (krypton-85), (2) tritium, (3) somewhat soluble fission products (carbon-14, selenium-79, strontium-90, niobium-93m, technetium-99,(a) ruthenium-106, cadmium-113m, antimony-125,(a) tellurium-125m, tin-126, cesium-134, cesium-135, cesium-137, and samarium-151), and (4) relatively insoluble fission products (zirconium-93, cerium-144, praseodymium-144, promethium-147, europium-152, europium-154, and europium-155).

2.2.2.1 Gaseous Fission Products

The noble gas krypton-85 is formed by the fission process. In an undamaged reactor, krypton-85 remains in the fuel rods. During the accident and subsequent defueling, krypton-85 was released. It is estimated that less than 1 curie of krypton remains associated with the residual fuel.

(a) Some technetium-99 and antimony-125 may be present as activation products in metal components containing molybdenum or tin, respectively.

2.2.2.2 Tritium

Tritium was produced within the reactor fuel by several mechanisms, including ternary fission of uranium. As a result of the accident, a fraction of the tritium was released to the containment atmosphere and subsequently vented to the environment as either tritium gas or water vapor. The remaining tritium is present in the accident-generated water (as discussed in Supplement 2 to the PEIS). Disposal of the accident-generated water from the facility will result in the removal of essentially all of the remaining tritium. The environmental impacts of this removal were evaluated in Supplement 2 and will not be considered further in this document. The amount of tritium expected to be present in any remaining moisture inside the reactor building, AFHB, and tanks will be small and is estimated to be less than 1 curie. It is assumed that the remaining tritium would be present in sealed piping or as water that exchanged with water incorporated in the concrete present in the building.

2.2.2.3 Somewhat Soluble Fission Products

Fission products that are assumed to be at least partially soluble in water include carbon-14, selenium-79, strontium-90, niobium-93m, technetium-99, ruthenium-106, cadmium-113m, antimony-125, tellurium-125m, tin-126, cesium-134, cesium-135, cesium-137, and samarium-151. The degree of solubility varies among the isotopes listed and depends on the chemical form of the isotope. Because these isotopes are known to exist as water soluble compounds in some circumstances, they were assumed to have been distributed in various degrees throughout the reactor building and the AFHB during the accident. Measurements have been made to estimate the amount of cesium-137 and strontium-90 present in various portions of the reactor building and the AFHB. Assumptions, listed below, were made regarding the distribution of the other somewhat soluble fission products.

Strontium-90 and cesium-137 concentrations have been determined by measurements and samples taken throughout the two buildings. The measurements indicate that the major portion of the strontium-90 and cesium-137 in the reactor building (with the exception of the amount contained in the fuel) is located in the concrete-block wall surrounding the enclosed stairwell and elevator shaft in the reactor building basement. The licensee has estimated that approximately 21,000 curies of cesium-137 and 850 curies of strontium-90 are located in the concrete-block wall. The efforts that are being made to leach radioactivity from the concrete-block wall (see Section 2.1.1) may reduce this inventory somewhat. The licensee has further estimated that 350 curies of cesium-137 and 410 curies of strontium-90 would remain in the dried sludge on the basement floor after completion of defueling. These estimates of activity are slightly more conservative than estimates made by Munson and Harty (1985) based on dose rate data. Munson and Harty (1985) estimated that 11,000 to 19,000 curies of cesium-137 are present in the concrete-block wall. Munson and Harty (1985) further estimated that the total amount of radioactivity located in the reactor building basement, including activity located other than in the concrete-block wall, was between 12,000 and 21,000 curies of cesium-137. For the analysis in this report, it is assumed that a total of approximately 21,000 curies of cesium-137 are

present in the reactor building basement with 19,000 curies (approximately 90 percent) in the concrete block of the stairwell/elevator shaft, 400 curies (2 percent) in the sludge that will remain on the basement floor, and 1600 curies (8 percent) on the concrete slab walls, equipment, and overhead structures in the reactor building basement.

Recent analyses of core samples from the block wall indicate a ratio of cesium-137 to strontium-90 of 24:1. Using a slightly more conservative ratio, this analysis assumes that a total of 880 curies of strontium-90 are in the concrete-block wall of the stairwell/elevator shaft and 67 curies on concrete slab walls, equipment, and overhead structures in the reactor building basement. The licensee's estimate of 410 curies was used for the amount of strontium-90 activity in the sludge on the basement floor.

The licensee has estimated that the upper elevations of the reactor building (the 305-foot level and above, excluding the area below the 349-foot level of the D-rings) contain 5.6 curies of mixed isotopes loosely distributed. The licensee's estimate was intended to be conservative. However, even assuming that the amount of contamination on the upper elevations is twice as much as that estimated by the licensee, this quantity is negligible compared to the amount assumed to be present in the reactor building basement. Additional activity would be present on the leadscrews, the plenum, etc.; however, this activity is largely incorporated into the metal parts and is not easily removed. The cleanup of the AFHB is currently underway and the licensee has estimated that, at the completion of defueling, the dose levels in the AFHB will be similar to those found in an operating reactor at the end of its life. The amount of contamination remaining in the AFHB will thus be negligible in comparison to the amount present in the reactor building.

A ratio of 1:110 is used to estimate the amount of cesium-134 compared to cesium-137. This ratio assumes that the two isotopes are distributed similarly and is based on the ratio of cesium-134 to cesium-137 shown for January 1, 1989, in Table 2.3. Likewise, a ratio of 1:300,000 was used to estimate the amount of cesium-135 present compared to cesium-137. Using the estimate of 21,000 curies of cesium-137 in the reactor building basement, the amounts of cesium-134 and cesium-135 estimated to be present in the reactor building basement are 190 curies of cesium-134 and 0.07 curies of cesium-135. It is assumed that they are distributed as 90 percent in the concrete block wall, 2 percent in the sludge on the floor, and 8 percent in concrete slab walls, equipment, and overhead structures in the basement.

Although the carbon, selenium, niobium, technetium, ruthenium, cadmium, antimony, tellurium, tin, and samarium isotopes have not been detected (except in areas with defueling-generated contamination or in the accident-generated water), it is possible that some of these isotopes dissolved in the water and were distributed within the building, only in smaller quantities than the cesium and strontium isotopes. Based on the estimated 3.2 percent of the cesium-137 distributed in the reactor building (21,000 curies of the 660,000 curies following the accident and decay-corrected to January 1, 1989), it is conservatively assumed that 3.2 percent of the carbon, selenium, niobium, technetium, ruthenium, cadmium, antimony, tellurium, tin, and

samarium isotopes are similarly distributed throughout the reactor building and AFHB. This estimate is considered conservative because the chemical forms of these isotopes are generally less soluble than cesium. It is further assumed (based on the distribution of cesium-137) that 90 percent of the activity distributed in the reactor building is located in the enclosed stairwell/elevator structure, 2 percent in the sludge on the basement floor, and the remaining 8 percent on the concrete slab walls, equipment, overhead structures, etc.

In addition to being distributed within the building as carried by the water, a fraction of the somewhat soluble isotopes are assumed to have remained in association with the fuel. Although the majority of the fuel will be removed during defueling, a fraction of the debris that was distributed throughout the reactor coolant system and in the reactor building basement will remain. The licensee has estimated that 55 to 320 pounds (25 to 150 kilograms) of uranium oxide will be left in fuel particles distributed outside the reactor vessel after defueling is completed. This corresponds to a maximum of 0.16 percent of the estimated mass of uranium oxide, 207,000 pounds (94,000 kilograms) originally in the reactor vessel. Isotopes that were somewhat soluble were likely leached from the fuel debris to some extent. The amount leached varied with the solubility of the isotope. Based on measurements of fuel from the reactor vessel, it is assumed that 25 percent of the original cesium remained with the fuel debris, 60 percent of the ruthenium, 25 percent of the antimony, and 95 percent of the strontium. To be conservative, it is assumed that close to 100 percent of the remaining somewhat soluble fission products (carbon, selenium, niobium, technetium, cadmium, tellurium, tin, and samarium) remained with the fuel debris.

2.2.2.4 Relatively Insoluble Fission Products

The remaining fission products (zirconium-93, cerium-144, praseodymium-144, promethium-147, europium-152, europium-154, and europium-155), which are considered highly insoluble, are assumed to remain totally in association with the fuel. Analyses of removed fuel tend to confirm this assumption. These isotopes would be removed almost completely by defueling, except for the small amounts distributed with the fuel particles through the reactor coolant system. The estimated number of curies for these isotopes is based on the percentage of the fuel (0.16 percent) expected to remain in the facility after defueling.

2.2.3 Actinides

The actinides include uranium isotopes (uranium-234, uranium-235, uranium-236, uranium-237, and uranium-238), uranium daughter products (thorium-231, thorium-234, protactinium-234m), and transuranics formed by neutron capture (plutonium-238, plutonium-239, plutonium-240, plutonium-241, and americium-241). These isotopes, like the insoluble fission products, are expected to remain in close association with the fuel. Radiochemical analysis of removed fuel tends to confirm the close association of these isotopes with the fuel. Small quantities of these isotopes were distributed with the fuel particles throughout the reactor coolant system. The estimated

activity of each radionuclide remaining in the facility is based on the percentage of fuel (0.16 percent) assumed to remain in the facility after the defueling.

2.3 REGULATORY AND ADMINISTRATIVE CONSIDERATIONS

Cleanup of TMI-2, including any storage of waste, must be carried out in accordance with applicable Federal and State laws, regulations, and permits as discussed in the following sections.

2.3.1 U.S. Environmental Protection Agency Regulations

The U.S. Environmental Protection Agency (EPA) has the responsibility and authority to set standards for the release of radionuclides to the environment to protect the public from radioactivity. The EPA also has the authority to regulate the handling, storage, and disposal of hazardous non-radioactive materials. These authorities arise from various Federal laws and executive orders, including the Atomic Energy Act, the Clean Water Act, the Safe Drinking Water Act, the Resource Conservation and Recovery Act (RCRA), and the Clean Air Act.

Any release of radioactivity (to the atmosphere or to any body of water) must meet EPA's environmental standards for the uranium fuel cycle in 40 CFR 190, which require that "the annual dose equivalent does not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ of the body as the result of exposures to planned discharges of radioactive materials, radon and its daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations" (CFR 1986).

Any release of radioactivity to waters of the United States, including the Susquehanna River, must meet EPA's National Interim Primary Drinking Water Standards in 40 CFR 141 that limit beta particle and photon radioactivity from manmade radionuclides in community water systems to that level which "... shall not produce an annual dose equivalent to the total body or any internal organ greater than 4 millirem/year." This standard applies to concentrations at community water intakes downstream of the discharge point.

Wastes from cleanup of the reactor are not expected to meet the definition of hazardous waste requiring regulation under RCRA. Hazardous wastes are regulated by the EPA under 40 CFR 260-271.

2.3.2 U.S. Nuclear Regulatory Commission Regulations

The NRC regulations in 10 CFR 20, "Standards for Protection Against Radiation," apply to cleanup activities associated with the TMI-2 accident. These regulations implement the EPA standards in 40 CFR 190 and specify allowable discharge concentrations of radioactivity in effluents to air and water in unrestricted areas. Maximum permissible concentrations (MPCs) for isotopes present in the TMI-2 facility are presented in Appendix C of this supplement to the PEIS.

The NRC regulations in 10 CFR 71, "Packaging and Transportation of Radioactive Material," apply to the packaging and shipment of radioactive wastes. Packaging and related requirements are dependent on radionuclide content. U.S. Department of Transportation (DOT) regulations in 49 CFR 170-189 also apply to the packaging, marking and labeling, placarding, monitoring, accident reporting, and shipping papers for radioactive shipments.

Also, NRC regulations in 10 CFR 61, "Licensing Requirements for Land Disposal of Radioactive Waste," apply to the disposal of cleanup wastes in a licensed low-level waste (LLW) burial site. Although these regulations pertain to the licensing, operation, and closing of a low-level commercial waste burial ground, they also contain specifications for the packaging, content, and characteristics of acceptable LLW. For example, liquid wastes must be solidified. Low-level radioactive wastes are classified as Classes A, B, C, or unacceptable, depending on radioactive material content and concentration and on characteristics other than radioactivity.

The NRC regulates the storage of LLW at licensee sites. Because of waste volume limitations of the Low Level Waste Policy Act and its amendments (see Section 2.3.4 for a discussion of these acts), many sites have made provisions for storing LLW for periods beyond those normally required by operational considerations. The NRC has permitted this within carefully controlled limits, but has clarified its policy in Generic Letter 85-14(a), which states: "It is the policy of the NRC that licensees should continue to ship waste for disposal at existing sites to the maximum extent practicable."

2.3.3 U.S. Nuclear Regulatory Commission Proposed Decommissioning Regulations

Although it is not within the scope of this supplement to evaluate decommissioning of the TMI-2 facility, ultimately a decision will need to be made regarding decommissioning or refurbishment of the facility. The NRC has issued proposed decommissioning criteria that were published in the Federal Register on February 11, 1985 (50 FR 5600). The public comment period has ended and final regulations were submitted for Commission approval in December, 1987.

The proposed criteria specifically address three decommissioning alternatives: DECON, SAFSTOR, and ENTOMB.

DECON is the alternative in which equipment, structures, and portions of a facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after cessation of operations.

(a) A letter to all reactor licensees from the NRC, August 1, 1985. Subject: Commercial Storage at Power Reactor Sites of Low-Level Radioactive Waste Not Generated by the Utility.

SAFSTOR is the alternative in which the nuclear facility is placed and maintained in such condition that it can be safely stored, monitored, and subsequently decontaminated (deferred decommissioning) to levels that permit release for unrestricted use. Benefits include occupational exposure or waste volume reduction.

ENTOMB is the alternative in which radioactive contaminants are encased in a structurally long-lived material, such as concrete. The entombed structure is appropriately maintained and continued surveillance is carried out until the radioactivity decays to a level permitting release for unrestricted use of the property.

The licensee's proposal of delayed cleanup (PDMS followed by cleanup) is analogous in many ways to the SAFSTOR decommissioning alternative but differs from it in that a decision to decommission has not been made. The decision as to whether a facility is to be permanently shut down is specifically identified as the licensee's in the supplementary information to the proposed decommissioning rule. Current and proposed regulations do not specify the length of time that a licensee can keep a facility in a shutdown mode and still retain an operating license.

The staff's alternative of immediate cleanup is likewise analogous to the DECON alternative, except that decommissioning is not necessarily the objective of cleanup and cleanup would be considered complete without reducing radiation and contamination to levels that would be acceptable for unrestricted use.

However, TMI-2 is not likely to be a candidate for ENTOMB, because it is likely that there would still be sufficient radioactive material after a period of 100 years to make unrestricted use unacceptable.

The proposed criteria also require that decommissioning plans be submitted within 2 years following a decision by a licensee to permanently cease operations or 1 year before the operating license expires. The TMI-2 licensee has not formally indicated what the disposition of TMI-2 will be. Unless an earlier decision to decommission is made, a preliminary decommissioning plan would be required 5 years before the license expires. The licensee would be required to formulate a complete decommissioning plan 1 year before the license expires.

2.3.4 Low-Level Radioactive Waste Policy Amendments Act of 1985

The Low-Level Radioactive Waste Policy Amendments Act of 1985, H.R. 1083-Public Law 99-240, effectively limits the quantity of low-level radioactive waste that the licensee can dispose of without petitioning the U.S. Secretary of Energy for additional waste disposal capacity. The licensee already has received one such emergency allocation for waste that will result from the proposed disposal of the accident-generated water. Immediate cleanup without PDMS could require additional emergency allocations.

Another provision of the act requires that States, either alone or in regional compacts, develop regional low-level radioactive waste disposal facilities by December 31, 1992. Accordingly, the Commonwealth of Pennsylvania has entered into a regional compact, which has been ratified by Congress. No site for the disposal facility has been selected.

However, for the purpose of this document it is assumed that waste generated before 1992, either from immediate cleanup or from preparation for PDMS, would be shipped to an existing disposal facility. For the purpose of bounding the impact of LLW disposal, a facility near Richland, Washington, was assumed. For waste generated during delayed cleanup following PDMS, a generic site 500 miles (800 kilometers) from TMI was assumed. The lack of a specific site does not hamper this environmental analysis because only the environmental impact of transportation to the site is addressed here. The impact of disposal at the site would be the subject of a separate analysis connected with licensing the site.

2.3.5 Permits

The licensee holds a National Pollutant Discharge Elimination System (NPDES) permit issued by Commonwealth of Pennsylvania, Department of Environmental Resources (PaDER), on September 16, 1986. It covers discharge of nonradioactive pollutants into the Susquehanna River. Any deliberate discharge of water into the Susquehanna River must comply with the provisions of the permit. The NPDES permit limits pH, free chlorine, and heat, and requires monitoring of several other parameters at the primary outfall. Suspended solids, oil, and grease are also limited at other outfalls.

3.0 PROPOSED AND ALTERNATIVE PLANS FOR COMPLETION OF TMI-2 CLEANUP

The licensee's proposal for completing the TMI-2 cleanup and NRC staff-identified alternatives are evaluated in this section. The licensee has proposed to complete the cleanup following a post-defueling monitored storage (PDMS) period (referred to as "delayed cleanup" in this document). Although not specified by the licensee, the NRC staff has evaluated a storage period of 20 years. The principal alternative to this proposal, as identified by the NRC staff, is the continuation and completion of the cleanup at the present level of effort without a storage period (referred to as "immediate cleanup" in this document). This alternative of immediate cleanup (continuing and completing the cleanup without a storage period) was chosen along with delayed cleanup to bound the potential environmental impacts associated with completing cleanup. Five other alternatives to immediate cleanup and delayed cleanup were identified by the NRC staff but not quantitatively evaluated: (1) immediate cleanup with a reduced level of effort, (2) additional cleanup before entering PDMS (i.e., completing a more extensive cleanup than that proposed by the licensee before entering PDMS), (3) delayed cleanup with storage less than 20 years, (4) delayed cleanup with storage longer than 20 years, and (5) no further cleanup following defueling, the "no-action" alternative, which is required by the National Environmental Policy Act (NEPA) of 1969 to be considered as part of all environmental impact statements.

As noted previously, the NRC staff's Programmatic Environmental Impact Statement on the TMI-2 cleanup (PEIS), NUREG-0683 (NRC 1981) presented an evaluation of potential environmental impacts for the entire cleanup process. That evaluation included NRC staff projections for alternative methods that could be used for conducting the cleanup and bounding estimates of the environmental impacts that could result from those alternative methods. Although Supplement 1 to the PEIS (NRC 1984) evaluated the potential radiological impact on the work force of an interim monitored storage/delayed-cleanup alternative similar to the licensee's current PDMS proposal, a complete environmental assessment of the licensee's current proposal has not been prepared by the NRC. Information in this section updates and supplements the evaluation of cleanup alternatives presented in Supplement 1.

To facilitate a comparison of cleanup alternatives, a common starting point and a common ending point were assumed. The common starting point coincides approximately with the end of the current efforts to remove the damaged fuel from the reactor vessel and to ship that material off the TMI site. The current defueling effort is expected to result in the removal and shipment of more than 99 percent of the fuel. Concurrent with the removal of fuel and before the beginning of a monitored storage period, additional activities are expected to be performed, including: (1) decontaminating building and equipment surfaces to levels approximating the licensee's established goals (Table 3.1), (2) packaging and disposing of radioactive wastes associated with decontamination activities, (3) removing the accident-generated water from the reactor building and the AFHB, and (4) quantifying the amount of residual fuel left in the reactor coolant system and the reactor building following the current defueling efforts. The environmental

TABLE 3.1. Licensee's Radiological Goals for the TMI-2 Facility at the End of Defueling(a)

Area	General Area Exposure Rate, mR/h
Reactor Building(b)	
Refueling canal	<15
Elevation 347 ft and above (except D-ring and NW-seal table)	<30
Elevation 347 ft and above	
D-ring	<70
NW-seal table	<70
Elevation 305 ft to 347 ft	<70
Basement (elevation 282 ft to 305 ft)	<35,000
Auxiliary and Fuel Handling Building(b)	
Corridors	<2.5
Other areas	<50
Other Buildings	
Turbine building	<2.5
Chemical cleaning building (except EPICOR II pump area to be left operable)	<2.5
Service building containment Drain tank area	<2.5

(a) Sources: GPU Nuclear Corporation, 1987b; and letter from F. R. Standerfer, GPUN, to the NRC, December 4, 1987. Subject: Post-Defueling Monitored Storage Environmental Evaluation Comment Responses, 4410-87-L-0179/0245P.

(b) The exposure rates given for these buildings refer to the general area and exclude "hot spots" (e.g., the stairwell and elevator shaft in the reactor building basement) and locked, high-radiation areas (e.g., seal injection valve room and makeup and purification demineralizer room).

impacts of these activities, as well as those associated with the disposal of the accident-generated water, have been evaluated in the PEIS and previous supplements (NRC 1981, 1984, and 1987), and will not be reevaluated in this document.

The common endpoint of the alternatives is the completion of the cleanup, as discussed in the PEIS and previous supplements. The NRC staff has consistently characterized the completion of cleanup as follows: (1) building and equipment decontamination to a point where general area dose rates approximate those in a nuclear power plant nearing the end of its

life, (2) fuel removal and decontamination of the reactor coolant system, (3) treatment of radioactive liquid wastes, and (4) packaging of radioactive wastes and shipment of the wastes to an offsite disposal facility. Following the completion of the cleanup, additional activities would be required either to decommission or refurbish the facility. These activities would be the subject of a separate regulatory action by the NRC and are not considered further in this document.

Alternatives that were identified and considered but not quantitatively evaluated, including the "no-action" alternative required by NEPA, are described in Section 3.1 along with the staff's rationale for selection of alternatives for quantitative evaluation. The licensee's proposal for delayed cleanup and the alternative of immediate cleanup are quantitatively evaluated in Sections 3.2 and 3.3, respectively. The evaluations include descriptions of the alternatives and the assessment of the potential environmental impacts, including radiation exposure to the offsite population from routine and accidental releases, occupational radiation dose, waste management impacts (including transportation impacts), socioeconomic impacts, commitment of resources, and regulatory considerations.

3.1 ALTERNATIVES CONSIDERED BUT NOT EVALUATED QUANTITATIVELY

Five alternatives to the licensee's proposal were identified by the NRC staff, but not quantitatively evaluated: (1) immediate cleanup with a reduced level of effort, (2) additional cleanup before PDMS, (3) delayed cleanup with storage less than 20 years, (4) delayed cleanup with storage longer than 20 years, and (5) no further cleanup following defueling, the no-action alternative required by NEPA. These alternatives are described in this section, and the reasons for not evaluating them quantitatively are provided.

3.1.1 Immediate Cleanup with a Reduced Level of Effort

In addition to the immediate cleanup alternative, the staff considered other immediate cleanup alternatives that involved continuing the cleanup after completion of defueling but at a reduced level of effort. This reduced level of cleanup effort would continue through the completion of cleanup. Although the immediate cleanup alternative would involve a continuation of approximately the same level of effort currently being employed to clean up the facility, a spectrum of alternatives involving continuing cleanup efforts at various reduced levels of intensity can also be projected. These alternatives would result in prolonging the cleanup period beyond that of the immediate cleanup alternative. Because the impacts of alternatives involving immediate cleanup at various levels of effort would fall between the impacts of immediate cleanup and those of delayed cleanup, they are not considered further in this document.

3.1.2 Additional Cleanup Before PDMS

The staff has also considered alternatives that vary in the degree to which the facility is decontaminated before being placed into storage. These

alternatives involve pre-PDMS decontamination efforts which further reduce radiation dose rates and radionuclide inventories beyond the licensee's stated goals for PDMS. One such alternative is the prompt completion of cleanup on the upper elevations of the reactor building followed by a monitored storage period; the final stage would be to complete the cleanup of the basement. A second such alternative is prompt cleanup of the upper elevations concurrent with the removal of the enclosed stairwell/elevator structure from the basement; further cleanup of the remaining basement areas would be delayed by a monitored storage period. Because these alternatives are actually combinations of the alternatives of immediate cleanup and delayed cleanup, the environmental impacts of these alternatives would be encompassed by the impacts of immediate cleanup and delayed cleanup. Accordingly, additional cleanup before PDMS is not further considered in this document.

Alternatives that involve significantly less pre-PDMS cleanup than proposed by the licensee appeared to be inferior because of the increased potential for radionuclide mobility and were not considered for detailed evaluation.

3.1.3 Delayed Cleanup with Storage Less Than 20 Years

The staff has also considered alternatives that vary in the length of time the facility is stored before the cleanup is completed. Because the licensee's proposal for PDMS covers an "indefinite" period of time, there is a range of alternatives that differ only in the duration of the storage period. As discussed previously, in order to place reasonable bounds on the alternatives considered, the staff has limited its evaluation of the environmental impact of delayed cleanup to a period of 20 years. The evaluation of immediate cleanup and delayed cleanup with a 20-year storage period effectively encompasses the impact of any storage period lasting between 0 and 20 years. Thus, these alternatives were not quantitatively evaluated.

3.1.4 Delayed Cleanup with Storage Longer Than 20 Years

The staff considered storage periods beyond 20 years, which would require an extension of the facility license (the current license for TMI-2 will expire in 2009). Although longer storage periods are feasible, they are best evaluated on the basis of experience gained from shorter-term storage periods.

3.1.5 No Further Cleanup Following Defueling (the No-Action Alternative)

As noted previously in the PEIS and supplements, the "no-action" alternative must be evaluated to fulfill the requirements of NEPA. The no-action alternative, for the period addressed by this supplement, following completion of defueling implies no further action to complete the cleanup. That is, the facility conditions associated with no action would be essentially the same as those described by the licensee's PDMS proposal, except that neither preparations for PDMS nor subsequent action to finish the cleanup would occur. The NRC staff has maintained, as a matter of policy, that the cleanup must be completed. In the PEIS (NRC 1981), the NRC staff concluded that the no-action alternative was unacceptable because (1) public health and

safety could not be adequately ensured until the radioactive materials decayed to innocuous levels, and (2) the TMI-2 site should not be allowed to become a waste disposal site. The times required for the predominant radionuclides to decay range from a minimum of about 300 years for cesium-137 and strontium-90 to thousands of years for the transuranic radioisotopes. Although substantial progress toward ensuring public health and safety has been made as a result of decontamination measures and defueling, additional actions will be required to decontaminate and to ultimately decommission the facility. Therefore, the NRC staff considers the no-action alternative unacceptable, and it is not evaluated further in this report.

3.2 DELAYED CLEANUP (POST-DEFUELING MONITORED STORAGE FOLLOWED BY COMPLETION OF CLEANUP)

Delayed cleanup, as proposed by the licensee, is described in Section 3.2.1. The offsite dose evaluation is discussed in Section 3.2.2, the occupational dose estimates in Section 3.2.3, the waste management impacts including transportation in Section 3.2.4, the socioeconomic impacts in Section 3.2.5, commitment of resources in Section 3.2.6, and regulatory considerations in Section 3.2.7.

3.2.1 Description of Delayed Cleanup

Delayed cleanup involves preparing the facility for storage, maintaining the facility in the monitored storage mode, and completing the cleanup process at the end of the storage period. The NRC staff has limited its evaluation of the environmental impact of delayed cleanup to a storage period of 20 years. It is expected that within that time period, the licensee will make a decision to begin decommissioning TMI-2 (alone or in combination with TMI-1) or indicate plans for refurbishing the facility. The cleanup process after the storage period would complete the process of decontaminating the facility, removing residual fuel, and disposing of radioactive wastes. The reactor would either be decommissioned or refurbished under a separate regulatory action not covered by the PEIS or the supplements. The specific tasks during final cleanup will depend on the available technology and the planned disposition of the reactor, as noted below.

This section addresses the status of TMI-2 systems during PDMS, preparations required for PDMS, the surveillance and maintenance activities occurring during PDMS, and the additional cleanup that would take place following PDMS.

3.2.1.1 System Status During PDMS

To maintain TMI-2 in a storage mode, the facilities and systems at TMI-2 would be placed into one of four classifications before PDMS: (1) operable for PDMS support, (2) operable for site support, (3) deactivated and preserved for future use, or (4) deactivated but not preserved.

Systems that would remain operable for PDMS support include the ventilation systems in the reactor building and the AFHB and some parts of the water processing systems and the fire protection system. Some of these systems would be modified to support PDMS. For example, fire detection sensors would be operational throughout the plant; however, the remote monitoring capability for the fire protection system, currently located in the TMI-2 control room, may require relocation.

Service facilities outside the protected area fence that are useful for site support would remain operable. Such facilities include the solid waste handling and packaging facility and the laundry/respirator facility. The environmental monitoring program including wells and air monitoring stations would be maintained. Areas within the AFHB that are shared with TMI-1 (e.g., the truck bay) would remain operable, although an identifiable boundary between TMI-1 and TMI-2 would be established and maintained.

Systems that are expected to have a future value to TMI-2, regardless of its disposition, would be deactivated and preserved (preventative maintenance would be applied to protect and preserve the system components). Deactivated and preserved systems include the polar crane and power circuits in the reactor building and the AFHB.

Systems and equipment that would not be needed during the storage period and that would not be expected to have a further value to the facility would be deactivated; however, no action would be taken to ensure their future availability.

In general, aqueous systems, such as the fuel transfer canal, reactor coolant system, and the submerged demineralizer system (SDS), would be drained. However, yard hose stations for fire protection would be capable of being returned to service for emergency use. Filters and demineralizer resin beds would be removed and disposed of. Systems containing residual fuel material, including sections of the reactor coolant system, would be deactivated and sealed as necessary to contain the radioactive material. Noncontaminated systems would be deactivated in a similar manner except that sealing would not be required. Fuel transfer tubes would be sealed to maintain containment integrity. The vessel head would remain at its present shielded storage location. The plenum would be stored dry in the deep end of the fuel transfer canal and shielded to reduce the radiation dose to the surrounding areas. The service structure, defueling platform, and internals indexing fixture would remain in their present locations on the reactor vessel.

3.2.1.2 Preparations for PDMS

Coincident with the completion of defueling, the licensee would prepare the TMI-2 facility for PDMS. The preparations would include modifying, deactivating, and preserving plant systems as discussed in the previous section. It is anticipated that the preparation phase will last between 6 months and 1 year and will take place concurrently with the shipping of fuel and the processing of accident-generated water.

3.2.1.3 Activities During PDMS

During PDMS, the reactor building and the AFHB would be locked; however, periodic entries would be made to inspect, monitor, and maintain the facility. Additional entries would be made in response to emergencies (e.g., fire). Entries might also be made to acquire additional data and plan the future disposition of the facility.

The reactor building would be maintained at atmospheric pressure. Before each entry, the reactor building would be ventilated at a maximum 50,000 cfm (1400 m^3/min) to ensure that the building atmosphere meets personnel protection standards for breathing as well as to ensure that radiation doses would be maintained as low as reasonably achievable (ALARA). The ventilated air would be discharged through double-stage high-efficiency particulate air (HEPA) filters and the discharged air would be monitored. Some passive airflow due to changes in atmospheric pressure (an estimated 10 air exchanges per year in the absence of ventilation) is predicted to occur between active ventilations. Passive airflow would occur through a single-stage HEPA filter.(a) The passive airflow would also be monitored. Passive airflow in the AFHB would also be expected. If necessary, before entries, the AFHB atmosphere would be actively ventilated through HEPA filters.

Inspection and monitoring in both buildings would be performed routinely to identify changes in radiation level, water intrusion, or other off-normal conditions; to verify containment of contamination; and to provide for equipment surveillance as required by the plant's technical specifications. Throughout the storage period, radiological survey results would be collected, reviewed, and evaluated for trends to detect any changes in radiological conditions.

The radiological monitoring would consist of air sampling, loose surface contamination sampling, and radiation dose rate surveys. In addition, thermoluminescent dosimeters could be placed in fixed locations for a period of time and then collected to monitor radiation dose rates. It is anticipated that routine radiological surveys would normally be performed only in areas where personnel access is not limited. The expected radiological conditions in the reactor building would allow regular personnel access for inspection and maintenance at the 305-foot and the 347-foot elevations. Access of limited duration would be allowed to selected areas of the basement. Routine surveys would not normally be performed in areas of high radiation or high contamination, sealed areas, or other normally inaccessible areas unless access were required for some other purpose. Surveys at the boundary of such areas would be performed to ensure containment of contamination.

The licensee's anticipated initial schedule for inspection and monitoring activities is shown in Table 3.2. It is expected that an initial program of data acquisition and assessment would be necessary to ensure that plant

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(a) GPU Nuclear Memorandum from D. R. Buchanan to J. J. Byrne, July 10, 1987. Subject: Reactor Building Breather Passive Air Exchange During PDMS.

TABLE 3.2. Anticipated Initial Schedule for Inspection and Monitoring Activities(a)

Monitoring/Inspection	Frequency	Worker Entry Required
Reactor Building		
Radiological Survey Air sampling Surface contamination surveys Dose rate surveys TLD placement	Monthly	Yes
Visual Surveys General conditions	Monthly	Yes
Sump Level Monitoring	Continuous	No
Fire Detection	Continuous	No
Auxiliary and Fuel-Handling Building		
Radiological Survey Air sampling Surface contamination surveys Dose rate surveys TLD placement	Monthly	Yes
Visual Surveys General conditions Animal intrusion Housekeeping	Monthly	Yes
Sump Level Monitoring	Continuous	No
Fire Detection	Continuous	No

(a) Source: GPU Nuclear Corporation 1987b.

conditions and trends are documented and well understood. During this time, workers would enter the reactor building and AFHB monthly to perform radiological surveys and visual inspections. Abnormal conditions, although not expected, would be investigated and corrected, and the inspection frequency adjusted as appropriate. The inspection and monitoring frequency would be determined by experience and need. The licensee anticipates that the inspection and monitoring frequency might decrease after the first few years if data accumulated from the inspections and surveys indicate that there were no unexpected or adverse changes in building conditions or radiation levels over long periods of time. In addition, the need for pre-entry ventilation of the

reactor building and the AFHB would be evaluated based on the air sampling results.

Maintenance activities would include repair of ventilation systems, changing of filters, and calibration and repair of instrumentation required by the plant's technical specifications. In addition, preventive maintenance of some mothballed equipment is anticipated.

No active program of building or equipment decontamination would be necessary during storage unless radiation surveys indicate that contamination has spread. In these cases, it might be necessary to perform decontamination. In addition, some decontamination might be required to support maintenance or inspection activities. Routine waste processing (solid and liquid) and waste handling and shipping would be performed for those wastes generated as a result of PDMS activities.

Water-processing capabilities would be available to dispose of rainwater inleakage, groundwater inleakage, and condensation (resulting from high humidity conditions). The environmental evaluation written by the licensee(a) indicated that a discharge of 5000 gallons (19,000 liters) annually could be expected during PDMS. This estimate was based on experience and the reduced number of operations occurring during PDMS. Water inleakage is not expected to occur in the reactor building, which is designed to contain radionuclides and prevent inleakage under an extreme variety of environmental conditions. Current experience indicates that any inleakage would occur at the building joint between the service building and the air intake tunnel, at the construction joint in the basement of the AFHB, at the electrical penetration in the southwest corner of the control building (201-foot elevation), and at the fire service penetration on the east wall of the turbine building (300-foot elevation). The licensee indicated that inleakage of groundwater and precipitation are anticipated to be the major sources of liquids during PDMS, although some water used for small decontamination jobs can also be expected. To the extent that the inleakage becomes contaminated by any residual contamination on floors or in sumps, it would be processed before discharge. Decontamination solutions and inleakage would be collected in the auxiliary building sump. Periodically, liquids in the sump that are not directly releasable pursuant to 10 CFR 20, App. B (see Appendix C in this supplement), would be pumped to the auxiliary building sump tank and then to the miscellaneous waste holdup tank (MWHT), or directly from the sump to the MWHT. When the tank was nearly full, the water would be processed through the EPICOR II system, which will be available during PDMS and is located in the chemical cleaning building. The processed water would be sampled and disposed of in accordance with the TMI-2 technical specifications.

The licensee's current environmental monitoring program would continue throughout the storage period. The facility would be continuously monitored

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(a) Letter from F. R. Standerfer to the NRC, March 11, 1987. Subject: Environmental Evaluation for TMI-2 Post-Defueling Monitored Storage, Document ID 0161P.

for possible effluent releases. The offsite environmental monitoring program would also be continued pursuant to the technical specifications. Groundwater monitoring would be performed quarterly.

3.2.1.4 Cleanup Following PDMS

By the end of PDMS, it is expected that the licensee will have made a decision on the future disposition of the plant and the final cleanup will be performed along with either refurbishment or decommissioning. However, for the purposes of this supplement, it is assumed that the facility will be cleaned to levels expected in an operating reactor at the end of its life before decommissioning or refurbishment begins. The licensee has not formulated plans for reactor building cleanup following PDMS. Any such plans would be tentative because of (1) limited knowledge of robotic capabilities and other technological advances that will be available, (2) incomplete information (although currently being obtained) on the amount and location of contamination, and (3) the absence of a decision on the disposition of the facility. However, the NRC staff has considered the principal activities during cleanup following PDMS to include decontamination of the reactor coolant system and general cleanup of the reactor building, especially the basement.

The cleanup processes are assumed to be similar to those projected by the staff in evaluating the immediate cleanup alternative in Section 3.3.1. The assumed differences are as follows: (1) a full 4 years would be necessary for cleanup and would include the time required to assemble a work force and educate them regarding conditions in the facility, (2) modest advances in robotic technology would have occurred during the intervening period, (3) radiation dose rates would be somewhat lower because of radioactive decay, and (4) a regional repository within 500 miles (800 kilometers) of the site would be available to accept the waste.

3.2.2 Offsite Dose Evaluation for Delayed Cleanup

The evaluation of radiation dose to the offsite population as a result of the delayed cleanup alternative includes an assessment of the dose from both routine releases and potential accidental releases of radioactive material.

3.2.2.1 Routine Releases

The magnitude of the routine release of radioactive material will vary depending on the stage of the cleanup. These stages as described in Section 3.2.1 include (1) preparations for PDMS, (2) PDMS, and (3) cleanup following PDMS.

Preparations for PDMS. The preparations to place the TMI-2 facility into a PDMS mode are expected to take place concurrent with the completion of defueling. The activities to prepare the facility for PDMS are not expected to increase the amount of airborne or waterborne contamination. Thus, the routine releases that would be expected to result from preparations to place the facility in a PDMS mode would not be distinguishable from releases expected during the final stage of defueling or from releases

currently occurring. Current releases are shown in Table 3.3 for the period January 1 to June 30, 1987. Airborne discharges during this period were less than 0.03 percent of the technical specification limits. Liquid discharges during this same time period were less than 0.00002 percent of the applicable limits. These release rates and quantities are consistent with results reported for previous calendar quarters.(a)

TABLE 3.3. Airborne and Liquid Discharges During the Period January 1 to June 30, 1987(a)

		Activity Released, μCi	
	Radionuclide	January 1 to March 31, 1987	April 1 to June 30, 1987
Atmosphere	Tritium	11,000,000	15,000,000
	Gross alpha	0.002	0.003
	Unidentified beta radiation	46	3
	Cesium-137	8	<LLD(b)
Liquid	Tritium	4	360
	Strontium-90 and unidentified beta radiation	26	19
	Cesium-134	<LLD(b)	0.1
	Cesium-137	9	10

(a) Source: Letter from F. R. Standerfer to the NRC, August 28, 1987. Subject: Quarterly Dose Assessment Report - Second Quarter 1987; Semi-Annual Radioactive Effluent Release Report, Document ID 0080P.
(b) Less than the lower limit of detection.

During PDMS. Routine releases of radionuclides during PDMS are expected by both atmospheric and liquid pathways.

Atmospheric Releases. As noted in Section 3.2.1, the reactor building atmosphere would be ventilated through double-stage HEPA filters before each entry. Entries may occur as frequently as once a month. The amount of radioactivity released during ventilation is based on an estimate of the fraction of radioactive material on surfaces in the reactor building that could become suspended in the reactor building atmosphere. Three major sources of potentially suspendable contamination are identified based on the information presented in Chapter 2: (1) the enclosed stairwell/elevator structure, (2) the sludge residue on the reactor building basement floor (which may have a greater potential for mobility as it dries during the PDMS period), and (3) the remaining surface contamination on the concrete slab walls, equipment, overhead structures, etc.

(a) Attachment 1 of letter from F. R. Standerfer to the NRC, August 28, 1987. Subject: Quarterly Dose Assessment Report - Second Quarter 1987; Semi-Annual Radioactive Effluent Release Report, Document ID 0080P.

It was conservatively assumed that a fraction of the radionuclides absorbed within the stairwell/elevator structure would migrate to the surface of the concrete block as the structure dried. Studies indicate that cesium migration occurs to some extent as concrete dries (Arora and Dayal 1986). To conservatively bound this phenomenon, one-eighth (approximately 13 percent) of all radioactive material in the structure was assumed to be available for suspension.(a) Although studies with strontium (Arora and Dayal 1986) indicate that it does not migrate as easily as cesium, it is conservatively assumed that one-eighth of the strontium-90, as well as all other isotopes assumed to have been dispersed through the reactor building and present in the concrete block, migrate near to the surface and are available for suspension in the reactor building atmosphere.

Because the suspension of radioactive material from the dried sludge in the reactor building basement has not been investigated, the staff has conservatively assumed that 100 percent of the radioactive material in the sludge [including the 7.1 pounds (3.2 kilograms) of fuel debris assumed to be present on the basement floor] would be available for suspension in the reactor building atmosphere. In addition, one-tenth of the radioactive material in the concrete slab walls, equipment, overhead structures, etc., is assumed to be near the surface and available for resuspension.

A resuspension factor [the ratio of air contamination ($\mu Ci/m^3$) to the surface contamination ($\mu Ci/m^2$)] was used to estimate the amount of surface contamination that may become airborne. Resuspension factors quoted by the International Atomic Energy Agency (IAEA) in Technical Report Series No. 20 vary from 0.000002/meter to 0.003/meter (Clayton 1970). Dunster (1962) indicates that "for controlled areas the lower figure of 0.000002/meter is certainly safe for long term use." Because there will be little or no traffic in the reactor building during PDMS (especially in the basement where most of the contamination is located) and no forced ventilation (except before worker entries), the lower figure was used and conservatively applied to the entire air volume of the basement.

For forced ventilation, double-stage HEPA filters will be used to remove particulate radioactive material. The fraction of the radioactive particulate material that penetrates a single-stage HEPA filter is conservatively assumed to be 0.01 (NRC 1979b). For double-stage HEPA filters, this fraction

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(a) This fraction is based on the conservative assumption that the activity in the first 1/2 inch (1.3 centimeters) of the concrete block becomes available for resuspension over time. Because much of the concrete block is available to the atmosphere on two sides, 1/2 inch (1.3 centimeters) on each side accounts for one-eighth of the activity in the structure.

is assumed to decrease to 0.0001.(a) In addition to the 12 forced ventilation releases assumed each year, the air in the reactor building is expected to passively exchange to some degree with the outside air because of changes in atmospheric pressure. As discussed in Section 3.2.1.3, an estimated 10 passive air exchanges a year would occur during the time between active ventilation. Passive air exchange would occur through a single-stage HEPA filter and would be monitored continuously. A penetration factor of 0.01 was used for the single-stage HEPA filters during these releases.

The amount of radioactive material assumed to be released annually into the atmosphere during the PDMS period is shown in Table D.1 of Appendix D for the first year of release. Releases in subsequent years are based on the releases during the first year and account for radioactive decay.

The 50-year dose commitment(b) to the maximally exposed member of the public, from inhalation, consumption of food products, and external exposure as a result of routine atmospheric releases during the PDMS period, is shown in Table 3.4. The maximally exposed individual is assumed to breathe air at the offsite boundary location of highest airborne concentration [0.34 miles (0.55 kilometers) west] and to consume food products raised exclusively in the offsite boundary location that receives the maximum ground deposition of the released radioactive material. The maximally exposed individual is in the age group that receives the highest dose. The collective 50-year dose commitment to the estimated 2.2 million to 3.2 million people living within a 50-mile (80-kilometer) radius between 1989 and 2009 from inhalation, consumption of food products, and external exposure is also shown in Table 3.4. The dose attributable to TMI-2 received by the population (of unspecified size) outside the 50-mile (80-kilometer) radius from inhalation, external exposure, and consumption of food products exported from within the 50-mile (80-kilometer) radius is also shown in Table 3.4. The collective dose to the population and the dose to the maximally exposed individual are calculated for the entire 20-year period for PDMS. The bases for the calculations are given in Appendix E.

(a) Forced ventilation will be through two HEPA filters in series. Each has an in-place tested efficiency of at least 99.95 percent for removal of particulates of 0.3-micron (0.0003-millimeters) diameter. Therefore, only a fraction, 0.0005, of the particulates in the building atmosphere would pass through the first stage and the same fraction (0.00000025 of the initial particulates) would pass through the second stage to the atmosphere. However, Regulatory Guide 1.140 (NRC 1979b), which gives guidelines for operating nuclear power plants, specifies a very conservative penetration factor of 0.01 (corresponding to 99-percent efficiency) for filtration systems that test, in place, to an efficiency of 99.95 percent or more. Although the regulatory guide gives no additional credit for HEPA filters in series, because of the extensive conservatism, the penetration fraction through each stage of HEPA filters was assumed to be 0.01, thus giving an overall penetration factor of 0.0001.

(b) The 50-year dose commitment is the total radiation received from the initial exposure through the succeeding 50 years.

TABLE 3.4. 50-Year Dose Commitments from Routine Atmospheric Releases During PDMS

	50-Year Dose Commitment from a 20 Year Period of Releases	
	Critical Organ (Bone)	Total Body
Maximally exposed individual	25 mrem	3 mrem
Total population within a 50-mile radius(a)	14 person-rem	9 person-rem
Total population outside the 50-mile radius(b)	2 person-rem	0.4 person-rem

(a) Approximately 2.2 million to 3.2 million people between 1989 and 2009.

(b) A population of unspecified size that receives radiation dose from external exposure, inhalation, and the consumption of food products exported from within the 50-mile (80-kilometer) radius.

Liquid Releases. The evaluation of offsite doses resulting from the routine liquid releases is based on 5000 gallons (19,000 liters) of groundwater, precipitation inleakage, and small amounts of decontamination liquids released each year, after they are processed through the EPICOR II system. Although actual concentrations of radionuclides in the water after treatment by the EPICOR II system cannot be determined at this time, it is conservatively estimated that they would be comparable to the concentrations given in Supplement 2 (Table 2.2) for 100-percent processing of the accident-generated water through the SDS and/or EPICOR II systems. The radionuclides of interest are those that were dispersed through the reactor building during the accident (see Table 2.4) with the exception of krypton-85 (a gas) and tritium, which is expected to have been essentially removed from the facility before the start of PDMS (see Section 2.2.2.2). For those radionuclides assumed to be dispersed through the facility but not detected in the accident-generated water after processing, a concentration equal to the lower limit of detection was assumed. Radionuclides specifically associated with the fuel debris were not considered because it is assumed that they would be isolated in the reactor coolant system piping, or located in the reactor building basement, which is not expected to receive any inleakage.

The amount of radioactive material assumed to be released annually in liquid releases during the PDMS period is shown in Table D.2 of Appendix D.

The 50-year dose commitment to the maximally exposed individual is based on the release of 5000 gallons (19,000 liters) to the Susquehanna River each year for 20 years, and is shown in Table 3.5. The maximally exposed individual is a member of the public who drinks Susquehanna River water, consumes fish inhabiting the river, and participates in rivershore activities. In addition, this individual is assumed to consume shellfish from Chesapeake Bay at a maximum rate of shellfish consumption for the

TABLE 3.5 50-Year Dose Commitments from Routine Liquid Releases to the Susquehanna River During PDMS

	50-Year Dose Commitment from a 20-Year Period of Releases	
	Critical Organ (Bone)	Total Body
Maximally Exposed Individual	mrem	mrem
Consumption of Susquehanna River water and fish/ participation in rivershore activities	0.01	0.003
Consumption of Chesapeake Bay shellfish	0.001	0.00007
Total Population	person-rem	person-rem
Consumption of Susquehanna River water and fish/ participation in rivershore activities(a)	0.2	0.004
Consumption of Chesapeake Bay shellfish(b)	0.7	0.04

(a) Approximately 2.2 million to 3.2 million persons between 1989 and 2009. An estimated 300,000 to 440,000 persons were assumed to obtain their drinking water from the Susquehanna River downstream of TMI-2.

(b) A population of unspecified size that consumes shellfish from Chesapeake Bay. Approximately 12 percent of the dose from the consumption of Chesapeake Bay shellfish is received by persons within 50 miles (80 kilometers) of TMI-2.

mid-Atlantic region, 97 pounds per year or 44 kilograms per year (Rupp, Miller and Baes 1980). The collective 50-year dose commitment to the population within a 50-mile (80-kilometer) radius (an estimated 2.2 to 3.2 million people between 1989 and 2009) that drinks Susquehanna River water (an estimated 300,000 to 440,000 people living downstream from TMI are assumed to obtain their drinking water from the Susquehanna River), consumes fish inhabiting the river, and participates in swimming, boating, and rivershore activities is also shown in Table 3.5. The dose expected from the consumption of shellfish from the Chesapeake Bay by the population (of unspecified size) that consumes shellfish is also shown. A fraction of this dose (approximately 12 percent) is assumed to be received by the persons within the 50-mile (80-kilometer) radius that consume Chesapeake Bay shellfish. The remainder is received by persons outside the 50-mile (80-kilometer) radius. The bases for these calculations, including the flow rate assumed for the Susquehanna River, are given in Appendix E.

Cleanup Following PDMS. Routine releases of radionuclides during the cleanup following PDMS are expected by atmospheric and liquid pathways.

Atmospheric Releases. The routine airborne releases during the cleanup period following PDMS are expected to be similar to those occurring during the defueling period and the preparations for PDMS. However, aggressive decontamination efforts that may occur during certain cleanup activities could result in an increased release of radioactive material. Aggressive decontamination includes mechanical decontamination operations such as those that would likely occur in the basement during the decontamination or removal of the concrete-block stairwell/elevator structure. These operations may increase the amount of activity in the reactor building atmosphere, thus increasing the amount of activity released from the facility. However, 20 years of radioactive decay would have reduced the amount of radioactive material in the facility and some isotopes would have decayed to negligible amounts (for instance, manganese-54, cerium-144, and promethium-144). In addition, improved techniques and equipment would likely be available for decontamination work to further reduce the potential for airborne contamination.

To estimate radionuclide releases into the atmosphere during the cleanup period following PDMS, it is assumed that some of the radionuclides in the reactor building would become airborne during decontamination processes and a fraction of these radionuclides would escape into the atmosphere through the double-stage, HEPA filtered, ventilation system as described earlier. During decontamination of the reactor coolant system, it is assumed that 10 percent of the activation products and 10 percent of the fuel debris distributed throughout the piping of the reactor coolant system could potentially be suspended in the reactor building atmosphere during a 1-year period. During activities such as cleanup of the reactor building basement, it was conservatively assumed that 100 percent of the radionuclides dispersed throughout the facility (and mostly found in the reactor building basement), including the 7.1 pounds (3.2 kilograms) of fuel assumed to remain on the reactor building basement floor, would be available for suspension for a period of approximately 1 year. During the remaining 3 years, airborne releases are expected to be substantially smaller.

To ensure a conservative approach to calculating the offsite radiation dose from the cleanup period following PDMS, airborne effluents were based on a release rate 2 orders of magnitude (100 times) larger than the first quarter 1987 release rates shown in Table 3.3 for particulates (unidentified beta, cesium, and alpha). This release rate was reduced to account for 20 years of radioactive decay. The quantity of each radionuclide assumed to be available for suspension in the reactor building was used to determine the quantity released from the facility by scaling to the particulate release rate. The calculated release rates were assumed to occur over the entire 4-year cleanup period, even though releases would decrease substantially after the decontamination of the reactor coolant system and decontamination and/or removal of the concrete-block stairwell/elevator structure. The annual release rates assumed for atmospheric releases during the cleanup period following PDMS are shown in Table D.3 of Appendix D.

Table 3.6 shows the estimated 50-year dose commitment to the maximally exposed individual (described previously), the collective 50-year commitment to the population within a 50-mile (80-kilometer) radius (with a projected population of 3.2 million people in the year 2009), and the collective 50-year dose commitment to the population outside the 50-mile (80-kilometer) radius that receives radiation dose (attributable to TMI-2) from external exposure, inhalation, and the consumption of food products exported from within the 50-mile (80-kilometer) radius. The dose to the maximally exposed individual and the collective dose to the population were calculated for the entire 4-year period assumed for the cleanup following PDMS.

TABLE 3.6. 50-Year Dose Commitments from Routine Atmospheric Releases During the Cleanup Following PDMS

	50-Year Dose Commitment from a 4-Year Period of Releases	
	Critical Organ (Bone)	Total Body
Maximally exposed individual	4 mrem	0.5 mrem
Total population within a 50-mile radius(a)	3 person-rem	2 person-rem
Total population outside the 50-mile radius(b)	0.1 person-rem	0.04 person-rem

(a) Projected 3.2 million persons in 2009.
(b) A population of unspecified size that receives radiation dose from external exposure, inhalation, and the consumption of food products exported from within the 50-mile (80-kilometer) radius.

Liquid Releases. Liquid releases to the Susquehanna River will also occur during the 4-year period expected for the cleanup following PDMS. The liquids will be largely composed of water used for decontamination, for instance, during the flushing and decontamination of the reactor coolant system and the reactor coolant drain tank, as well as during the removal of contamination from other areas of the facility. Before release the liquids would be processed through the EPICOR II system. Maximum releases of 250,000 gallons (950,000 liters) a year were assumed, based on information given in the PEIS. It is assumed that all radionuclides listed in Table 2.4 [with the exception of krypton-85, a gas, and tritium, which is expected to be essentially removed from the facility before the start of PDMS (see Section 2.2.2.2)] could be present in the liquid after processing the water through the SDS and/or the EPICOR II system. Although the actual concentration of each radionuclide after treatment by the EPICOR II system cannot be determined at this time, it was conservatively estimated that they would be comparable to the concentrations given in Supplement 2 (Table 2.2) for processing 100 percent of the accident-generated water. For those radionuclides not detected in the accident-generated water, the lower limit of detection was assumed. The amount of radioactive material assumed to be

released annually in liquid releases during the cleanup period following PDMS is shown in Table D.4 of Appendix D.

The resulting dose estimates are shown in Table 3.7. The dose estimates include a 50-year dose commitment to the maximally exposed individual (described previously), the collective 50-year dose commitment to the population within a 50-mile (80-kilometer) radius of TMI (a projected population of 3.2 million people in the year 2009, with an estimated 440,000 people obtaining their drinking water from the Susquehanna River), as well as the population that consumes shellfish from the Chesapeake Bay. These dose estimates were calculated for the entire 4-year period assumed for cleanup following PDMS.

TABLE 3.7. 50-Year Dose Commitments from Routine Liquid Releases to the Susquehanna River During the Cleanup Following PDMS

	50-Year Dose Commitment from a 4-Year Period of Releases	
	Critical Organ (Bone)	Total Body
Maximally Exposed Individual	mrem	mrem
Consumption of Susquehanna River water and fish/ participation in rivershore activities	0.1	0.03
Consumption of Chesapeake Bay shellfish	0.01	0.0007
Total Population	person-rem	person-rem
Consumption of Susquehanna River water and fish/ participation in rivershore activities(a)	2	0.05
Consumption of Chesapeake Bay shellfish(b)	8	0.5

(a) A projected population of 3.2 million persons in 2009. An estimated 440,000 persons were assumed to obtain their drinking water from the Susquehanna River downstream of TMI-2.

(b) A population of unspecified size that consumes shellfish from Chesapeake Bay. Approximately 12 percent of the dose from the consumption of Chesapeake Bay shellfish is received by persons within 50 miles (80 kilometers) of TMI-2.

3.2.2.2 Accidental Releases

The potential for accidents resulting in airborne releases of radionuclides during delayed cleanup was evaluated. Three potential accidents resulting in an atmospheric release were developed from the list of potential accidents given in the PEIS: a fire in the stairwell/elevator structure, the rupture of a HEPA filter during decontamination activities, and the spill of decontamination solution in the reactor building.(a) A single accident resulting in a liquid release was identified for evaluation. This accident involved the rupture of a tank containing liquid that had not been completely treated to remove radioactive material. These four accidents were evaluated to determine their effect on the offsite population.

Preparations for PDMS. The potential for accidental releases during preparations for PDMS is expected to be similar to the accident potential during defueling, which was evaluated in the PEIS. The preparations to place the TMI-2 facility into PDMS are similar and actually a continuation of current cleanup activities and are not expected to increase the potential for releasing airborne or waterborne contamination even in the event of an accident.

During PDMS. The potential for both accidental atmospheric releases and accidental liquid releases was evaluated for the PDMS period.

Accidental Atmospheric Releases. The fire in the stairwell/elevator structure was identified as the only accident that could occur during PDMS that would result in an atmospheric release of radionuclides. The accident scenario involving a fire was evaluated although a fire is considered unlikely during PDMS, since combustible materials and ignition sources are not expected to be present. It was assumed that the accident would occur early in the storage period, before appreciable decay of the radionuclides occurred. For the purposes of this analysis, it was conservatively assumed that 20 percent of the stairwell/elevator structure below the 8-foot (2.4-meter) mark would be involved in the fire. Although the contamination in the structure is not distributed uniformly, it was conservatively assumed that 20 percent of the activity in the stairwell/elevator structure would be involved in the fire. It was further assumed that the 7.1 pounds (3.2 kilograms) of fuel debris thought to remain on the floor of the basement after desludging would also be involved in the fire (even though desludging has occurred in the area of the stairwell/elevator structure and measurements taken before desludging indicated that fuel debris is not located near the stairwell/elevator structure). The fraction of activity to be released into the reactor building atmosphere during the burning of the contaminated material was assumed to be 0.0005, based on

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(a) Recriticality was not considered as a credible accident. Most of the remaining fuel debris would be sealed in piping or enclosed in components located in the reactor building. The amount of fuel debris remaining, its dispersed distribution, and the lack of a moderator would preclude criticality during the storage period. Further, during cleanup, precautions would be taken to ensure that recriticality would not occur.

research by Mishima and Schwendiman (1973). The amount released from the building would be further reduced because the HEPA filters would remove at least 99 percent of the radioactive particulates. The fraction of the radioactive particulate material that would penetrate the single-stage HEPA filter used when the reactor building was secured but not actively ventilated was conservatively assumed to be 0.01 (NRC 1978).

The amount of radioactive material assumed to be released during this accident is shown in Table D.5 of Appendix D. Table 3.8 shows the estimated 50-year dose commitment to the maximally exposed individual (as described in Section 3.2.2.1), the collective 50-year dose commitment to the 2.2 million people within a 50-mile (80-kilometer) radius, and the collective 50-year dose commitment to the population outside the 50-mile (80-kilometer) radius that receives radiation dose (attributable to TMI-2) from external exposure, inhalation, and the consumption of food products exported from within the 50-mile (80-kilometer) radius.

TABLE 3.8. 50-Year Dose Commitments from an Accidental Atmospheric Release (Fire in the Stairwell/Elevator Structure) During PDMS

	50-Year Dose Commitment	
	Critical Organ (Bone)	Total Body
Maximally exposed individual	2 mrem	0.2 mrem
Total population(a) within a 50-mile radius	1 person-rem	0.6 person-rem
Total population outside the 50-mile radius(b)	0.1 person-rem	0.04 person-rem

(a) Projected 2.2 million persons in 1989.
(b) A population of unspecified size that receives radiation dose from external exposure, inhalation, and the consumption of food products exported from within the 50-mile (80-kilometer) radius.

Accidental Liquid Releases. During PDMS, water-processing capabilities would be available to dispose of the small amount of liquid [assumed to be 5000 gallons (19,000 liters)] produced by inleakage, condensation, and small amounts of decontamination. Liquids that are not directly releasable pursuant to 10 CFR 20, Appendix B, Table II, would be collected in the MWHT and then transferred to the chemical cleaning building and processed through the EPICOR II system before final sampling and discharge. Based on the environmental assessment prepared by the NRC staff on the use of the EPICOR II system at TMI-2 (NRC 1979c), there are no credible accidents that would result in a liquid release during the transfer or processing of the PDMS-produced liquids. The operating history of this system in the intervening time has not altered this conclusion. In addition, any leakage from the MWHT in the AFHB would be contained in the AFHB.

Cleanup Following PDMS. The potential for both accidental atmospheric releases and accidental liquid releases was evaluated for the cleanup period following PDMS.

Accidental Atmospheric Releases. All three of the potential accidents for atmospheric releases that were identified earlier could result in atmospheric releases during the cleanup period following PDMS. The analysis of the potential effect of a fire in the stairwell/elevator structure was based on assumptions similar to those given above during PDMS. However, the amount of activity assumed to be present is less because of the 20-year period of radioactive decay. In addition, a double-stage HEPA filter would be used; thus, the fraction of radioactive material released from the reactor building atmosphere was conservatively assumed to be 0.0001. The amount of radioactive material assumed to be released during this accident is shown in Table D.6 of Appendix D. Table 3.9 shows the estimated 50-year dose commitment to the maximally exposed individual (described in Section 3.2.2.1) from this release, the collective 50-year dose commitment to the 3.2 million people within a 50-mile (80-kilometer) radius, and the collective 50-year dose commitment to the population outside the 50-mile (80-kilometer) radius that receives radiation dose (attributable to TMI-2) from external exposure, inhalation, and the consumption of food products exported from within the 50-mile (80-kilometer) radius during 2009.

High-efficiency particulate air (HEPA) filters may fail because of physical damage such as puncture, because of extreme pressure differentials, and because of water damage over a long period of time. For this reason, periodic in-place testing is required; however, for the purposes of accident analysis, the failure of a HEPA filter was assumed to occur at the most critical time during the cleanup process, when the largest amount of airborne

TABLE 3.9. 50-Year Dose Commitments from an Accidental Atmospheric Release (Fire in the Stairwell/Elevator Structure) During Cleanup Following PDMS

	50-Year Dose Commitment	
	Critical Organ (Bone)	Total Body
Maximally exposed individual	0.01 mrem	0.001 mrem
Total population within a 50-mile radius(a)	0.007 person-rem	0.004 person-rem
Total population outside the 50-mile radius(b)	0.0003 person-rem	0.0001 person-rem

(a) Projected 3.2 million persons in 2009.
(b) A population of unspecified size that receives radiation dose from external exposure, inhalation, and the consumption of food products exported from within the 50-mile (80-kilometer) radius.

contamination would be present in the reactor building. This was assumed to be during the postulated demolition of the stairwell/elevator structure. Although it is expected that precautions would be taken to minimize airborne contamination, a fraction of the radionuclide inventory (0.01 percent) was assumed for this analysis to become uniformly dispersed in the reactor building air. A failure of the HEPA filters in one of the ventilation trains would be discovered because of the increased radiation levels recorded by the ventilation stack monitor and the ventilation would be closed off or diverted to the other ventilation train. However, assuming a maximum ventilation rate of 25,000 cfm (710 m^3/min) and a 10-minute interval between failure and corrective action, an estimated 250,000 cubic feet (7100 cubic meters) of air would have been ventilated with a fraction (0.125) of the airborne activity that would be suspended in the reactor building. The maximum amount of radioactive material estimated to be released during this type of accident is shown in Table D.7 of Appendix D. Table 3.10 shows the estimated 50-year dose commitment to the maximally exposed individual of the public (as described in Section 3.2.2.1), the collective 50-year dose commitment to the projected 3.2 million people within a 50-mile (80-kilometer) radius resulting from this release, and the collective 50-year dose commitment to the population outside the 50-mile (80-kilometer) radius that receives radiation dose (attributable to TMI-2) from external exposure, inhalation, and the consumption of food products exported from within the 50-mile (80-kilometer) radius.

The consequences of an atmospheric release resulting from an accidental spill inside the reactor building of decontamination solution from the reactor coolant system were discussed in the PEIS. The consequences are evaluated in this report based on the quantity of radionuclides (activation products and radionuclides associated with fuel debris) assumed to remain in the reactor coolant system after the end of the PDMS period. For this evaluation, it was assumed that during the decontamination process, 10 percent of the maximum possible amount of activity in the untreated

TABLE 3.10. 50-Year Dose Commitments from an Accidental Failure of a HEPA Filter During Cleanup Following PDMS

	50-Year Dose Commitment	
	Critical Organ (Bone)	Total Body
Maximally exposed individual	12 mrem	1 mrem
Total population within a 50-mile radius(a)	8 person-rem	5 person-rem
Total population outside the 50-mile radius(b)	0.4 person-rem	0.1 person-rem

(a) Projected 3.2 million persons in 2009.
(b) A population of unspecified size that receives radiation dose from external exposure, inhalation and consumption of food products exported from within the 50-mile (80-kilometer) radius.

decontamination solution could be spilled before corrective action would be taken. Of this 10 percent, 0.1 percent of the spilled activity was assumed to become airborne. The fraction of the airborne radioactive material that would penetrate the double-stage HEPA filters was conservatively assumed to be 0.0001. The amount of radioactive material assumed to be released during this type of accident is shown in Table D.8 of Appendix D. Table 3.11 shows the estimated 50-year dose commitment to the maximally exposed individual (described in Section 3.2.2.1), the collective 50-year dose commitment to the projected 3.2-million people within a 50-mile (80-kilometer) radius, and the collective 50-year dose commitment to the population outside the 50-mile (80-kilometer) radius that receives radiation dose (attributable to TMI-2) from external exposure, inhalation, and consumption of food products exported from within the 50-mile (80-kilometer) radius.

TABLE 3.11. 50-Year Dose Commitments from an Accidental Spill of Reactor Coolant System Decontamination Solution During Cleanup Following PDMS

	50-Year Dose Commitment	
	Critical Organ (Bone)	Total Body
Maximally exposed individual	0.005 mrem	0.0001 mrem
Total population within a 50-mile radius(a)	0.009 person-rem	0.0005 person-rem
Total population outside the 50-mile radius(b)	0.00008 person-rem	0.000004 person-rem

(a) Projected 3.2 million persons in 2009.
(b) A population of unspecified size that receives radiation dose from external exposure, inhalation, and consumption of food products exported from within the 50-mile (80-kilometer) radius.

Accidental Liquid Releases. Although the licensee has not made any detailed plans for the cleanup following PDMS, it is assumed that during the cleanup, the contaminated liquids would be processed through the SDS and could potentially be stored in an outside 11,000-gallon (42,000-liter) storage tank before being processed through the EPICOR II system. The assumed pathway for an accidental waterborne release involves the rupture of an 11,000-gallon (42,000-liter) storage tank of unprocessed water with the entire inventory released to the Susquehanna River. Supplement 2 evaluated this accident, assuming that the entire inventory of the tank spilled directly into the river even though it was considered unlikely that more than a few thousand gallons would reach the Susquehanna River via normal rainwater runoff channels. The concentration of each radionuclide that could be in the water (based on the list in Table 2.4) was conservatively estimated to be comparable to the concentration given in Table 2.2 of Supplement 2 (for the case where 40 percent of the total stored accident-generated water had been processed through the SDS and EPICOR II systems). The exceptions are

krypton-85 (a gas) and tritium, which is expected to be essentially removed from the facility before the start of PDMS (see Section 2.2.2.2). The lower limit of detection was assumed for those radionuclides not detected in the accident-generated water. The amount of radioactive material assumed to be released during this accident is shown in Table D.9 of Appendix D. Table 3.12 shows the estimated 50-year dose commitment to the maximally exposed individual (described in Section 3.2.2.1), and the collective 50-year dose commitment to the population downstream of TMI-2 (a projected population of 3.2 million people in the year 2009, with an estimated 440,000 people obtaining their drinking water from the Susquehanna River) as well as to the population that consumes shellfish from the Chesapeake Bay.

TABLE 3.12. 50-Year Dose Commitments from an Accidental Liquid Release to the Susquehanna River During Cleanup Following PDMS

	50-Year Dose Commitment	
	Critical Organ (Bone)	Total Body
Maximally Exposed Individual	mrem	mrem
Consumption of Susquehanna River water and fish/ participation in rivershore activities	0.01	0.002
Consumption of Chesapeake Bay shellfish	0.001	0.00003
Total Population	person-rem	person-rem
Consumption of Susquehanna River water and fish/ participation in rivershore activities(a)	0.3	0.006
Consumption of Chesapeake Bay shellfish(b)	0.8	0.02

(a) A projected population of 3.2 million persons in 2009. An estimated 440,000 persons were assumed to obtain their drinking water from the Susquehanna River downstream of TMI-2.

(b) A population of unspecified size that consumes shellfish from Chesapeake Bay. Approximately 12 percent of the dose from consumption of Chesapeake Bay shellfish is received by persons within 50 miles (80 kilometers) of TMI-2.

3.2.3 Occupational Radiation Dose Evaluation for Delayed Cleanup

The occupational radiation dose to place the TMI-2 facility in a PDMS mode, maintain the facility in that mode, and then complete cleanup is estimated to be between 48 to 1500 person-rem, as shown in Table 3.13. This dose is in addition to the occupational radiation dose already received and that required to complete defueling.

The estimate presented in Table 3.13 is based on a task-by-task analysis of the work to be done. It is presented as a range of values because of the uncertainties in the cleanup process and the technology that will be available when poststorage cleanup is performed. The range of values occurs because of uncertainties in the location and depth of penetration of the contamination, the robotic technology that will be available, and the approach to cleanup that will be taken. For example, it is not known if workers would need to enter the basement during decontamination, and if waste would have to be manually packaged before or after it is removed from the basement.

This estimate is somewhat lower than the estimate for delayed cleanup involving an interim monitored storage phase, which was described in Supplement 1 to the PEIS. The principal reason is that the estimates in Supplement 1 did not include the use of robotics before storage. However, robotics currently are being used effectively by the licensee in desludging and scabbling the basement; continued use is likely.

TABLE 3.13. Occupational Radiation Dose Estimate for Delayed Cleanup

Task Description	Worker Dose Range, person-rem
Pre-PDMS preparations	2.0 to 20
Maintain facility in PDMS mode(a)	9.0 to 65
Reactor disassembly and defueling	0.8 to 8.0
Reactor coolant system decontamination	17 to 460
Reactor building basement cleanup	19 to 920
Other reactor building cleanup	1.5 to 56
Total(b)	48 to 1500

(a) Does not include the dose to make inspections and evaluations in order to plan post-PDMS work.
(b) The totals may not be exact because of rounding.

3.2.4 Waste Management Considerations of Delayed Cleanup

The quantity, radiation level, and classification of waste that would be produced by delayed cleanup have been evaluated based on current regulatory requirements. Preparation for PDMS would generate additional compacted, dry radioactive waste, which would all be Class A waste as defined by 10 CFR 61 (see Section 2.3.2 and Appendix F for a discussion of waste classification). Maintenance of the TMI-2 facility in the PDMS configuration could generate waste consisting of HEPA filters and disposable protective clothing. Treatment of water and decontamination solutions would generate additional waste that could be Class A, B, or C. However, the quantities would be rather small and it is expected that they would be shipped offsite as they were generated. Table 3.14 shows the estimated range of quantities of waste expected to be generated during preparation for PDMS and during PDMS.

Cleanup activities following PDMS will generate waste from a number of processes, including decontamination of the reactor coolant system, removal of contaminated portions of the reactor vessel head and control rod drive mechanisms, removal of the stairwell and elevator shaft in the basement, and removal of temporary shielding that has been placed in the reactor building. These activities will also generate secondary waste consisting of disposable protective clothing, tools, etc. The estimated volumes and classes of waste that would be generated during final cleanup following PDMS are shown in Table 3.15.

TABLE 3.14. Waste from Preparation for PDMS and from PDMS

Class of Waste(a)	Total Volume	
	ft^3	m^3
Preparation of PDMS		
Class A	100 to 200	2.8 to 5.7
PDMS		
Class A dry radioactive waste	600 to 2000	17 to 57
Class B, or C air filters	0 to 1250	0 to 41
Class A, B, or C residue from liquid waste treatment	100 to 400	2.8 to 11

(a) Waste is classified according to 10 CFR 61 criteria. See discussion in Section 2.3.2.

TABLE 3.15. Waste from Cleanup Following PDMS

Class of Waste(a)	Total Volume	
	ft^3	m^3
Class A waste	3,400 to 7,800	98 to 220
Class C waste	19,000 to 33,000	540 to 930
Class A, B, or C waste	9,600 to 29,000	270 to 810
Greater than Class C	Some possible	Some possible

(a) Waste is classified according to 10 CFR 61 criteria. See discussion in Section 2.3.2.

For delayed cleanup, the staff has assumed that waste generated before and during the PDMS period would be disposed of at a currently licensed site and waste generated during cleanup work would be disposed of at a regional site. The currently licensed site was assumed to be the facility operated by U.S. Ecology near Richland, Washington. An unspecified site 500 miles (800 kilometers) from the plant was assumed for the regional disposal site. The impact of the waste after disposal at either of these sites is considered to be outside the scope of this supplement and is the subject of a separate licensing action in connection with the waste disposal sites.

It is possible that some of the waste generated could exceed Class C limits, in which case it could not be accepted by a licensed burial site. The licensee, however, has a unique arrangement with the U.S. Department of Energy that allows such wastes to be transferred to the DOE on a cost-reimbursement basis. (It is under this agreement, known as the Memorandum of Understanding,(a) that the fuel is being transferred to the DOE Idaho Falls site.)

The environmental impact of transporting waste generated during preparation for PDMS and during PDMS was estimated assuming the Class A waste was packaged in 217-cubic-foot (6.1-cubic-meter) containers with shielding that was equivalent to 2.73 inches (6.9 centimeters) of lead. All other waste was assumed to be Class C and transported in 142-cubic-foot (4.0-cubic-meter) casks, which provide the equivalent of 4.5 inches (11.4 centimeters) of lead. Casks with these dimensions are currently licensed for such use. It was estimated that there would be 4 to 11 shipments of Class A waste and between 1 and 12 shipments of unspecified (Class A, B, or C) wastes to the Richland, Washington, site. For the purposes of estimating impacts, it was assumed that the unspecified waste would all be Class C waste.

(a) Memorandum of Understanding Between the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy Concerning the Removal and Disposition of Solid Nuclear Wastes from Cleanup of the Three Mile Island Unit 2 Nuclear Plant, March 15, 1982.

The environmental impact of transporting the waste generated during cleanup following PDMS was also estimated. The same shipping containers were assumed. It was estimated that there would be 16 to 36 shipments of Class A waste, and between 202 and 437 additional shipments of unspecified (Class A, B, or C) wastes. For the purposes of estimating impacts, it was again assumed that the unspecified waste would all be Class C waste.

The methodology for the assessment of shipping impacts is described in Appendix F. Transportation of this waste would result in the exposure of some members of the public to a very low radiation dose. The principally exposed group would be the truck crews; however, others such as those present at truck stops, travelers on the highways, and residents along the highways would also be exposed. The total population dose, excluding the dose from accidents that may occur during shipments, is expected to be 5 to 6 person-rem. The truck crews would receive by far the greatest portion of this dose, 3 to 4 person-rem.

As with transportation of any materials, there is a possibility that incidents during transportation may result in traffic accidents with or without injuries or fatalities. The estimated number of traffic accidents that might occur during the entire shipping program for delayed cleanup was 0.5 to 1 (between approximately 5 and 10 chances out of 10 that an accident would occur), depending upon the final waste volume. The staff estimated the number of injuries occurring over this shipping program at about 0.3 to 0.6 (between approximately 3 and 6 chances out of 10 that a injury would occur) and the number of fatalities at about 0.02 to 0.05 (between approximately 2 and 5 chances out of 100 that a fatality would occur). Appendix F provides additional details regarding the analysis of transportation accidents.

There is also a small probability that accidents may be severe enough to result in the breach of a waste container and release of some of the waste. To determine the risk of radiation exposure from a damaged waste container, the staff used a model that estimates the population dose by multiplying accident frequencies (the expected number of accidents) by accident consequences. Using this methodology, which is described more fully in Appendix F and the referenced documents, the staff estimated that a dose of about 0.0007 person-rem would result from accidents during shipment of all the waste generated during delayed cleanup.

3.2.5 Socioeconomic Impacts of Delayed Cleanup

The direct socioeconomic impacts of delayed cleanup were evaluated. The basis for the evaluation is included in Appendix G. The socioeconomic impacts of delayed cleanup are expected to be slight. The current work force of approximately 1150 would be reduced to 100 to 125 in the first year of PDMS and reduced to 70 to 75 during subsequent years. Cleanup following PDMS would probably be completed with a somewhat smaller staff than currently in use but larger than the PDMS staff. The staffing level for this phase has not been quantitatively determined and it is expected that it would depend on available technology and future plans for the facility.

Approximately 70 percent of the current work force resides in the Harrisburg-Lebanon-Carlisle labor market (Cumberland, Dauphin, Lebanon, and Perry Counties) and 25 percent in Lancaster County. In these areas, the economic impact of the reduced labor force might be most noticeable. Licensee-funded jobs in this area are expected to support approximately half again the number of jobs in the surrounding communities. However, because the employment reduction at the beginning of PDMS amounts to 0.2 percent of the local baseline employment, the impact should be minor.

The annual labor cost for the current staffing level is about $57.5 million per year, which would be reduced to $6.2 million for the first year and $3.8 million per year during the remainder of PDMS. The impact to the total income of the local communities is expected to be about twice the payroll level, $12 million to $13 million for the first year and about $7 million to $8 million per year thereafter.

3.2.6 Commitment of Resources During Delayed Cleanup

The principal resources committed in the delayed cleanup of TMI-2 will be money and radioactive burial ground space. Other resources, such as energy and ion exchange resins, will be relatively minor.

The cost of delayed cleanup has been evaluated by the NRC staff, based largely on information provided by the licensee on waste-volume estimates presented previously. All estimates are in 1988 dollars, although it is recognized that most of the resources required will be needed at the time of facility cleanup, assumed in this evaluation to begin in the year 2009. The cost of delayed cleanup, $200 million to $320 million as presented in Table 3.16, includes the labor cost addressed in Section 3.2.5, the estimated charges to dispose of the waste volume estimated in Section 3.2.4, and the waste transportation charges discussed in Appendix F.

Uncertainties in the labor cost are due to inflation, overhead costs, and uncertainties in staffing requirements. The greatest uncertainty in the labor cost will be the staffing required to complete the cleanup in 20 years. The staff assumed that a work force between 50 and 100 percent of the size of the defueling work force could complete the cleanup in 4 years. It was further assumed that any robotic costs would reduce the labor cost; therefore, they are not estimated as a separate cost. This estimate could be much too high if major portions of the work could be performed by relatively inexpensive, unsupervised robots. Costs might be too low if there is significant escalation in waste disposal or requirements for waste handling and packaging. Depending on the length of PDMS, an additional cost may result from retraining workers before the resumption of cleanup operations. This cost, which would mostly be seen in additional training expense, is also not readily quantified.

The waste disposal costs for both present and future waste disposal are 1988 rates. The 1988 disposal charge is approximately $50/cubic foot ($1800/cubic meter) plus surcharges for higher-than-normal radiation dose rates or curie content. These rates were raised approximately 18 percent from 1987 to 1988. Future rates are highly uncertain, especially disposal

TABLE 3.16. Cost of Delayed Cleanup (in millions of dollars[a])

Type of Cost	Projected Cost, $ million
Labor Costs	
Preparation for PDMS	3.8 to 6.2
1 year of PDMS at $6.2 million/yr	6.2
19 years of PDMS at $3.8 million/yr	72
4 years of cleanup following PDMS	120 to 230
Waste Disposal Costs	
Pre-PDMS waste, 800 to 3,850 ft^3	0.04 to 0.2
Post-PDMS waste, 32,000 to 70,000 ft^3	2.2 to 4.3
Waste Transportation Costs	0.5 to 1.3
Total[b]	200 to 320

(a) In 1988 dollars.
(b) The totals may not be exact because of rounding.

rates at a regional repository. The required waste burial ground space is estimated to be 33,000 to 74,000 cubic feet (930 to 2100 cubic meters). Some of this will be at currently licensed sites, but the majority is assumed to be at a regional site that has not yet been specified.

Waste disposal costs are related not only to waste volume and classification, about which there are uncertainties at present, but also to the technology used to dispose of the waste. Current waste disposal technology involves shallow land burial. Many of the regions are considering alternative technologies such as disposal in concrete bunkers and other engineered structures. Such alternative technologies may be more costly.

Waste transportation costs are closely related to the cost of energy and the distance between the disposal site and the TMI site. Accordingly, costs for transportation of waste to a regional site will be less than those for transportation to the currently operated disposal facility near Richland, Washington.

3.2.7 Regulatory Considerations of Delayed Cleanup

There are no regulatory considerations that would prevent the licensee from implementing long-term monitored storage of the facility. The PEIS supplement is part of the required evaluation necessary before the TMI-2 license can be amended.

3.3 IMMEDIATE CLEANUP

Immediate cleanup, as currently envisioned by the NRC staff, is described in Section 3.3.1. The offsite dose evaluation is discussed in Section 3.3.2, occupational dose estimates in Section 3.3.3, waste management impacts including transportation in Section 3.3.4, socioeconomic impacts in Section 3.3.5, commitment of resources in Section 3.3.6, and regulatory considerations in Section 3.3.7.

3.3.1 Description of the Immediate Cleanup Alternative

Immediate cleanup involves continuation of the cleanup process at the current level of effort and use of a work force the size of the current work force. The staff has assumed immediate cleanup could be performed over a 3-to-4-year period.

To progress from the end of defueling to the completion of cleanup will require additional decontamination of the reactor coolant system and the reactor building and shipment and disposal of the resulting waste. Little, if any, additional work would be required in the AFHB because by the time defueling is completed radiation dose rates in these areas will generally be at the level of an operating reactor at the end of its life, as discussed in Section 2.1. A description of the reactor coolant system cleanup and the decontamination of the reactor building follows.

3.3.1.1 Reactor Coolant System Cleanup

The selection of methods and processes for additional reactor coolant system decontamination is expected to depend on the technology available, the results of measurements being made at the present time, and the future disposition of the facility.

A discussion of possible methods for the decontamination of the reactor coolant system components is found in the PEIS (NRC 1981) and Supplement 1 (NRC 1984). For the purpose of this evaluation, it is assumed that the reactor coolant system decontamination would involve some mechanical decontamination methods followed by a general chemical decontamination. Mechanical decontamination would be performed in accessible areas such as the steam generator channel heads and pressurizer; it could involve vacuuming and possibly the use of slightly abrasive methods, such as grit blasting. Some use would probably be made of shielded work areas, long-handled tools, and power tools because robotics in its current state is not able to replace workers for all tasks.

Chemical decontamination methods are assumed to require placing the head or some other cover on the reactor vessel, filling the reactor coolant system with aqueous solutions, and circulating those solutions for a period of time with continuous filtration and chemical treatment to remove contamination. Various modifications to the reactor coolant system would be made to introduce and remove solutions. Valve lineups would be verified before beginning decontamination. Postdecontamination radiation surveys would also be

performed. The NRC staff has assumed that solutions would be processed in a modified, shielded area of the AFHB and solidified for offsite disposal. Chemical decontamination is discussed further in Chapter 6 of the PEIS and Section 2.1.3 of Supplement 1.

Although the exact process has not yet been defined, the NRC staff has assumed that such a procedure would reduce radiation dose rates from reactor coolant system components to levels that are typical of an operating reactor at the end of its life. Any hot spots left by the mechanical and chemical decontamination methods would be removed by cutting out the pipe or component to complete cleanup.

3.3.1.2 Reactor Building Cleanup

The current general area dose rates on the 305-foot elevation and the 347-foot elevation (see Section 2.1) indicate that some additional decontamination work would probably be required at these locations. In addition, the temporary shielding around equipment, such as the air coolers, ducts, floor hatches, lower section of the open stairwell, and the polar crane operator station, would need to be removed and additional decontamination and/or equipment removal performed. Electrical cables and trays, piping supports, and overhead structures are also expected to require decontamination or removal.

The most difficult area in the reactor building to decontaminate would be the basement. Cleanup of the basement is expected to require the removal of the concrete-block stairwell/elevator structure. This structure is reinforced with metal and would require aggressive methods to dismantle. It is expected that a combination of techniques, including robotic application of high-pressure water, water-air, or water-abrasive mixtures, might be used to dismantle sections of the structure. A plasma arc torch might also be adapted for robotic application. If robotics were not available to accomplish all demolition tasks, these tasks would be left until the majority of contaminated material was removed, and shielding would be placed so that workers could perform the tasks with long-handled tools. The handling and removal of the waste from demolition would require considerable worker time in the building. Workers would also be required to raise, lower, maintain, and modify the robots. Some spread of airborne contamination might result from demolition of the stairwell; additional building cleanup following dismantlement of the stairwell would probably be required.

The basement still contains debris such as tool boxes and construction materials that would require removal. Removal of this debris, as well as removal of insulation, equipment, and electrical boxes in the basement, could be performed robotically. Although packaging the waste and attaching the hoisting equipment using the robots would likely be slow, these methods would probably be used for most of the more highly contaminated material. Some manual handling of packaging operations on the upper elevations would be required. New access hatches could be cut through the floor on the 305-foot elevation. Once radiation dose rates were sufficiently low to permit entry into the basement, additional radiation surveys would be performed and the

remaining sources of contamination, which may be inaccessible with the robots, could be removed using manual methods.

3.3.2 Offsite Dose Evaluation for Immediate Cleanup

The evaluation of the radiation dose to the offsite population resulting from the alternative of immediate cleanup includes an assessment of the dose from both routine and accidental releases of radiation.

3.3.2.1 Routine Releases

The routine releases of radiation from the TMI-2 facility occurring by atmospheric and liquid pathways are not expected to vary much from those currently occurring as shown in Table 3.3. Some rise in effluent concentrations, however, may be experienced during aggressive decontamination efforts such as those that might accompany the decontamination or removal of the concrete-block stairwell/elevator structure. These operations could increase the amount of activity in the reactor building atmosphere, thus increasing the amount of activity released from the facility through the double-stage HEPA filters.

Atmospheric Releases. Radionuclide releases from the reactor building into the atmosphere during immediate cleanup were estimated by applying the same method used to estimate releases for the cleanup period following PDMS (Section 3.2.2.1) except that the quantity of radionuclides released was not adjusted to account for 20 years of radioactive decay. The amount of radioactive material assumed to be released annually into the atmosphere during a 4-year cleanup period is shown in Table D.10 of Appendix D.

Table 3.17 shows the estimated 50-year dose commitment to the maximally exposed individual (described in Section 3.2.2.1), the collective 50-year dose commitment to the 2.2 million people within a 50-mile (80-kilometer) radius, and the collective 50-year dose commitment to the population outside the 50-mile (80-kilometer) radius that receives radiation dose (attributable to TMI-2) from external exposure, inhalation, and the consumption of food products exported from within the 50-mile (80-kilometer) radius from routine airborne releases during immediate cleanup. The dose to the maximally exposed individual and the collective dose to the population were calculated for the entire 4-year cleanup period.

Liquid Releases. Liquid releases to the Susquehanna River will occur during the 4-year period expected for immediate cleanup. The liquids will be largely the result of water used during the decontamination process, for instance, to flush and decontaminate the reactor coolant system and the reactor coolant drain tank, as well as to remove contamination in other areas of the facility. Before release the liquids would be processed through the EPICOR II system. Maximum releases of 250,000 gallons (950,000 liters) a

TABLE 3.17. 50-Year Dose Commitments from Routine Atmospheric Releases During Immediate Cleanup

	50-Year Dose Commitment from a 4-Year Period of Releases	
	Critical Organ (Bone)	Total Body
Maximally exposed individual	7 mrem	0.7 mrem
Total population within a 50-mile radius(a)	4 person-rem	2 person-rem
Total population outside the 50-mile radius(b)	0.4 person-rem	0.2 person-rem

(a) Approximately 2.2 million persons in 1989.
(b) A population of unspecified size that receives radiation dose from external exposure, inhalation, and consumption of food products exported from within the 50-mile (80-kilometer) radius.

year were assumed, based on information given in the PEIS. As in Section 3.2.2.1 for the cleanup period following PDMS, it is assumed that all radionuclides listed in Table 2.4 [with the exception of krypton-85, a gas, and tritium, which is expected to be essentially removed from the facility before the start of PDMS (see Section 2.2.2.2) could be present in the liquid after processing the water through the SDS and/or the EPICOR II system. Although the actual concentration of each radionuclide after treatment by the EPICOR II system cannot be determined at this time, it was conservatively estimated that they would be comparable to the concentrations given in Supplement 2 (Table 2.2) for 100-percent processing of the accident-generated water. For those radionuclides not detected in accident-generated water, the lower limit of detection was assumed. The amount of radioactive material assumed to be released annually for liquid releases during immediate cleanup is shown in Table D.11 of Appendix D.

The resulting dose estimates are shown in Table 3.18. The dose estimates include a 50-year dose commitment to the maximally exposed individual (as described in Section 3.2.2.1), the collective 50-year dose commitment to the population within a 50-mile (80-kilometer) radius of TMI (approximately 2.2 million people; with an estimated 300,000 people obtaining their drinking water from the Susquehanna River), as well as to the population that consumes shellfish from Chesapeake Bay. These dose estimates were calculated for the entire 4-year period assumed for cleanup following PDMS.

3.3.2.2 Accidental Releases

The potential exists for accidental releases of both airborne and liquid contamination during immediate cleanup.

TABLE 3.18. 50-Year Dose Commitments from Routine Liquid Releases to the Susquehanna River During Immediate Cleanup

	50-Year Dose Commitment from a 4-year Period of Releases	
	Critical Organ (Bone)	Total Body
Maximally Exposed Individual	mrem	mrem
Consumption of Susquehanna River water and fish/ participation in rivershore activities	0.1	0.03
Consumption of Chesapeake Bay shellfish	0.01	0.0007
Total Population	person-rem	person-rem
Consumption of Susquehanna River water and fish/ participation in rivershore activities(a)	1	0.03
Consumption of Chesapeake Bay shellfish(b)	5	0.3

(a) Approximately 2.2 million persons in 1989. An estimated 300,000 persons were assumed to obtain their drinking water from the Susquehanna River downstream of TMI-2.

(b) A population of unspecified size that consumes shellfish from Chesapeake Bay. Approximately 12 percent of the dose from the consumption of Chesapeake Bay shellfish is received by persons with 50 miles (80 kilometers) of TMI-2.

Accidental Atmospheric Releases. The potential for accidents resulting in the release of radionuclides during immediate cleanup has been investigated. Three potential accidents resulting in airborne releases were developed from the list of potential accidents given in the PEIS that have a probability of occurring during the cleanup process. The three accidents evaluated are the same as those evaluated for cleanup following PDMS in Section 3.2.2.2: a fire in the stairwell/elevator structure, the rupture of a double-stage HEPA filter during decontamination efforts, and the spill of decontamination solution in the reactor building. A single accident resulting in a liquid release was identified for evaluation. This accident involved the rupture of a tank containing liquid that had not yet been processed completely to remove radioactive material. The assumptions made for the analysis of each accident occurring during immediate cleanup are similar to those discussed for delayed cleanup in Section 3.2.2.2, with the exception of the lack of radioactive decay (because immediate cleanup does not involve a storage period) and other minor exceptions detailed below.

For the accidental fire in the stairwell/elevator structure during immediate cleanup, the fraction of activity in the structure that is assumed to be released is the same as for the case of a fire during cleanup following PDMS. During immediate cleanup, double-stage HEPA filters would also be used routinely in each train of the reactor building ventilation system. Thus, for the case of a fire in the stairwell/elevator structure during immediate cleanup, the fraction of radioactive particulates penetrating the second filter was conservatively estimated at 0.0001 (see Section 3.2.2.1). The amount of radioactive material estimated to be released during this accident is shown in Table D.12 of Appendix D. Table 3.19 shows the estimated 50-year dose commitment to the maximally exposed individual (as described in Section 3.2.2.1), the collective 50-year dose commitment to the 2.2 million people within a 50-mile (80-kilometer) radius, and the collective 50-year dose commitment to the population outside the 50-mile (80-kilometer) radius that receives radiation dose (attributable to TMI-2) from external exposure, inhalation, and the consumption of food products exported from within the 50-mile (80-kilometer) radius.

The accidental failure of a double-stage HEPA filter is assumed to occur at the most critical time during the immediate cleanup process, which is assumed to be during the postulated demolition of the stairwell/elevator structure, as in the cleanup following PDMS (see Section 3.2.2.2). Both the assumptions given in Section 3.2.2.2 and the inventory of radionuclides are assumed to be the same for immediate cleanup, except that the inventory has not decayed for a 20-year period. The maximum amount of radioactive material estimated to be released during this accident is shown in Table D.13 of Appendix D. Table 3.20 shows the estimated 50-year dose commitment to the maximally exposed individual (as described in Section 3.2.2.1), the collective 50-year dose commitment to the 2.2 million people within a 50-mile (80-kilometer) radius, and the collective 50-year dose commitment to the

TABLE 3.19. 50-Year Dose Commitments from an Accidental Atmospheric Release (Fire in the Stairwell/Elevator Structure) During Immediate Cleanup

	50-Year Dose Commitment	
	Critical Organ (Bone)	Total Body
Maximally exposed individual	0.02 mrem	0.002 mrem
Total population within a 50-mile radius(a)	0.01 person-rem	0.006 person-rem
Total population outside the 50-mile radius(b)	0.001 person-rem	0.0004 person-rem

(a) Approximately 2.2 million persons in 1989.
(b) A population of unspecified size that receives radiation dose from external exposure, inhalation, and consumption of food products exported from within the 50-mile (80-kilometer) radius.

population outside the 50-mile (80-kilometer) radius that receives radiation dose (attributable to TMI-2) from external exposure, inhalation, and the consumption of food products exported within the 50-mile (80-kilometer) radius from this release.

TABLE 3.20. 50-Year Dose Commitments from an Accidental Failure of a HEPA Filter During Immediate Cleanup

	50-Year Dose Commitment	
	Critical Organ (Bone)	Total Body
Maximally exposed individual	19 mrem	2 mrem
Total population within a 50-mile radius(a)	11 person-rem	8 person-rem
Total population outside the 50-mile radius(b)	2 person-rem	0.6 person-rem

(a) Approximately 2.2 million persons in 1989.
(b) A population of unspecified size that receives radiation dose from external exposure, inhalation, and consumption of food products exported from within the 50-mile (80-kilometer) radius.

The consequences of an atmospheric release from an accidental spill of reactor coolant system decontamination solution inside the reactor building are discussed in Section 3.2.2.2 for the cleanup following PDMS. The assumptions made for the occurrence of this accident during immediate cleanup are the same as those presented in Section 3.2.2.2 for cleanup following PDMS, except that the radioactive material has not undergone an additional 20 years of radioactive decay. The amount of radioactive material assumed to be released from the reactor building during this accident is shown in Table D.14 of Appendix D. Table 3.21 shows the estimated 50-year dose commitment to the maximally exposed individual (as described in Section 3.2.2.1), the collective 50-year dose commitment to the 2.2 million people within a 50-mile (80-kilometer) radius, and the collective 50-year dose commitment to the population outside the 50-mile (80-kilometer) radius that receives radiation dose (attributable to TMI-2) from external exposure, inhalation, and the consumption of food products exported from within the 50-mile (80-kilometer) radius.

TABLE 3.21. 50-Year Dose Commitments from an Accidental Spill of Reactor Coolant System Decontamination Solution During Immediate Cleanup

	50-Year Dose Commitment	
	Critical Organ (Bone)	Total Body
Maximally exposed individual	0.008 mrem	0.0004 mrem
Total population within a 50-mile radius(a)	0.01 person-rem	0.002 person-rem
Total population outside the 50-mile radius(b)	0.0003 person-rem	0.00001 person-rem

(a) Approximately 2.2 million persons in 1989.
(b) A population of unspecified size that receives radiation dose from external exposure, inhalation, and consumption of food products exported from within the 50-mile (80-kilometer) radius.

Accidental Liquid Releases. The assumed pathway for an accidental release of radionuclides via waterborne pathways is the release of contaminated water to the Susquehanna River based on the rupture of an 11,000-gallon (42,000-liter) storage tank (as in the accident analysis for cleanup following PDMS in Section 3.2.2.2). The concentration of each radionuclide that could be in the water (based on the list in Table 2.4) was conservatively estimated to be comparable to the concentration given in Table 2.2 of Supplement 2 (for the case where 40 percent of the total stored accident-generated water had been processed through the SDS and EPICOR II system). The exceptions are krypton-85 (a gas) and tritium, which is expected to be essentially removed from the facility before the start of PDMS (see Section 2.2.2.2). The lower limit of detection was used for those radionuclides not detected in the accident-generated water. The amount of radioactive material assumed to be released during this accident is shown in Table D.15 of Appendix D. Table 3.22 shows the estimated 50-year dose commitment to the maximally exposed individual (as described in Section 3.2.2.1), and the collective 50-year dose commitment to the population downstream of TMI-2 (approximately 2.2 million people, with an estimated 300,000 people obtaining their drinking water from the Susquehanna River), as well as to the population that consumes shellfish from Chesapeake Bay.

3.3.3 Occupational Radiation Dose Evaluation for Immediate Cleanup

The occupational radiation dose expected during the cleanup process described for the immediate cleanup alternative is estimated to be between 300 and 3100 person-rem, as shown in Table 3.23. This is the dose required to achieve radiation levels similar to those in an undamaged reactor nearing the end of its life and is in addition to the occupational radiation dose already received and the dose required to complete the defueling period.

TABLE 3.22. 50-Year Dose Commitments from an Accidental Liquid Release to the Susquehanna River During Immediate Cleanup

	50-Year Dose Commitment	
	Critical Organ (Bone)	Total Body
Maximally Exposed Individual	mrem	mrem
Consumption of Susquehanna River water and fish/ participation in rivershore activities	0.01	0.002
Consumption of Chesapeake Bay shellfish	0.001	0.00003
Total Population	person-rem	person-rem
Consumption of Susquehanna River water and fish/ participation in rivershore activities(a)	0.2	0.004
Consumption of Chesapeake Bay shellfish(b)	0.5	0.01

(a) Approximately 2.2 million persons in 1989. An estimated 300,000 persons were assumed to obtain their drinking water from the Susquehanna River downstream of TMI-2.

(b) A population of unspecified size that consumes shellfish from Chesapeake Bay. Approximately 12 percent of the dose from the consumption of Chesapeake Bay shellfish is received by persons within 50 miles (80 kilometers) of TMI-2.

The estimates given in Table 3.23 are based on a task-by-task analysis of the work to be done and are presented as a range of values because of the uncertainties in the cleanup process and technology. The range is wide because of uncertainties in the location and depth of penetration of contamination and the uncertainties involved in reactor coolant system decontamination. In addition, uncertainties exist regarding the effectiveness of the robots for performing many of the tasks.

This estimate is significantly lower than the estimate that was presented for immediate cleanup in Supplement 1 to the PEIS. The reason for this difference is almost entirely because the Supplement 1 estimate did not consider that robotics would be used to any appreciable extent. However, robotics currently are being used effectively by the licensee in desludging and scabbling concrete in the basement. The current estimate is within the range of the estimate presented in Supplement 1 for cleanup employing robotics. Supplement 1 presumed robotics to be available between 0 and 20 years following the completion of defueling.

TABLE 3.23. Occupational Radiation Dose Estimate for Immediate Cleanup

Task Description	Worker Dose Range, person-rem
Reactor disassembly and defueling	0.8 to 8.0
Reactor coolant system decontamination	53 to 920
Reactor building basement cleanup	240 to 2100
Other reactor building cleanup	7.0 to 79
Total(a)	300 to 3100

(a) The totals may not be exact because of rounding.

3.3.4 Waste Management Considerations of Immediate Cleanup

Cleanup activities will generate waste from a number of processes, including decontamination of the reactor coolant system, removal of contaminated portions of the reactor vessel head and control rod drive mechanisms, removal of the stairwell and elevator shaft in the basement, and removal of temporary shielding that has been placed in the reactor building. These activities will also generate secondary waste consisting of disposable protective clothing, tools, etc. The estimated volumes and classes of waste that would be generated are the same as those for cleanup following PDMS, as shown in Table 3.15.

The environmental impact of transporting the waste generated during immediate cleanup was estimated from the curie estimates given in Section 2.2. The staff assumed that the waste would be shipped in the same containers that were assumed for delayed cleanup (Section 3.2.4). Wastes were considered to be shipped to the licensed LLW disposal site near Richland, Washington, with 16 to 36 shipments of Class A waste, and between 202 and 437 additional shipments of unspecified (Class A, B, or C) waste. For the purpose of assessing transportation impacts it was assumed that the unspecified waste would all be Class C waste.

It is possible that some of the waste generated could exceed maximum Class C limits, in which case it could not be accepted by a licensed burial site. The licensee, however, has a unique arrangement with the U.S. Department of Energy that allows such wastes to be transferred to the DOE on a cost-reimbursement basis. (It is under this agreement, known as the

Memorandum of Understanding,(a) that the fuel is being transferred to the DOE Idaho Falls site.)

The methodology for the assessment of shipping impacts is described in Appendix F. Transportation of this waste would result in the exposure of some members of the public to a very low radiation dose. The principal exposed group would be the truck crews; however, others such as those present at truck stops, travelers on the highways, and residents along the highways could also be exposed. The total population dose, excluding the dose from accidents that may occur during shipments, is expected to be 20 to 25 person-rem. The truck crews would receive by far the greatest portion of this dose, 13 to 16 person-rem.

As with transportation of any materials, there is a possibility that incidents during transportation may result in traffic accidents with or without injuries or in fatalities. The estimated number of traffic accidents that might occur during immediate cleanup was 1 to 3, depending on the final waste volume. The staff estimated the number of injuries occurring over this shipping program at 1 to 3 and the number of fatalities at 0.1 to 0.2 (between approximately 1 and 2 chances out of 10 that a fatality would occur). Appendix F provides additional details regarding the analysis of transportation accidents.

There is also a small probability that accidents may be severe enough to result in the breach of a waste container and release of some of the waste. To determine the risk of radiation exposure from a damaged waste container, the staff used a model which estimates the population dose by multiplying accident frequency (the expected number of accidents) by accident consequences. Using this methodology, which is described more fully in Appendix F and the referenced documents, the staff estimated that a dose of about 0.002 person-rem would result from accidents during the shipment of all of the waste generated during immediate cleanup.

3.3.5 Socioeconomic Impacts from Immediate Cleanup

The direct socioeconomic impacts of immediate cleanup were evaluated. The basis for the evaluation is included in Appendix G. The socioeconomic impacts of the immediate cleanup alternative are expected to be minor. The present economic impact of TMI-2 cleanup (e.g, the same amount of employment and payroll or slightly less) would be maintained for a period of 3 to 4 additional years. At the completion of cleanup, the employment level could change significantly depending on the disposition of the facility.

The number of workers required to complete cleanup would be the same or less than the number that is involved in the current defueling and decontamination effort, approximately 1150. Approximately 70 percent of the current

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(a) Memorandum of Understanding Between the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy Concerning the Removal and Disposition of Solid Nuclear Wastes from Cleanup of the Three Mile Island Unit 2 Nuclear Plant, March 15, 1982.

work force resides in the Harrisburg-Lebanon-Carlisle labor market (Cumberland, Dauphin, Lebanon, and Perry Counties) and 25 percent in Lancaster County. This distribution would not be expected to change significantly during cleanup. These jobs are expected to support approximately half again the number in the surrounding communities, as outlined in Appendix G.

The annual labor cost would be about $57.5 million per year for 1150 workers. The impact to the total income of the local communities from immediate cleanup is expected to be approximately twice the payroll level.

3.3.6 Commitment of Resources During Immediate Cleanup

The principal resources committed in the immediate cleanup of TMI-2 would be money and radioactive burial ground space. Other resources, such as energy and ion exchange resins, will be relatively minor.

The cost of immediate cleanup has been evaluated by the NRC staff based largely on information provided by the licensee. All estimates are in 1988 dollars. The cost of immediate cleanup, as presented in Table 3.24, includes the labor cost addressed in Section 3.3.5, the estimated charges to dispose of the waste volume estimated in Section 3.3.4, and the waste transportation charges discussed in Appendix F.

TABLE 3.24. Cost of Immediate Cleanup

Type of Cost	Projected Cost, $ million(a)
Labor Costs 3 to 4 years at $57.5 million/yr	170 to 230
Waste Disposal Costs, 32,000 to 70,000 ft^3 (including decontamination wastes)	2.2 to 4.3
Waste Transportation Costs	1.5 to 3.2
Total(b)	170 to 240

(a) In 1988 dollars.
(b) The totals may not be exact because of rounding.

Uncertainties in the labor cost are due to the duration of cleanup, inflation, uncertainties in estimating nonlabor overhead costs, and uncertainties in staffing requirements. The staff assumed that a work force the size of the defueling work force could complete the cleanup in 3 to 4 years. It was further assumed that the cost of any new robots would reduce the labor cost; therefore, they are not estimated as a separate cost.

Burial ground volume, the other significant resource required in the immediate cleanup alternative, would be required for the disposal of between 32,000 and 70,000 cubic feet (910 and 2000 cubic meters) of low-level radioactive waste. The waste disposal costs are based on 1988 rates of $50/cubic foot ($1800/cubic meter) plus surcharges for wastes with higher-than-normal radiation dose rates or curie content. Uncertainties in waste disposal costs arise from uncertainties in waste volume and future waste disposal costs.

3.3.7 Regulatory Considerations of Immediate Cleanup

There are no significant regulatory considerations for immediate cleanup. The NRC staff will continue to review major cleanup activities for approval.

4.0 AFFECTED ENVIRONMENT

This section briefly describes the environment (including the population) that could be affected by the licensee's proposed action and alternatives evaluated in this supplement. This information has been taken primarily from the PEIS (NRC 1981). However, population distribution estimates have been updated since the PEIS and include projected populations for the year 2009. Other sections have been reviewed and information updated as appropriate.

Four geographic areas that potentially could be affected by the cleanup and storage activities have been identified: (1) the area in the vicinity of the TMI site, (2) the area downstream including the Susquehanna River and the Chesapeake Bay, (3) the transportation routes used for movement of materials to and from the site, and (4) the offsite disposal locations. In addition, there is a population outside the TMI vicinity that receives radiation dose attributable to the TMI-2 cleanup from inhalation, external exposure, and consumption of food products exported from within the 50-mile (80-kilometer) radius as well as from the consumption of shellfish from the Chesapeake Bay area.

4.1 THREE MILE ISLAND SITE VICINITY

The TMI site vicinity is the area within an approximate 12-mile (20-kilometer) radius of TMI. However, for purposes of evaluating radiation doses, the area within a 50-mile (80-kilometer) radius is considered. Figures 4.1 and 4.2 show the location of the site and its relationship to population centers and municipalities in the area.

The area surrounding TMI is predominantly rural and supports farming operations. The soils in the vicinity, combined with favorable physiographic and climatological features, produce higher-than-average crop yields for the State. Field crops such as corn and wheat are prominent, as are dairy, poultry, and livestock operations.

4.1.1 Population Distribution

In spite of extensive agricultural operations, the population density within the 12-mile radius in 1980 was about 570 persons per square mile (220 persons per square kilometer), substantially higher than the population density for the State as a whole. Several municipalities are located within the area; the largest city, 12 miles (20 kilometers) to the northwest, is Harrisburg with a population of about 53,000 (in 1980). Urban development is concentrated around population centers and along major transportation corridors.

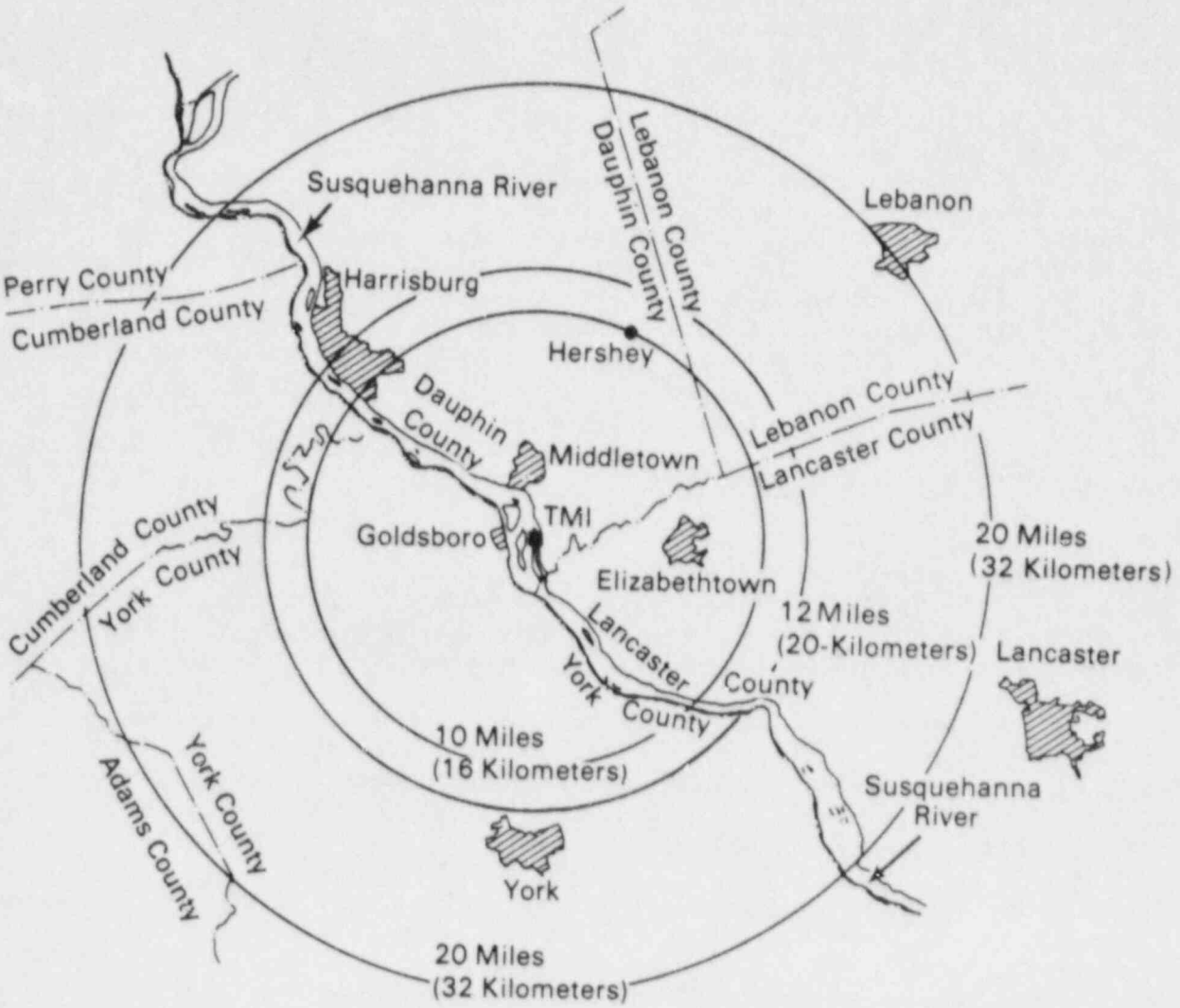

FIGURE 4.1. Map of the Area Within a 10-Mile (16-Kilometer), a 12-Mile (20-Kilometer), and a 20-Mile (32-Kilometer) Radius of the Three Mile Island Site

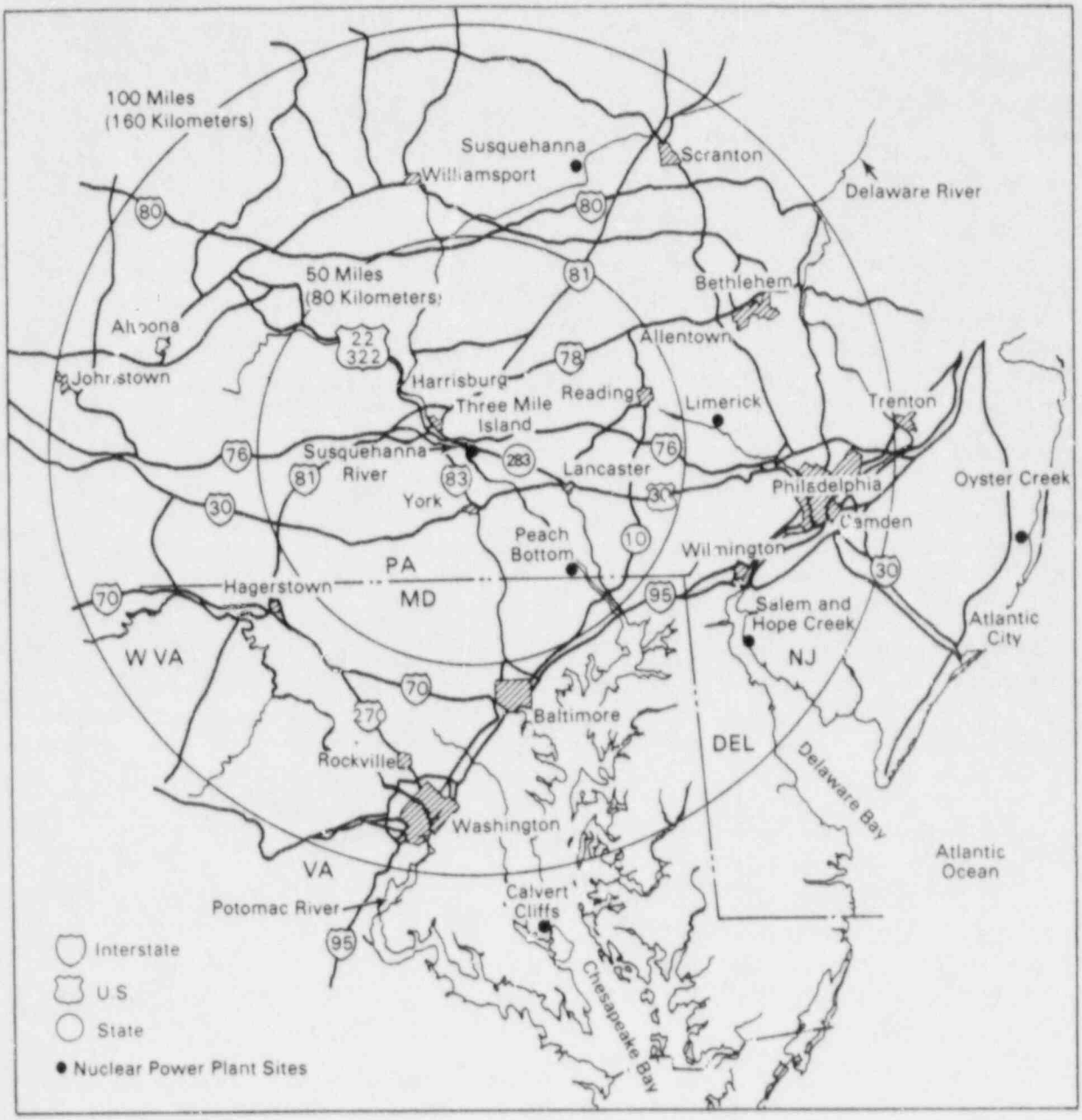

FIGURE 4.2. Map of the Area Within a 50-Mile (80-Kilometer) and a 100-Mile (160-Kilometer) Radius of the Three Mile Island Site

The total population in the 50-mile (80-kilometer) radius is estimated to be 2.2 million.(a) Approximately 350,000 persons live within a 12-mile (20-kilometer) radius of TMI. Figures 4.3 and 4.4 show the population distribution within a 12-mile (20-kilometer) and a 50-mile (80-kilometer) radius of TMI. The projected population for the year 2009(b) is 3.2 million persons within a 50-mile (80-kilometer) radius of TMI. Figures 4.5 and 4.6 show the projected population distribution within a 10-mile (16-kilometer) and a 50-mile (80-kilometer) radius of TMI for the year 2009.

4.1.2 Meteorology

The climate of southeastern Pennsylvania varies seasonally. In winter, the predominant air mass over the region is continental polar air moderated by the influences of the Appalachian Mountains and the Chesapeake and Delaware Bays. Winters are relatively mild for the latitude (40°9'N). In summer, maritime tropical air masses originating over the Gulf of Mexico or the Caribbean Sea predominate. Summers are warm and humid. While the extreme temperatures recorded for the area were 107°F (42°C) in July 1966 and -14°F (-26°C) in January 1912, temperatures of 90°F (32°C) or higher may be reached on only 20 to 25 days annually, and temperatures of 0°F (-18°C) or lower may be expected 1 to 2 days annually. The annual average relative humidity is about 70 percent.

The predominant wind flow is from the northwest. Figure 4.7 shows the onsite wind data at the 100-foot (30-meter) level. The meteorology of the TMI site has been compared with the meteorology of other reactor sites and was found to be fairly typical of valley sites in the frequency of inversions and other stable air phenomena.

Total annual precipitation in the area is expected to exceed 40 inches (102 centimeters), including a normal average snowfall of 37 inches (94 centimeters). The average annual evaporation is within the range of 33 inches (84 centimeters) (lake evaporation) to 45 inches (114 centimeters) (evaporation pan measure).

4.1.3 Surface Water

The TMI site is located in the Susquehanna River drainage basin, which has a total drainage area of 27,510 square miles (71,810 square kilometers) where it enters the Chesapeake Bay. Recorded data beginning in 1890 indicate that the flow rate of the Susquehanna River is highly variable, ranging from a minimum flow of 1700 ft^3/sec (48 m^3/sec) in 1964 to the maximum flood on

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(a) Based on data from an internal NRC document prepared by the Site Analysis Branch of the Office of Nuclear Reactor Regulation, "1981 Residential Population Estimates 0-80 Kilometers for Nuclear Power Plants."

(b) The projected population distribution for the year 2009 was assumed to be approximately equal to the population distribution projected for 2010 in a letter from F. R. Standerfer to the NRC, February 3, 1988. Subject: Post-Defueling Monitored Storage Environmental Evaluation.

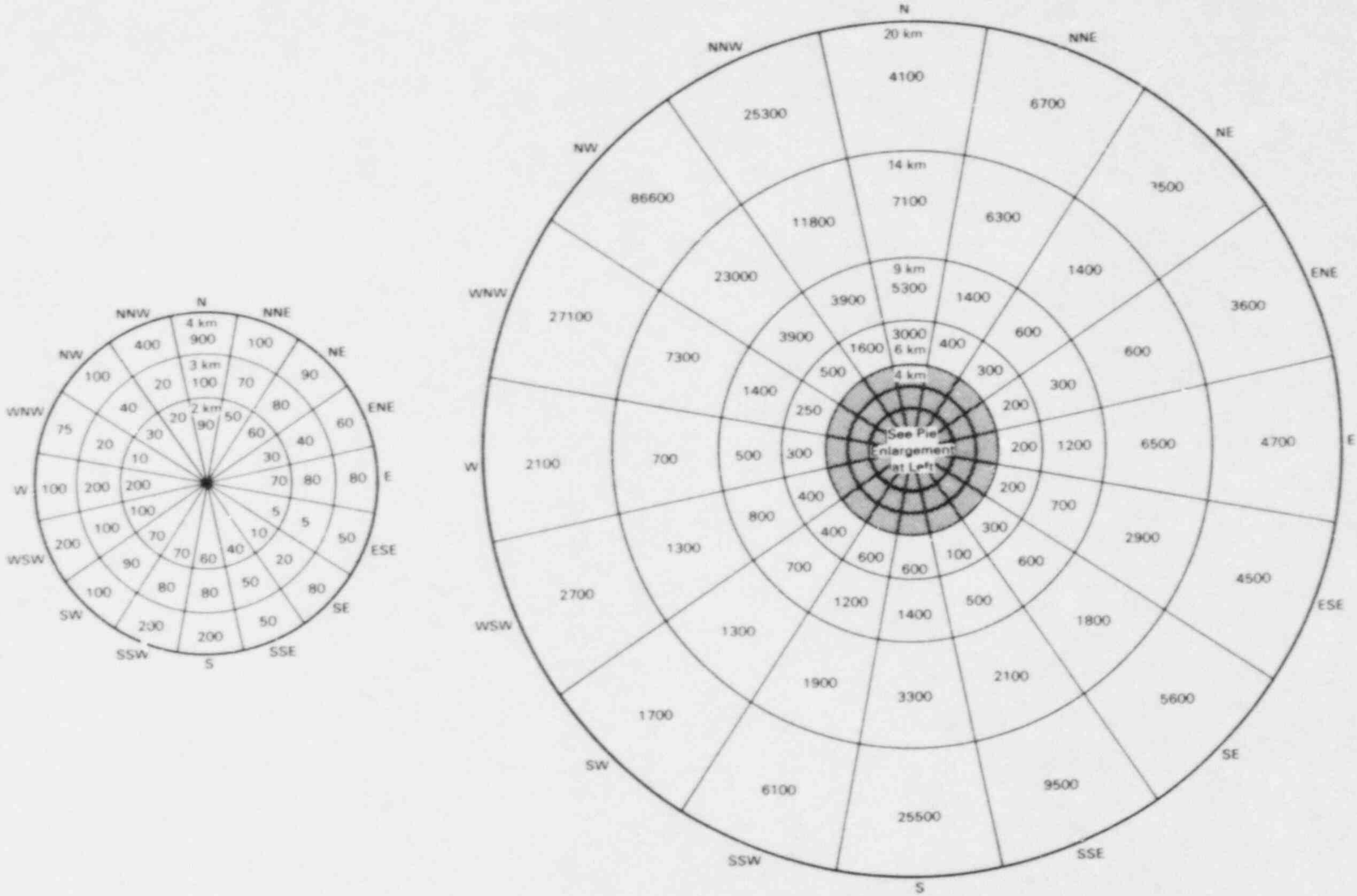

FIGURE 4.3. Population Distribution Within a 12-Mile (20-Kilometer) Radius of Three Mile Island (Based on data from an internal NRC document prepared by the Site Analysis Branch of the Office of Nuclear Reactor Regulation, "1981 Residential Population Estimates 0-80 Kilometers For Nuclear Power Plants".)

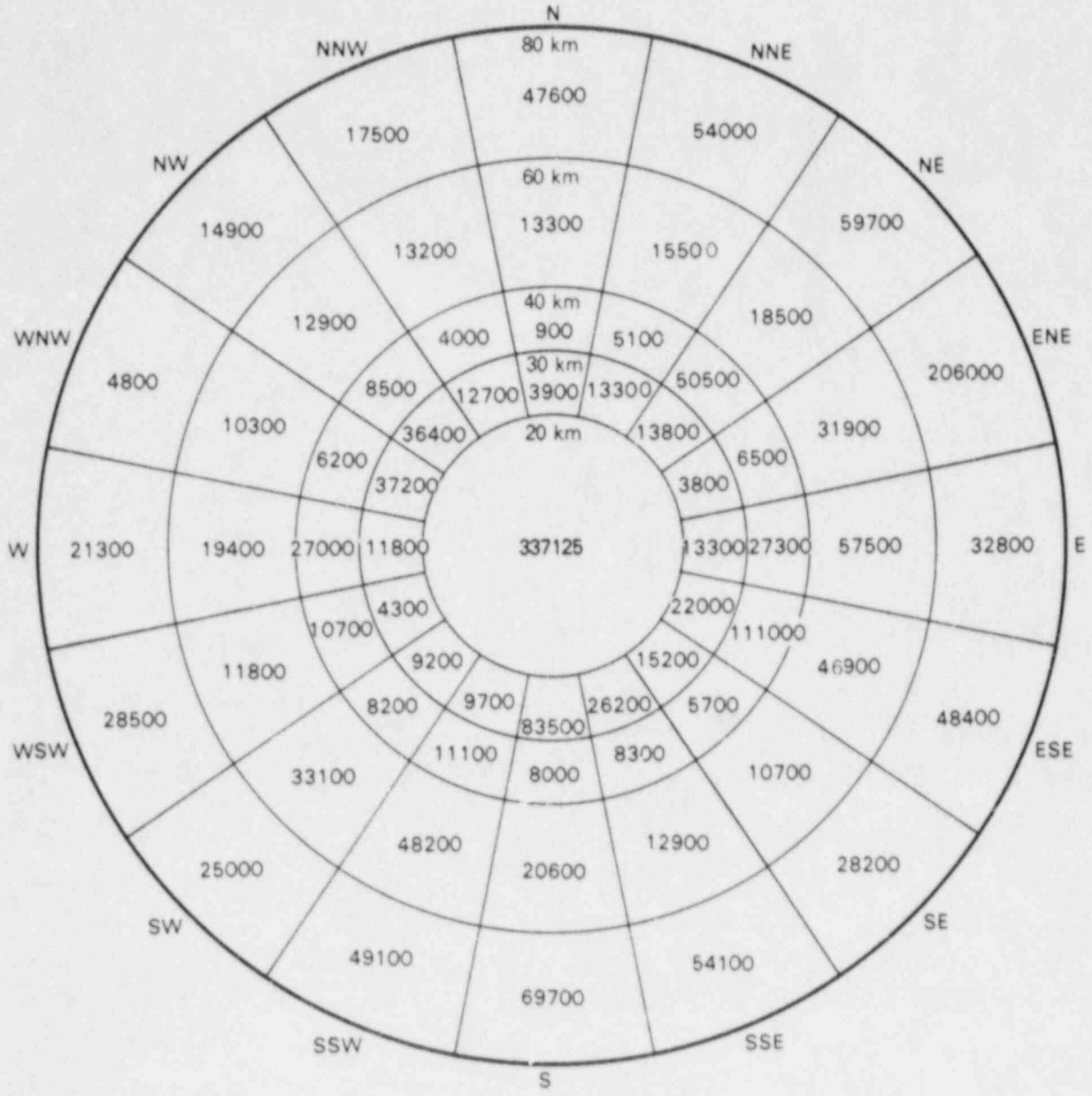

FIGURE 4.4. Population Distribution Within a 50-Mile (80-Kilometer) Radius of Three Mile Island (Based on data from an internal NRC document prepared by the Site Analysis Branch of the Office of Nuclear Reactor Regulation, "1981 Residential Population Estimates 0-80 Kilometers For Nuclear Power Plants.")

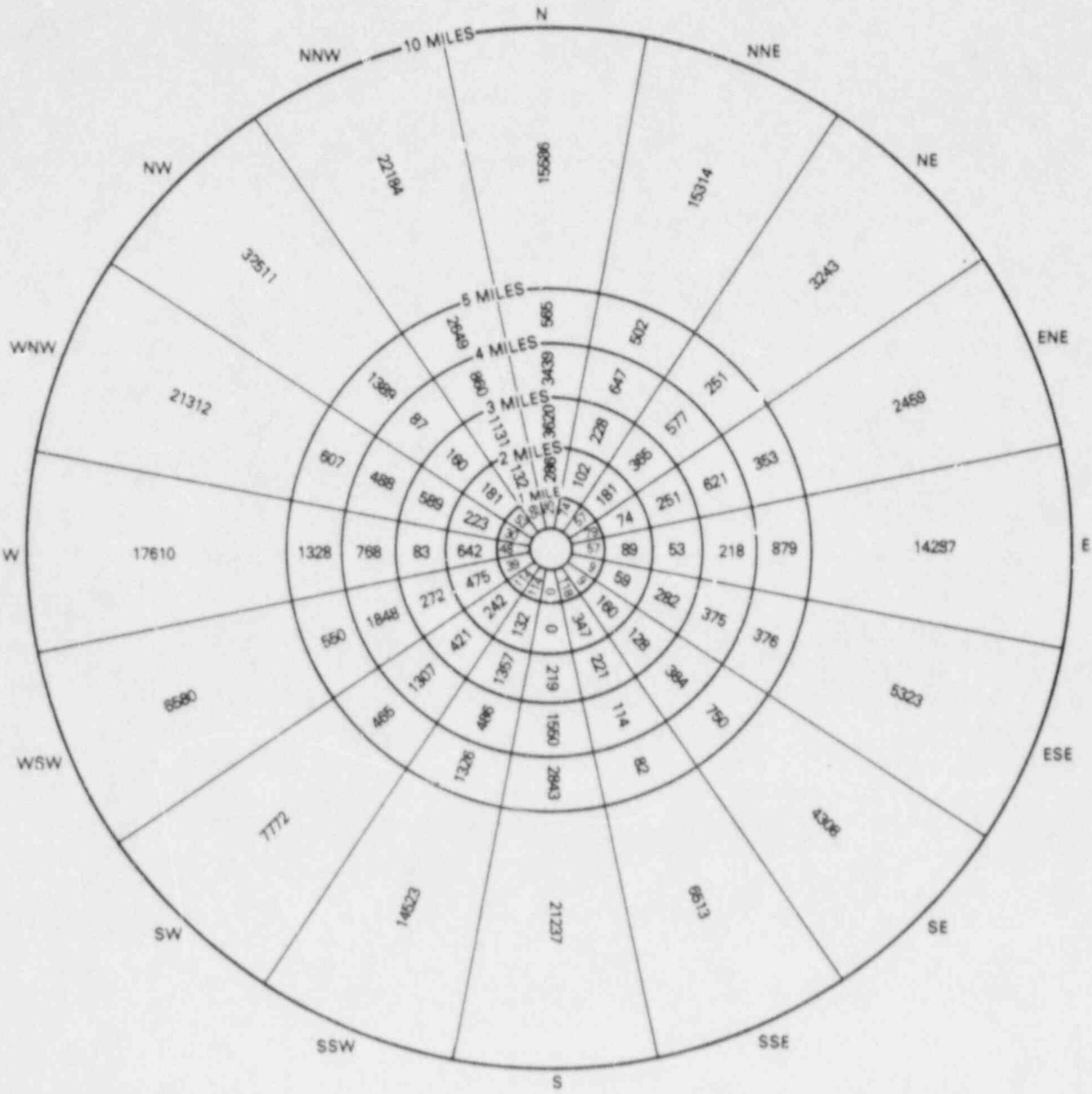

FIGURE 4.5. Projected Population Distribution for 2009 Within a 10-Mile (16-kilometer) Radius of Three Mile Island (data from a letter from F. R. Standerfer to NRC, February 3, 1988. Subject: Post-Defueling Monitored Storage Environmental Evaluation)

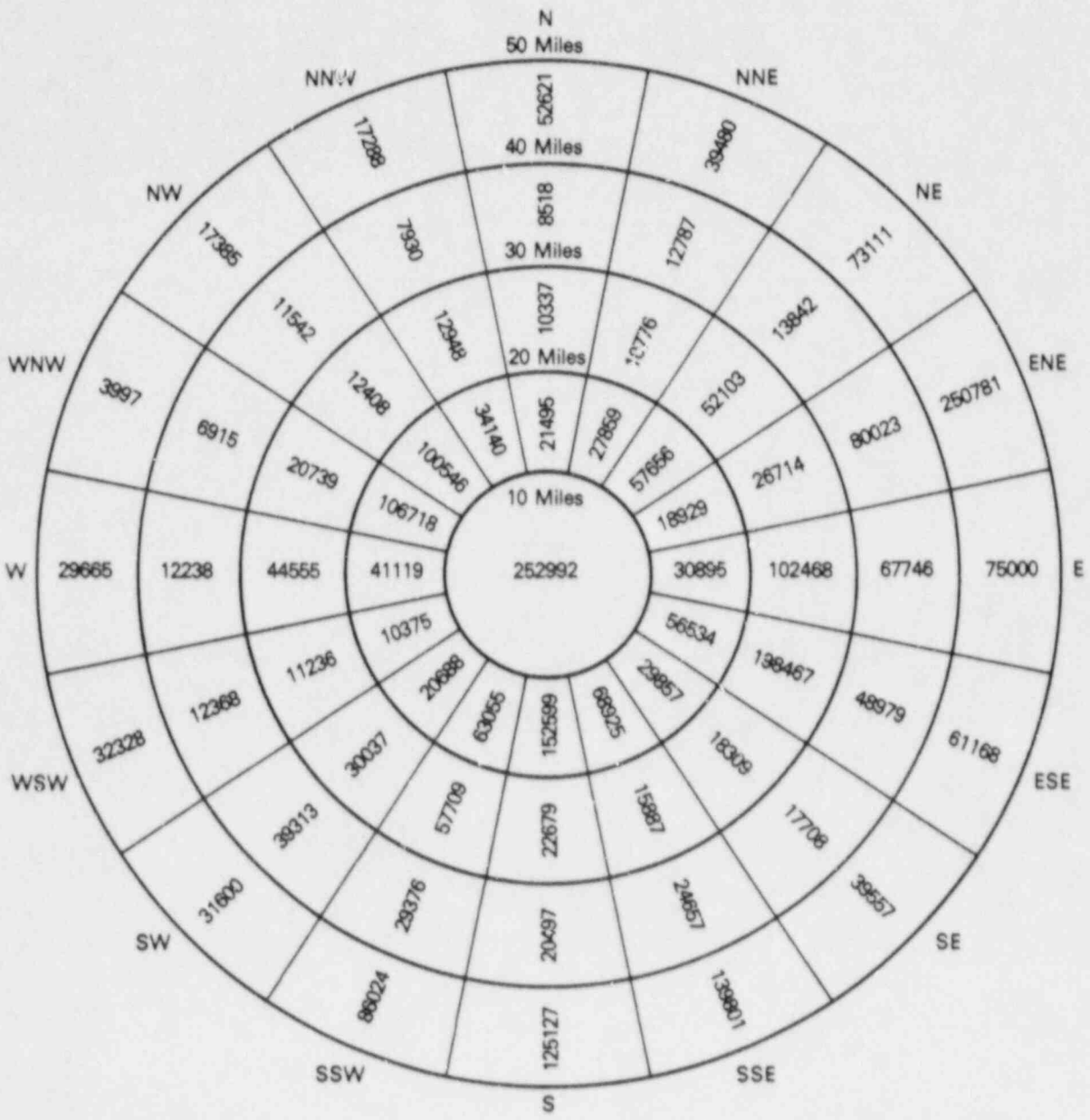

FIGURE 4.6. Projected Population Distribution for 2009 Within a 50-Mile (80-kilometer) Radius of Three Mile Island (data from a letter from F. R. Standerfer to the NRC, February 3, 1988. Subject: Post-Defueling Monitored Storage Environmental Evaluation)

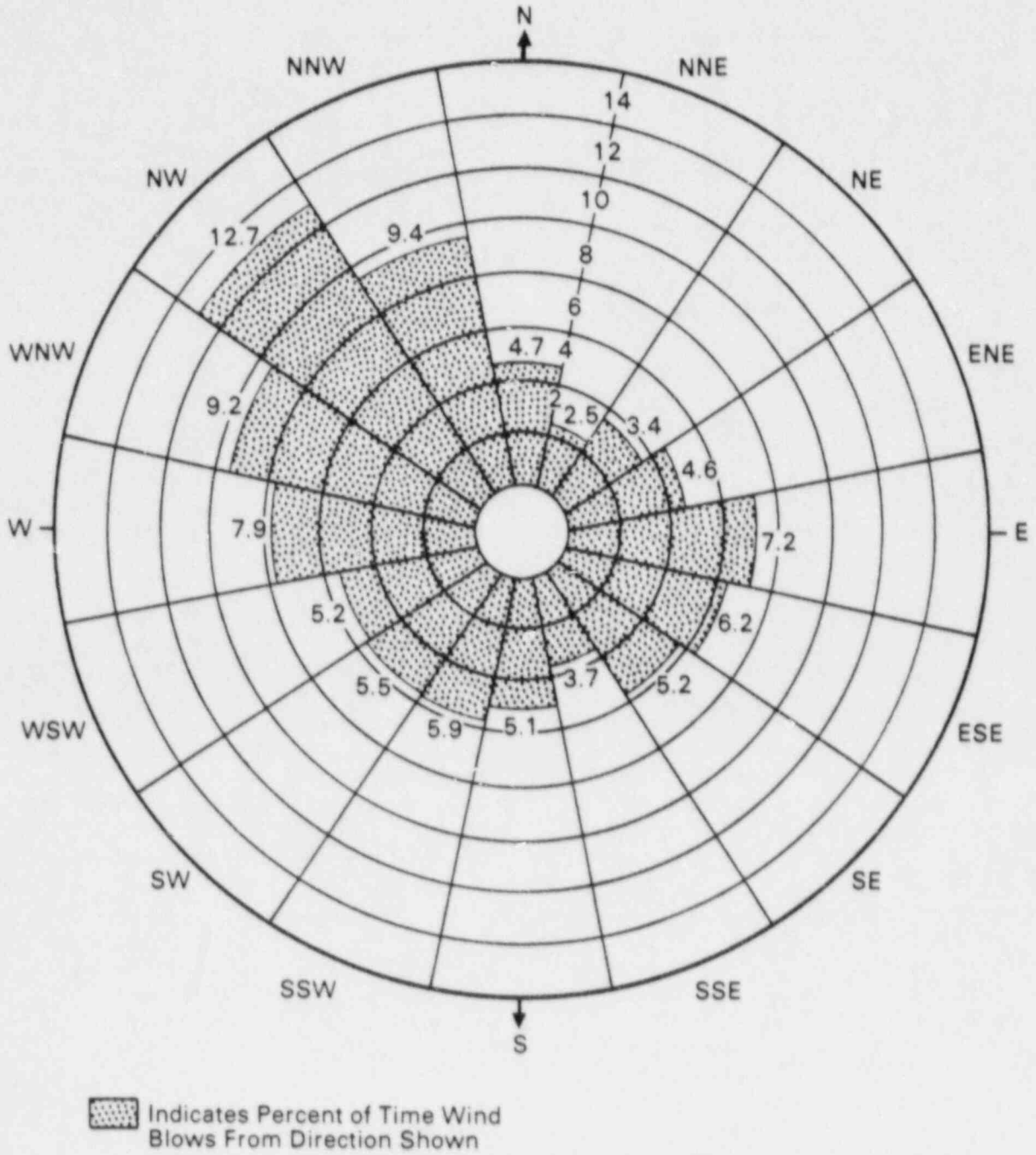

FIGURE 4.7. Three Mile Island Annual Average Wind Direction at 100 Feet (1972 to 1975 data)

record of 1,020,000 ft^3/sec (29,000 m^3/sec) during spring flooding in 1972 (NRC 1976). Mean monthly flows for the period 1891 to 1979 ranged from 11,700 to 82,600 ft^3/sec (330 to 2300 m^3/sec) with the low flow occurring in late summer and the high flow occurring in early spring. The average annual flow rate is 34,000 ft^3/sec (963 m^3/sec). Several dams and reservoirs are located on the Susquehanna River above and below TMI for flood control, low-flow augmentation, and power generation.

The island on which both the TMI-1 and TMI-2 reactors are located is within the 500-year flood plain (0.2-percent chance of flooding in any given year), but not within the 100-year flood plain as determined by the U.S. Army Corps of Engineers (NRC 1987). The island is diked for flood protection, and the dikes are inspected and maintained by the licensee. In addition, TMI-2 flood procedures require that flood door panels be installed when the river elevation reaches 302 feet (92 meters). Installation of flood door panels effectively precludes the entry of river water.

The surface water of the Susquehanna River downstream from Harrisburg is acceptable for all general uses, including aquatic life and recreation. However, the river is not an attractive source of public water supply because of occasional high sulfate levels and high amounts of wastewater-derived coliform bacteria. Below Harrisburg, late summer blooms of algae occur, which indicate high nutrient levels in the water, primarily phosphates and nitrates. This is attributable both to wastewater treatment and runoff from agricultural areas.

Currently, the river and streams in the TMI vicinity are used for both public and industrial water supplies, power generation, boating, sport fishing, and recreation. Sport fishing, but not commercial fishing, takes place in all streams in the general area of the site. The nearest potable water user is 5 miles (8 kilometers) downstream at the Brunner Island steam-electric generating station. Figure 4.8 shows the principal water users downstream of the TMI plant. Although Chester County, Pennsylvania, and the city of Baltimore, Maryland, also have water intakes downstream, they are seldom used.

Specific water quality data can be found in the PEIS (NRC 1981). In general, the water is moderately high in total hardness, with high and variable sulfate and iron concentrations (often in excess of the State limit), a relatively low alkalinity, and a high fecal coliform count (also, often in excess of the State limit). These characteristics are largely attributable to drainage from old coal mines in the watershed and from domestic and agricultural wastes.

Radioactivity measurements of Susquehanna River water were made by the U.S. Geological Survey before the TMI-2 accident. The tritium concentration was measured during the 1977 water year and found to be fairly constant at 178 pCi/L. Gross beta activity was measured on November 8, 1976, and reported as follows:

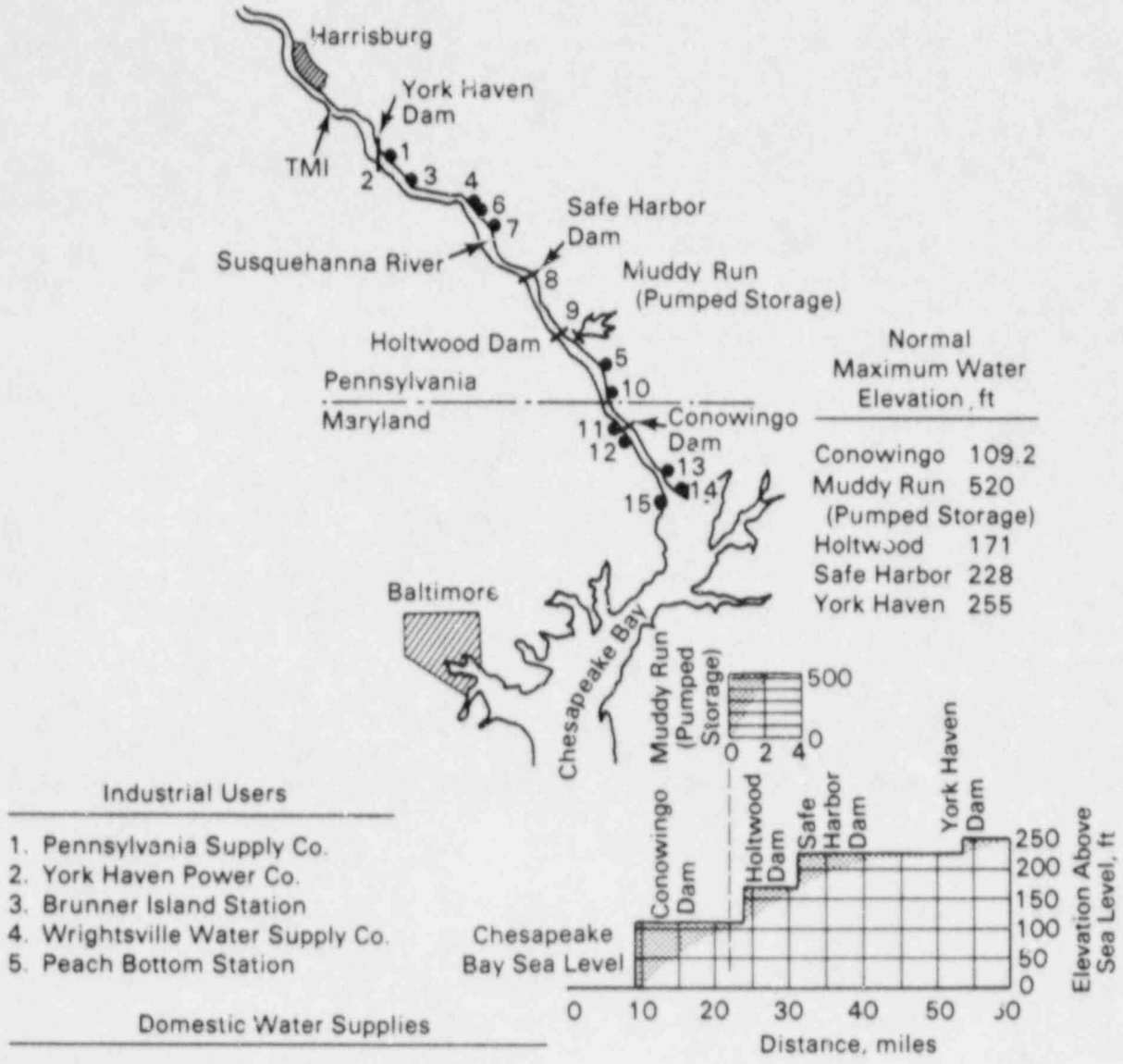

FIGURE 4.8. Principal Water Users Along the Susquehanna River in the Vicinity of Three Mile Island

Dissolved gross beta:	2.4 pCi/L as cesium-137
	1.9 pCi/L as strontium-90/yttrium-90
Suspended gross beta:	0.4 pCi/L as cesium-137
	<0.4 pCi/L as strontium-90/yttrium-90

Radium-226 was measured on the same date by the radon method as 0.08 pCi/L (alpha). Gross alpha activity on the same date was reported as follows:

Dissolved gross alpha:	<1.6 μg/L as natural uranium (<1.08 pCi/L)
Suspended gross alpha:	0.7 μg/L as natural uranium (0.5 pCi/L)

A measurement of uranium concentration, presumably by the chemical (fluorimetric) method, made on the same date gave a value of 0.06 μg/L. The contribution from the commercial nuclear fuel cycle is negligible compared to natural background. The radioactivity observed in the Susquehanna River at Harrisburg during 1977 was below the level regarded as normal for this latitude (40°9′N). For example, the average radioactivity levels in surface water in the Chicago area have been reported as alpha, 0.1 to 3 pCi/L, and beta, 5 to 10 pCi/L.

The tritium concentration of the Susquehanna River has been measured and found to be fairly constant. Samples of Susquehanna River water taken at Danville (upstream from TMI-2), collected and analyzed by the EPA between July 1985, and March 1987, have shown no detectable gamma activity and an average tritium concentration (±2 standard deviations) of 230 ± 200 pCi/L (EPA 1985, 1986a, 1986b, 1987a, 1987b, and 1987c). The Commonwealth of Pennsylvania has also collected a total of 2308 samples from the Susquehanna River and from water intakes which draw from the Susquehanna River both upstream and downstream of the plant (Commonwealth of Pennsylvania 1981, 1982a, 1982b, 1983, 1984, 1985, and 1986). Of these samples, 2307 contained less tritium than the lower limit of detection, which ranged from 230 to 440 pCi/L. A single sample taken at the Lancaster water intake showed 422 ± 192 pCi/L.

4.1.4 Groundwater

The licensee measures groundwater elevations monthly at 19 onsite monitoring stations. The mean groundwater elevation for these stations in 1986 was 283.1 feet (86.3 meters) mean sea level (MSL), as based on 218 readings. The Susquehanna River is normally at 277 feet (84.4 meters) MSL. With the exception of two stations, the readings ranged from 277.6 feet (84.6 meters) to 286.7 feet (87.4 meters). The station with the lowest reading recorded 275.5 feet (84.0 meters) MSL. The station with the highest reading recorded 293.2 feet (89.4 meters) MSL.(a) The potable water supplies nearest to TMI are three wells located on the east bank of the Susquehanna River, directly across from the site. All these wells have groundwater elevations above the river and above the groundwater level at TMI. Since the wells are upgradient, they are not likely to be affected by site activities.

(a) Letter from F. R. Standerfer to the NRC, June 23, 1987. Subject: Post-Defueling Monitored Storage Environmental Evaluation. 4410-87-6093 Document ID 0194P.

The site is underlain by sandy silts, sands, gravels, weathered bedrock, and hard siltstone (the Gettysburg Formation). The Gettysburg Formation has basic artesian characteristics in the site area. Groundwater flow is highly anisotropic along the strike direction, with specific capacities ranging from 0.33 to 15.0 gal/min/ft (1.2 to 57 L/min/m) of drawdown. The leakage of groundwater from the Gettysburg Formation would be expected to be upward, but would vary considerably with the degree of jointing and relationship to the strike direction. Therefore, accidental releases of effluents from the plant are not expected to migrate into the Gettysburg Formation.

Eight monitoring wells and nine observation wells were installed on the site to monitor groundwater. These wells were sited to detect leakage of contaminated water from the Unit 2 reactor, auxiliary buildings, and outside storage tanks. Groundwater quality has been monitored since the wells were installed in 1980.

4.1.5 Aquatic Ecology

The biota of the Susquehanna River includes organisms usually associated both with flowing waters and, because of the impoundments, with standing waters.

Large aquatic plants are rare in the river because of fluctuating flows and water levels and the type of river bottom substrates, which in most of the free-flowing areas are sand or rock. A dominant source of primary production is algae. The cycle of algae production is representative of algal succession in a lake (a spring bloom of diatoms, a summer abundance of green algae, and a late summer/early fall increase in blue-green algae and flagellates) and indicates the importance of the impoundments in the trophic structure of the river.

Zooplankton composition and abundance are variable; the dominant groups are rotifers (Branchionus sp.), cladocerans (Bosmina sp.), and copepods (Cyclops sp.). Periodic large populations of rotifers also suggest excessive domestic waste loadings of the river. The most abundant benthic (living on or near the bottom of the river) invertebrates are tubificid worms and insect larvae. The fish community can be characterized as a warm-water assemblage and is dominated by members of the minnow, perch, and sunfish families.

4.1.6 Terrestrial Ecology

Land use in the TMI vicinity is primarily agricultural, although a significant amount of land is also devoted to residential and urban development. Agriculture is diverse and includes corn and wheat farming, as well as dairy, poultry, and livestock operations. The forested areas surrounding TMI contain both hardwood and softwood trees. The plant community in these areas is less than 80 years old and consists of species that are common to this area.

In the vicinity of the TMI site, 212 species of terrestrial vertebrates were found, including 179 birds, 19 mammals, 8 reptiles, and 6 amphibians. Small-game animals include the eastern cottontail rabbit and the gray

squirrel. Mammalian predators include the longtail weasel and the red fox. The largest mammal found on the site was the white-tailed deer. Four species of upland game bird were found onsite: ring-necked pheasant, American woodcock, mourning dove, and rock dove. Whistling swan, Canada goose, nine species of dabbling duck, seven species of diving duck, and three species of mergansers also were reported. This sampling of species is also typical of the fauna found downriver of the site. Because the Susquehanna River is a major flyway, large numbers and many species of migratory and resident waterfowl nest and feed on the ponds and reservoirs along the river.

None of the species of birds, mammals, reptiles, or amphibians known to reside on or in the immediate vicinity of the TMI site have been designated as federally protected species in Pennsylvania. However, three of the federally listed species, the bald eagle, peregrine falcon, and Indiana bat, may migrate through the area. No known nesting sites of the three have been found in the TMI site vicinity, and no known sites are on record.

The Commonwealth of Pennsylvania's list of endangered and threatened species includes three species of bird that have the potential to pass through the TMI area. They include the king rail, osprey, and black tern. Only the osprey has been sighted in the immediate area of the TMI site, although the nearest recorded nesting site is 33 river-miles (53 kilometers) south.

The golden seal (Hydrastis canadensis), a federally protected plant species, has been confirmed to occur in the TMI vicinity. Wild ginseng (Panax quinquefolius), which is also on the Federal list, is on the historical record of species in the TMI vicinity, although no recent sightings have been recorded.

4.1.7 Background Radiation

Radiological conditions in the TMI area have been monitored by the EPA. The EPA, Office of Radiation Programs, has estimated approximately 87 mrem/yr external radiation for Harrisburg (Oakley 1972) and 100 mrem/yr for Pennsylvania, with 45 mrem/person from cosmic radiation and 55 mrem/person from natural terrestrial radioactivity (Klement et al. 1972). Neither of these dose estimates include the annual internal dose from natural radioactivity which is estimated at 25 mrem/yr total, which includes 17 mrem/yr from potassium-40, 1 mrem/yr from carbon-14, and 3 mrem/yr each from radon-222 and polonium-210 (Klement et al. 1972). The doses from radon-222 and polonium-210 are expected to vary considerably among individuals based on the location and air exchange rate of their homes and other factors.

The background concentrations of various radionuclides in air and precipitation in the vicinity of TMI-2 are representative of background concentrations elsewhere in the United States. The EPA has measured beta radioactivity in air in the Harrisburg, and TMI areas between July 1985, and March 1987 (EPA 1985, 1986a, 1986b, 1987a, 1987b, 1987c). A total of 264 samples analyzed in the field for beta activity (not including samples

taken in May and June of 1986) averaged 0.2 pCi/m^3.(a) The activity in the May and June samples is attributable to the Chernobyl accident which occurred April 26, 1986. The average gross beta activity in 30 samples collected at Harrisburg during May 1986, was 0.6 pCi/m^3. The 42 samples collected at TMI averaged 0.8 pCi/m^3. The average concentration in nine samples taken during June at Harrisburg was 0.3 pCi/m^3. The average of the seven samples taken at TMI in June was 0.7 pCi/m^3. In addition there were two samples taken at Middletown during June that averaged 0.3 pCi/m^3. The detection limit for these analyses was 0.1 pCi/m^3.

Air-sample filters from Harrisburg and TMI were combined for 6-month periods and analyzed for plutonium and uranium. The average isotopic concentrations (±2 standard deviations) are as follows: plutonium-238, 0.50 ± 0.70 aCi/m^3(b); plutonium-239/240, 0.33 ± 0.46 aCi/m^3; uranium-234, 15.7 ± 3.2 aCi/m^3; uranium-235, 0.44 ± 0.46 aCi/m^3; and uranium-238, 13.6 ± 2.8 aCi/m^3.

Precipitation samples were also collected and analyzed by the EPA between June 1985, and March 1987, at Harrisburg and Middletown. The samples were analyzed for gross beta activity, tritium, and in some cases gamma activity. Except for samples collected during May 1986, all samples were combined for a month. Results are reported as nCi/m^2.(c) Minimum detectable levels are determined by the amount of rainfall as well as other factors. The average beta activity (±2 standard deviations) in 17 monthly samples (excluding May 1986) at Harrisburg was 0.21 ± 0.06 nCi/m^2. The average of 19 monthly samples at Middletown was 0.15 ± 0.05 nCi/m^2. (The total beta activity for the May 1986, samples affected by Chernobyl at Harrisburg was 1.22 ± 0.77 nCi/m^2; at Middletown it was 2.87 ± 0.55 nCi/m^2.) Tritium results from 38 samples in Harrisburg and Middletown averaged 0.19 ± 0.2 nCi/L. In addition, many of these same precipitation samples were analyzed for gamma-emitting radionuclides. The only samples exceeding the lower limit of detection were those taken during or shortly after the Chernobyl accident.

4.2 SUSQUEHANNA RIVER/CHESAPEAKE BAY AREA

The predominant features of the area under evaluation include the Susquehanna River and the Chesapeake Bay. The 450-mile (724-kilometer) Susquehanna is a major river in the eastern United States and supplies about 50 percent of the fresh water in the bay. The Chesapeake Bay is one of the largest estuaries in the world, having a surface of about 4400 square miles (11,400 square kilometers), a length of nearly 200 miles (320 kilometers), and more than 7000 miles (11,000 kilometers) of shoreline. The Susquehanna River/Chesapeake Bay system supports commercial and recreational fishing and boating and supplies water for public and industrial use.

(a) There are one trillion (1,000,000,000,000) picocuries in a curie.
(b) There are one quintillion (1,000,000,000,000,000,000) attocuries in a curie.
(c) There are one billion (1,000,000,000) nanocuries in a curie.

Sport fishing is a popular activity in the Susquehanna River from the vicinity of TMI to Havre de Grace (see Figure 4.8). The portion of the river below the Conowingo Dam (shown in Figure 4.8) receives spawning migrations of some anadromous species, primarily members of the herring family and striped bass. Sport fishing for crappie, bass, walleye, channel catfish, and sunfish is popular on the entire river. Although the river primarily serves local residents, sizable numbers of fishermen from Maryland and Pennsylvania are attracted to the river.

Sport fishing on the Chesapeake Bay is also a popular activity involving both private and charter boats. The majority of the fishing is done by residents of Maryland, the District of Columbia, Delaware, Pennsylvania, and Virginia. There is also a large and growing use of the Chesapeake Bay for other water-oriented recreation, such as boating.

Shellfish and finfish that are commercially harvested from the Chesapeake Bay include blue crabs, oysters, soft-shelled crabs, surf clams, sea scallops, menhadden, croaker, bluefish, and flounder. The shellfish and finfish harvest is marketed fresh and processed. Regular markets are spread across the United States and parts of Canada.

In addition to the Chesapeake Bay's importance to commercial and sport fishing, the surrounding marshes and woodlands provide thousands of acres of natural habitat for a diversity of wildlife. In the shallow waters of the upper Chesapeake Bay, large aquatic plants and terrestrial plants, such as cord grass (Spartina sp.) and wild celery (Vallisneria sp.), are quite productive, making the area an attractive food source for waterfowl. This area is in the path of the Atlantic flyway and provides wintering and feeding grounds for migrating waterfowl. The waterfowl species that are attracted to the region in large numbers include Canada geese, ducks, whistling swans, other species of birds that use the wetlands for food and other habitat requirements, plus a variety of game birds. The wildlife resources of the area provide opportunities for hunting and trapping and for activities such as bird watching, nature walking, and nature photography.

4.3 TRANSPORTATION ROUTES

The vicinity of TMI is served by the transportation routes shown in Figure 4.2. Interstate 81 is oriented northeast to southwest. Interstate 80 runs east-west, north of the site. Interstate 70, south of the site, also runs east-west. State Route 10, although a much lower-volume road, is important locally. It is oriented north-south, less than 50 miles east of the site. Interstate 76, the Pennsylvania Turnpike, north of the site and south of Harrisburg, connects with urban centers to the east and west. U.S. Route 30 is a high-capacity road between Lancaster and York, oriented east-west and passing south of the site. Interstate 83, originating at Harrisburg, extends south to York and Baltimore. U.S. Route 22/322 passes by the site to the northwest.

Shipments of radioactive waste from the TMI site routinely pass over State Route 283 and Interstate Routes 83, 81, and 80 before they leave the

Commonwealth of Pennsylvania to the west. Interstate 76 is not normally used for westbound shipments because of tunnel restrictions. Interstate 81 is normally used for southbound shipments. The highway route to the LLW disposal site near Richland, Washington, is shown in Figure 4.9.

4.4 OFFSITE DISPOSAL LOCATIONS

All plans for cleanup of the reactor building and AFHB involve disposal of radioactive wastes at licensed LLW burial sites at offsite locations. Shipments of low-level wastes for disposal have been transported by truck to the commercial LLW burial site near Richland, Washington.

The shipment of low-level wastes to the commercial LLW burial site near Richland is assumed for waste disposal before and during PDMS and all wastes generated during immediate cleanup. Although other sites may be available at this time, because of the distance involved [2680 miles (4313 kilometers)], this LLW site is judged to be the bounding case from a transportation accident standpoint.

The LLW burial site near Richland is operated by U.S. Ecology, Inc., as a commercial radioactive waste disposal site. The site is located in a semi-arid area of relatively low population density. The facility is located 25 miles (40 kilometers) northwest of Richland on 100 acres (40 hectares) of leased land near the center of the DOE Hanford Nuclear Reservation. The facility is licensed by the NRC for the disposal of commercial radioactive waste. The site is currently used for storage or disposal of radioactive waste materials. The impact of LLW disposal at this site is the subject of separate environmental evaluations and is considered beyond the scope of this document.

Because the Low Level Waste Policy Amendments Act mandates State and/or regional disposal sites (or State possession of low-level waste) by 1990, a generic site within 500 miles (800 kilometers) was assumed to bound the impact of waste transportation for waste that is disposed of following PDMS. The characteristics of this site are unknown at the present time and operation will be the subject of a separate environmental review. Impact of the disposal of TMI waste at this site is beyond the scope of this document.

FIGURE 4.9. Route to the Low-Level Waste Disposal Facility from Three Mile Island

5.0 COMPARISON OF THE ENVIRONMENTAL IMPACTS OF DELAYED AND IMMEDIATE CLEANUP

This section compares the environmental impacts of delayed cleanup and immediate cleanup described in Section 3.0. The impacts are summarized in Section 5.1. The discussion of the radiological impacts in Section 5.2 includes an estimate of the possible health effects resulting from radiation doses to the hypothetical maximally exposed offsite individual, the population within the 50-mile (80-kilometer) radius, and the TMI-2 cleanup workers. The discussion of nonradiological impacts in Section 5.3 includes consideration of the cost, land commitment, and socioeconomic effects. In Section 5.4 the discussion of potential accidents includes consideration of radiological impacts resulting from accidents at the TMI-2 site and during waste transportation, and nonradiological impacts including traffic accidents, injuries, and fatalities.

The impacts that have been estimated to result from the alternatives considered in this supplement, are consistent with those estimated in the NRC staff's PEIS of March 1981.

5.1 SUMMARY OF THE IMPACTS FOR THE ALTERNATIVES CONSIDERED

Table 5.1 summarizes the expected radiological environmental impacts of routine releases that would result from delayed cleanup and immediate cleanup as evaluated in Section 3.0 (the impact of accidents is discussed in Section 5.4). For each alternative, the table lists the offsite dose pathways/locations in which the dose is incurred; the doses for the maximally exposed offsite individual, the offsite population within the 50-mile (80-kilometer) radius, the population outside the 50-mile (80-kilometer) radius that receives radiation dose that is attributable to the TMI-2 cleanup, the population receiving dose during waste transportation, and the TMI-2 cleanup workers.

For delayed cleanup, the total 50-year dose commitment estimated for the maximally exposed individual is 29 mrem to the bone and 4 mrem to the total body from releases to the atmosphere and 0.1 mrem to the bone and 0.03 mrem to the total body from releases to the Susquehanna River. These doses are based on exposures occurring over a period of 24 years and on a series of conservative assumptions, as discussed in Section 3.0 and Appendix E. For the alternative of immediate cleanup, the total 50-year dose commitment estimated for the maximally exposed individual is 7 mrem to the bone and 0.7 mrem to the total body from releases to the atmosphere and 0.1 mrem to the bone and 0.03 mrem to the body from releases to the river. These doses are based on exposures occurring over a 4-year period. They are also based on a series of conservative assumptions as discussed in Section 3.0 and Appendix E. The doses resulting from delayed or immediate cleanup are in addition to the approximately 87 mrem/yr to the total body (external dose) received by the average Harrisburg resident from natural background (Oakley 1972). Thus, the total body dose to the maximally exposed individual as a result of delayed cleanup is less than 0.2 percent of the background dose

TABLE 5.1. Estimated Radiological Environmental Impacts of Cleanup Alternatives

Section Number and Alternative	Offsite Dose Pathway	50-Year Dose Commitment[a]				
		Maximally Exposed Offsite Individual, mrem	Offsite Population, person-rem			
			Within 50-Mile[b] Radius	Outside 50-Mile Radius	Waste Transportation, Total Body	Occupational, person-rem
3.2 Delayed Cleanup During PDMS[c]					5 to 6	48 to 1500
	Atmosphere	25 bone 3 total body	14 bone 9 total body	2 bone 0.4 total body		
	River	0.01 bone 0.003 total body	0.3 bone 0.009 total body	0.6 bone 0.04 total body		
Cleanup[d] Following PDMS	Atmosphere	4 bone 0.5 total body	3 bone 2 total body	0.1 bone 0.04 total body		
	River	0.1 bone 0.03 total body	3 bone 0.1 total body	7 bone 0.4 total body		
Total[e]	Atmosphere	29 bone 4 total body	17 bone 11 total body	2 bone 0.4 total body		
	River	0.1 bone 0.03 total body	3 bone 0.1 total body	8 bone 0.4 total body		
3.3 Immediate Cleanup[d]					20 to 25	300 to 3100
	Atmosphere	7 bone 0.7 total body	4 bone 2 total body	0.4 bone 0.2 total body		
	River	0.1 bone 0.03 total body	2 bone 0.07 total body	4 bone 0.3 total body		

(a) Doses from offsite burial of low-level wastes are not included.
(b) Includes the dose (for river pathway) from consumption of 12 percent of the annual harvest of Chesapeake Bay shellfish.
(c) Cumulative 50-year dose commitment received over a 20 year period.
(d) Cumulative 50-year dose commitment received over a 4-year cleanup period.
(e) Cumulative 50-year dose commitment received over a 24-year period.

received by this individual during a 24-year period. Immediate cleanup would also result in a total body dose of about 0.2 percent of the background dose received by the maximally exposed individual during a 4-year period.

The total 50-year dose commitment to the population living within 50 miles (80 kilometers) of TMI-2 from delayed cleanup is estimated to be 17 person-rem to the bone and 11 person-rem to the total body from releases to the atmosphere, and 3 person-rem to the bone and 0.1 person-rem to the total body from releases to the Susquehanna River. For the alternative of immediate cleanup, the 50-year dose commitment estimated for the population living within 50 miles (80 kilometers) of TMI-2 is 4 person-rem to the bone and 2 person-rem to the total body from releases to the atmosphere, and 2 person-rem to the bone and 0.07 person-rem to the total body from releases to the river. During delayed cleanup, the population doses are potentially distributed to a population ranging from 2.2 million persons to 3.2 million persons within 50 miles (80 kilometers) of TMI. During immediate cleanup the population doses are distributed to a population of approximately 2.2 million persons within 50 miles (80 kilometers) of TMI. In addition to the doses incurred during cleanup, these populations are expected to receive (external) annual background radiation doses to the total body of approximately 190,000 person-rem per year and 280,000 person-rem per year for 2.2 million and 3.2 million persons, respectively (assuming an average background dose rate of 87 mrem/yr). Thus, the total body dose to the population within the 50-mile (80-kilometer) radius of TMI as a result of delayed cleanup is less than 0.0003 percent of the background dose received by this population during a 24-year period. Immediate cleanup would also result in less than a 0.0003-percent increase in the dose received by the population within a 50-mile (80-kilometer) radius of TMI-2 during a 4-year period.

An additional population of unknown size and geographic distribution lives outside the 50-mile (80-kilometer) radius and receives radiation dose attributable to the TMI-2 cleanup from external exposure, inhalation, the consumption of food exported from within the 50-mile (80-kilometer) radius and consumption of Chesapeake Bay shellfish. Because of the potentially large size of this population, the dose during either of the cleanup alternatives is an even smaller fraction of the background radiation dose than that given above for the population within the 50-mile (80-kilometer) radius.

The population dose due to waste transportation is distributed to truck crews and those persons along the transportation route. This dose ranges from 5 to 6 person-rem (total body) for delayed cleanup to 20 to 25 person-rem (total body) for immediate cleanup. The dose to the persons along the transportation route is also a small fraction of the total dose from background sources that is received by this population.

Occupational dose estimates for delayed cleanup range from 48 to 1500 person-rem and for immediate cleanup from 300 to 3100 person-rem.

Table 5.2 summarizes the nonradiological impacts that could result from delayed cleanup and immediate cleanup as discussed in Section 3.0. These include the cost of implementation (in 1988 dollars), the long-term

TABLE 5.2. Estimated Nonradiological Environmental Impacts of Cleanup Alternatives

Section Number and Alternative	Cost, $ millions(a)	LLW Burial Ground Space, ft^3(b)	Estimated Number of Traffic Accidents
3.2 Delayed Cleanup	200 to 320	33,000 to 74,000	0.5 to 1
3.3 Immediate Cleanup	170 to 240	32,000 to 70,000	1 to 3

(a) Constant 1988 dollars
(b) LLW burial ground space is in cubic feet. For metric equivalents, see text.

commitment of space for radioactive waste burial, and the estimated number of transportation accidents expected during waste shipments.

The cost of implementing delayed cleanup ranges from $200 million to $320 million. The cost for immediate cleanup is estimated at $170 million to $240 million. These costs are in 1988 dollars and include the estimated waste-disposal costs.

Neither delayed cleanup nor immediate cleanup would require any new long-term commitment of land onsite but both require storage space in a LLW commercial burial site. Delayed cleanup would require between 33,000 and 74,000 cubic feet (930 and 2100 cubic meters) of storage space, and immediate cleanup would require between 32,000 and 70,000 cubic feet (910 and 2000 cubic meters) of storage space.

The number of transportation accidents estimated to occur during cleanup ranges from 0.5 to 1 for delayed cleanup, and from 1 to 3 for immediate cleanup. An accident is defined as any form of traffic accident and does not necessarily mean personnel injuries, fatalities, or any disturbance to the cargo. The number of injuries, fatalities, and radiological events resulting from traffic accidents is described in Section 5.4. The number of accidents estimated to result during delayed cleanup is smaller than for immediate cleanup because of the significant reduction in shipping distance assumed to occur if cleanup is delayed until a regional LLW disposal facility is available.

5.2 RANGE OF RADIOLOGICAL IMPACTS AND POSSIBLE HEALTH EFFECTS

In estimating potential health effects from both offsite and occupational radiation exposures as a result of TMI-2 cleanup, the 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 (NAS) Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR

1972; BEIR 1980). 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-rem and 220 potential cases of all forms of genetic disorders per million person-rem.

The cancer-mortality risk estimates are based on the "absolute risk" model described in BEIR I (BEIR 1972). 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 could be 0 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 BEIR III (BEIR 1980).

Values for genetic risk estimators range from 60 to 1100 potential cases of all forms of genetic disorders per million person-rem (BEIR 1980). The value of 220 potential cases for 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 Measurements (NCRP 1975), the NAS (BEIR 1980), and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR 1982).

The risk of potentially fatal cancers in the exposed work-force population is estimated as follows: multiplying the plant-worker-population dose (a range of 48 to 1500 person-rem for delayed cleanup and a range of 300 to 3100 person-rem for immediate cleanup) by the somatic risk estimator (135 potential deaths from cancer per million person-rem), the staff estimates between 0.006 and 0.2 cancer deaths may occur in the total population of exposed workers involved in delayed cleanup and between 0.04 and 0.4 cancer deaths for workers involved in immediate cleanup. The higher value of 0.2 cancer deaths for delayed cleanup means that the probability of 1 cancer death over the lifetime of the entire work force as a result of the delayed cleanup disposal operation is approximately 2 chances in 10. Likewise, the higher value of 0.4 cancer deaths for immediate cleanup means that the probability of 1 cancer death over the lifetime of the entire work force as a result of the immediate cleanup disposal operation is approximately 4 chances in 10. 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.

Conservative estimates of the radiological doses and dose commitments resulting from the cleanup alternatives have been estimated in Section 3.0.

Accurate measurements of radiation and radioactive contaminants can be made with a very high sensitivity so that much smaller amounts of radionuclides 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 the site boundaries, or to the total population outside the boundaries can be readily calculated and recorded. These risk estimates for the cleanup alternatives evaluated are presented below.

The risk to the maximally exposed individual is estimated by multiplying the preceding risk estimator (135 potential deaths from cancer per million person-rem) by the estimated dose to the total body (4 mrem for the delayed cleanup and about 0.7 mrem for immediate cleanup). This calculation results in a risk of potential premature death from cancer to the maximally exposed individual from exposure to radioactive effluents (gaseous or liquid) from delayed cleanup of approximately 0.5 chances in 1 million and for immediate cleanup of approximately 0.1 chances in 1 million. The risk of potential premature death from cancer to the average individual within 50 miles (80 kilometers) of the reactors from exposure to radioactive effluents from TMI-2 is much less than the risk to the maximally exposed individual. These risks are very small in comparison to cancer incidence from causes unrelated to the cleanup of the TMI-2 facility.

Multiplying the dose to the general population within 50 miles (80 kilometers) of TMI-2 from exposure to radioactive effluents (i.e., 11 person-rem to the total body for delayed cleanup and 2 person-rem to the total body for immediate cleanup) by the preceding somatic risk estimator, the staff estimates that fewer than 0.002 cancer deaths (i.e., approximately 2 chances in 1000 of a single fatal cancer) may occur in the entire exposed population from delayed cleanup and fewer than 0.0003 cancer deaths (i.e., approximately 3 chances in 10,000 of a single fatal cancer) from immediate cleanup. The significance of this risk can be determined by comparing it to the total projected incidence of cancer deaths in the population within 50 miles (80 kilometers) of TMI-2. Multiplying the estimated population within 50 miles (80 kilometers) of TMI-2 assumed for the year 1989 (2.2 million people) by the incidence of eventual actual cancer fatalities of about 20 percent (American Cancer Society 1985), the staff estimates that about 440,000 cancer deaths are expected. The incidence of actual cancer fatalities for the year 2009 is unknown.

For purposes of evaluating the potential genetic risks, the progeny of workers are considered members of the general public. However, it is assumed that only about one-third of the occupational radiation dose is received by workers who have offspring after the workers' radiation exposure (e.g., see Paragraph 80 of ICRP 1977). Multiplying the sum of the dose to the

population within 50 miles (80 kilometers) of TMI-2 from exposure to radioactivity attributable to effluent from the delayed cleanup (i.e., 11 person-rem total body, including gonads), and the estimated dose from occupational exposure (i.e., one-third of between 48 and 1500 person-rem) by the preceding genetic risk estimator (220 potential cases of all forms of genetic disorders per million person-rem), the staff estimates that between about 0.006 and 0.1 potential genetic disorders may occur in all future generations of the exposed population from delayed cleanup activities. For immediate cleanup activities, the number of potential genetic disorders is estimated to be between about 0.02 and 0.2 for all future generations of the exposed population. Because BEIR III (BEIR 1980) 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 cleanup operations 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 50 miles (80 kilometers) of the plant (about 2.2 million persons in the year 1989) by the current incidence of actual genetic ill health in each generation (about 11 percent), it is estimated that about 1.2 million genetic abnormalities are expected in the first five generations of the population (BEIR 1980) from causes unrelated to TMI-2 cleanup.

No significant radiological impact is expected on aquatic or terrestrial biota, including endangered species, as a result of either alternative.

5.3 RANGE OF NONRADIOLOGICAL IMPACTS

The major nonradiological impacts identified include the cost of implementation, long-term commitment of land and burial ground space, and the socioeconomic effects. No significant chemical releases are expected for either cleanup alternative.

Cost estimates were made based on staff consideration of the cost of major activities expected for each cleanup alternative. The estimates are not based on an extremely detailed level of information, but they are believed to provide an adequate basis for comparing the cost impact of the cleanup alternatives. The estimated cost of delayed cleanup ranges from $200 million to $320 million and the cost of immediate cleanup from $170 million to $240 million.

Neither alternative requires a new long-term commitment of land at the TMI site. Delayed cleanup would require slightly more disposal space [33,000 to 74,000 cubic feet (930 to 2100 cubic meters)] than immediate cleanup [32,000 to 70,000 cubic feet (910 to 2000 cubic meters)].

The current work force would be reduced from approximately 1150 persons to 100 to 125 persons in the first year of delayed cleanup and to 70 to 75 persons for subsequent years; however, the employment reduction in the surrounding area amounts to only about 0.2 percent of the local baseline employment in the surrounding area. Thus, the socioeconomic impact on the local economy should be minor. The staffing level for completing cleanup

after PDMS would probably be somewhat smaller than the current staffing level although larger than that used during PDMS. The socioeconomic impacts for the immediate cleanup alternative are expected to be minor, essentially maintaining the present economic impact of TMI-2 cleanup or slightly less for a period of 3 to 4 years.

The estimated time commitment for delayed cleanup is 24 years following completion of defueling. This includes an assumed storage period of 20 years and a cleanup period of 4 years following the end of the PDMS period. The immediate cleanup alternative could be completed in about 3 to 4 years following completion of defueling. In either case, additional activities, either decommissioning or refurbishment, would be required following completion of cleanup.

No significant nonradiological impact is expected on aquatic or terrestrial biota, including endangered species, as a result of either cleanup alternative.

5.4 RANGE OF ACCIDENT IMPACTS AND THEIR PROBABILITY

The accident impacts include both radiological impacts resulting from potential accidents at the TMI-2 facility and radiological and nonradiological impacts of accidents during transportation of the waste to a low-level waste site. Table 5.3 lists the major radiological accidents for delayed and immediate cleanup alternatives as well as the resulting dose estimates. The 50-year dose commitments resulting from an accident during delayed cleanup are as follows: 0.005 to 12 mrem to bone and 0.0001 to 2 mrem to the total body for the hypothetical maximally exposed individual; 0.007 to 8 person-rem to bone and 0.0005 to 5 person-rem to the total body for the population within a 50-mile (80-kilometer) radius; and 0.00008 to 0.4 person-rem to bone and 0.000004 to 0.1 person-rem to the total body for the population outside the 50-mile (80-kilometer) radius that receives radiation dose attributable to the TMI-2 cleanup. The 50-year dose commitments resulting from an accident during immediate cleanup is 0.008 to 19 mrem to bone and 0.0004 to 3 mrem to the total body for the maximally exposed individual; 0.009 to 11 person-rem to bone and 0.002 to 8 person-rem to the total body for the population within a 50-mile (80-kilometer) radius; and 0.0003 to 2 person-rem to bone and 0.00002 to 0.6 person-rem to the total body for the population outside the 50-mile (80-kilometer) radius that receives radiation dose attributable to the TMI-2 cleanup.

Table 5.4 lists the major radiological and nonradiological consequences of transportation accidents. An estimated 0.5 to 1 accidents would occur for delayed cleanup, with 0.3 to 0.6 injuries (approximately 3 to 6 chances out of 10 that an injury would occur during the entire shipping program) and 0.02 to 0.05 fatalities (approximately 2 to 5 chances out of 100 that a fatality would occur during the entire shipping program) and a population dose of about 0.0007 person-rem. An estimated 1 to 3 accidents would occur for immediate cleanup, with 1 to 3 injuries, 0.1 to 0.2 fatalities (approximately 1 to 2 chances out of 10 that a fatality would occur during the entire shipping program), and a population dose of about 0.002 person-rem. The

TABLE 5.3. Estimated Environmental Impacts of Radiological Accidents

Section Number and Alternative	Accident Description	Nontransportation Dose: Maximally Exposed Offsite Individual, mrem	Nontransportation Dose: Offsite Population, person-rem: Within 50-Mile Radius(a)	Nontransportation Dose: Offsite Population, person-rem: Outside 50-Mile Radius
3.2 Delayed Cleanup	During storage			
	Fire in stairwell	2 bone 0.2 total body	1 bone 0.6 total body	0.1 bone 0.04 total body
	Cleanup following PDMS			
	Fire in stairwell	0.01 bone 0.001 total body	0.007 bone 0.004 total body	0.0003 bone 0.0001 total body
	HEPA filter failure	12 bone 1 total body	8 bone 5 total body	0.4 bone 0.1 total body
	Decontamination liquid spill	0.005 bone 0.0001 total body	0.009 bone 0.0005 total body	0.00008 bone 0.000004 total body
	Storage tank rupture	0.01 bone 0.002 total body	0.3 bone 0.008 total body	0.7 bone 0.02 total body
3.3 Immediate Cleanup	Fire in stairwell	0.02 bone 0.002 total body	0.01 bone 0.006 total body	0.001 bone 0.0004 total body
	HEPA filter failure	19 bone 2 total body	11 bone 8 total body	2 bone 0.6 total body
	Decontamination liquid spill	0.008 bone 0.0004 total body	0.01 bone 0.002 total body	0.0003 bone 0.00001 total body
	Storage tank rupture	0.01 bone 0.002 total body	0.2 bone 0.006 total body	0.4 bone 0.009 total body

(a) Includes the dose (for river pathway) from consumption of 12 percent of the annual harvest of Chesapeake Bay shellfish.

TABLE 5.'. Estimated Radiological and Nonradiological Impacts from Offsite-Shipment Truck Accidents

Section Number and Alternative	Radiological Impacts Population Dose, person-rem	Radiological Impacts, Estimated Number		
		Accidents	Injuries	Fatalities
3.2 Delayed Cleanup	0.0007	0.5 to 1	0.3 to 0.6	0.02 to 0.05
3.3 Immediate Cleanup	0.002	1 to 3	1 to 3	0.1 to 0.2

number of accidents, injuries, and fatalities estimated during delayed cleanup is smaller than for immediate cleanup because of the significant reduction in shipping distance assumed to occur if cleanup is delayed until a regional LLW disposal facility is available.

6.0 CONCLUSIONS

On the basis of a review of the environmental impacts associated with the licensee's proposed cleanup method, involving facility storage followed by completion of cleanup (i.e., delayed cleanup or post-defueling monitored storage), and NRC staff-identified alternatives, including completion of cleanup without a storage period (i.e., immediate cleanup), the staff concludes:

- The licensee's proposed plan and the NRC staff-identified alternatives for completion of the TMI-2 cleanup are within applicable regulatory limits and could each be implemented without significant environmental impact. The potential health impact on both workers and the offsite public from any of the cleanup alternatives is very small.

- Neither of the two alternatives evaluated quantitatively (delayed cleanup and immediate cleanup) is clearly preferable from the perspective of environmental impacts. Although the quantitative estimate of potential impacts varied between alternatives, these differences are not judged sufficiently large to allow for identification of a clearly preferable alternative.

- The alternative of "no further cleanup following defueling" (or "no action"), required by the National Environmental Policy Act of 1969 to be considered as part of environmental impact statements, is not acceptable because this course would not result in the completion of cleanup and the elimination of the public health and safety risk associated with the damaged TMI-2 facility.

- The environmental impacts estimated for the cleanup alternatives evaluated in this supplement fall within the range of impacts estimated in the NRC staff's original Programmatic Environmental Impact Statement (NRC 1981) on the cleanup.

- The licensee's proposed action involving post-defueling monitored storage of the TMI-2 facility followed by completion of cleanup is environmentally acceptable.

Therefore, the staff finds that the benefits of cleanup outweigh the small associated impacts and that the licensee's proposed approach to completing the cleanup will not significantly affect the quality of the human environment.

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8.0 INDEX

APPENDIX A

RESERVED FOR COMMENTS ON THE DRAFT SUPPLEMENT TO THE PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT

APPENDIX B

CONTRIBUTORS TO THE SUPPLEMENT

APPENDIX B

CONTRIBUTORS TO THE SUPPLEMENT

The overall responsibility for the preparation of this supplement was assigned to the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. The statement was prepared by members of the Office of Nuclear Reactor Regulation with assistance from other NRC organizations and the Pacific Northwest Laboratory. The major contributors to the draft supplement, their affiliations, and function or expertise are listed below.

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APPENDIX C

MAXIMUM PERMISSIBLE CONCENTRATIONS IN AIR AND WATER

APPENDIX C

MAXIMUM PERMISSIBLE CONCENTRATIONS IN AIR AND WATER

The regulations in 10 CFR 20, "Standards for Protection Against Radiation," specify the allowable concentrations for discharge of radioactivity in effluents to air and water in unrestricted areas. Table C.1 lists the maximum permissible concentrations in air and water in unrestricted areas for those isotopes present in the TMI-2 facility following defueling. The maximum permissible concentrations, obtained from 10 CFR 20, Appendix B, Table II, are concentrations above background.

TABLE C.1. Maximum Permissible Concentrations in Air and Water Above Background in Unrestricted Areas(a) (from 10 CFR 20, Appendix B, Table II)

Isotope(b)		Air, μCi/mL	Water, μCi/mL
Carbon-14	S	1×10^{-7}	8×10^{-4}
	Sub(c)	1×10^{-6}	---(d)
Manganese-54	S	1×10^{-8}	1×10^{-4}
	I	1×10^{-9}	1×10^{-4}
Iron-55	S	3×10^{-8}	8×10^{-4}
	I	3×10^{-8}	2×10^{-3}
Cobalt-60	S	1×10^{-8}	5×10^{-5}
	I	3×10^{-10}	3×10^{-5}

(a) When more than one radionuclide is present, the sum of the concentrations of each radionuclide, divided by the concentration in the table, must be less than or equal to 1 (10 CFR 20, Appendix B, Footnote 1).

(b) S = soluble
I = insoluble
Sub = submersion in a semispherical infinite cloud of airborne material.

(c) As carbon dioxide, CO_2.

(d) "---" indicates no value was given.

TABLE C.1. contd

Isotope(b)		Air, μCi/mL	Water, μCi/mL
Nickel-63	S	2 x 10-9	3 x 10-5
	I	1 x 10-8	7 x 10-4
Selenium-79		1 x 10-10	3 x 10-6
Krypton-85	Sub	3 x 10-7	---(d)
Strontium-90	S	3 x 10-11	3 x 10-7
	I	2 x 10-10	4 x 10-5
Yttrium-90	S	4 x 10-9	2 x 10-5
	I	3 x 10-9	2 x 10-5
Zirconium-93	S	4 x 10-9	8 x 10-4
	I	1 x 10-8	8 x 10-4
Niobium-93m	S	4 x 10-9	4 x 10-4
	I	5 x 10-9	4 x 10-4
Technetium-99	S	7 x 10-8	3 x 10-4
	I	2 x 10-9	2 x 10-4
Ruthenium-106	S	3 x 10-9	1 x 10-5
	I	2 x 10-10	1 x 10-5
Rhodium-106	Sub	3 x 10-8	---(d)
Cadmium-113m		1 x 10-10	3 x 10-6
Tin-126		1 x 10-10	3 x 10-6
Antimony-125	S	2 x 10-8	1 x 10-4
	I	9 x 10-10	1 x 10-4
Antimony-126m	Sub	3 x 10-8	---(d)
Antimony-126		1 x 10-10	3 x 10-6
Tellurium-125m	S	1 x 10-8	2 x 10-4
	I	4 x 10-9	1 x 10-4
Cesium-134	S	1 x 10-9	9 x 10-6
	I	4 x 10-10	4 x 10-5
Cesium-135	S	2 x 10-8	1 x 10-4
	I	3 x 10-9	2 x 10-4

TABLE C.1. contd

Isotope(b)		Air, μCi/mL	Water, μCi/mL
Cesium-137	S	2 x 10-9	2 x 10-5
	I	5 x 10-10	4 x 10-5
Barium-137m	Sub	3 x 10-8	---(d)
Cerium-144	S	3 x 10-10	1 x 10-5
	I	2 x 10-10	1 x 10-5
Praseodymium-144	Sub	3 x 10-8	---(d)
Promethium-147	S	2 x 10-9	2 x 10-4
	I	3 x 10-9	2 x 10-4
Samarium-151	S	2 x 10-9	4 x 10-4
	I	5 x 10-9	4 x 10-4
Europium-152	S	4 x 10-10	8 x 10-5
	I	6 x 10-10	8 x 10-5
Europium-154	S	2 x 10-10	2 x 10-5
	I	2 x 10-10	2 x 10-5
Europium-155	S	3 x 10-9	2 x 10-4
	I	3 x 10-9	2 x 10-4
Thorium-231	S	5 x 10-8	2 x 10-4
	I	4 x 10-8	2 x 10-4
Thorium-234	S	2 x 10-9	2 x 10-5
	I	1 x 10-9	2 x 10-5
Protactinium-234m	Sub	1 x 10-6	3 x 10-8
Uranium-234	S	2 x 10-11	3 x 10-5
	I	4 x 10-12	3 x 10-5
Uranium-235	S	2 x 10-11	3 x 10-5
	I	4 x 10-12	3 x 10-5
Uranium-236	S	2 x 10-11	3 x 10-5
	I	4 x 10-12	3 x 10-5
Uranium-238	S	3 x 10-12	4 x 10-5
	I	5 x 10-12	4 x 10-5
Plutonium-238	S	7 x 10-14	5 x 10-6
	I	1 x 10-12	3 x 10-5

TABLE C.1. contd

Isotope(b)		Air, μCi/mL	Water, μCi/mL
Plutonium-239	S	6×10^{-14}	5×10^{-6}
	I	1×10^{-12}	3×10^{-5}
Plutonium-240	S	6×10^{-14}	5×10^{-6}
	I	1×10^{-12}	3×10^{-5}
Plutonium-241	S	3×10^{-12}	2×10^{-4}
	I	1×10^{-9}	1×10^{-3}
Americium-241	S	2×10^{-13}	4×10^{-6}
	I	4×10^{-12}	3×10^{-5}

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APPENDIX D

ESTIMATED RADIONUCLIDE RELEASES AND RELEASE RATES

APPENDIX D

ESTIMATED RADIONUCLIDE RELEASES AND RELEASE RATES

This appendix provides the calculated release rates (Ci/yr) for routine releases of radioactive material and the calculated releases (curies) for accidental releases of radioactive material from the TMI-2 facility. The calculated releases are based on the information given in Section 3.0. The releases estimated for delayed cleanup are given in Tables D.1 through D.9. Releases estimated for immediate cleanup are given in Tables D.10 through D.15.

TABLE D.1. Routine Atmospheric Release Rates During Post-Defueling Monitored Storage

Radionuclide	20-Year Release, First-Year Release Rate,[a] Ci/yr
Carbon-14	2.9 x 10-7
Selenium-79	6.5 x 10-8
Krypton-85	3.1 x 10-8
Strontium-90/Yttrium-90	1.1 x 10-3
Zirconium-93	7.9 x 10-10
Niobium-93m	1.2 x 10-7
Technetium-99	2.1 x 10-6
Ruthenium-106/Rhodium-106	1.2 x 10-3
Cadmium-113m	3.8 x 10-8
Antimony-125	2.2 x 10-4
Tellurium-125m	5.7 x 10-5
Tin-126/Antimony-126m/Antimony-126	4.4 x 10-8
Cesium-134	1.2 x 10-4
Cesium-135	4.1 x 10-8
Cesium-137/Barium-137m	1.2 x 10-2
Cerium-144	1.9 x 10-7
Praseodymium-144	1.9 x 10-7
Promethium-147	9.2 x 10-6
Samarium-151	3.2 x 10-4
Europium-152	1.3 x 10-9
Europium-154	1.6 x 10-7
Europium-155	5.9 x 10-7
Uranium-234	5.9 x 10-9
Uranium-235/Thorium-231	1.9 x 10-10
Uranium-236	1.5 x 10-10
Uranium-237	1.1 x 10-10
Uranium-238/Thorium-234/Protactinium-234m	1.3 x 10-9
Plutonium-238	3.3 x 10-8
Plutonium-239	4.4 x 10-7
Plutonium-240	1.1 x 10-7
Plutonium-241	4.9 x 10-6
Americium-241	9.3 x 10-8

(a) Release rates for subsequent years are based on the first-year release rate and account for radioactive decay.

TABLE D.2. Routine Liquid Release Rates to the Susquehanna River During Post-Defueling Monitored Storage

Radionuclide	20-Year Release Rate, Ci/yr
Carbon-14	1.9×10^{-3}
Selenium-79	1.5×10^{-6}
Strontium-90/Yttrium-90	1.9×10^{-4}
Niobium-93m	1.5×10^{-6}
Technetium-99	1.9×10^{-5}
Ruthenium-106/Rhodium-106	6.3×10^{-6}
Cadmium-113m	1.5×10^{-6}
Antimony-125	4.4×10^{-6}
Tellurium-125m	4.4×10^{-6}
Tin-126/Antimony-126m/Antimony-126	1.5×10^{-6}
Cesium-134	1.7×10^{-6}
Cesium-135	1.5×10^{-6}
Cesium-137/Barium-137m	7.6×10^{-5}
Samarium-151	1.5×10^{-6}

TABLE D.3. Routine Atmospheric Release Rates During Cleanup Following Post-Defueling Monitored Storage

Radionuclide	4-Year Release Rate, Ci/yr
Carbon-14	4.3×10^{-7}
Iron-55	3.9×10^{-7}
Cobalt-60	1.6×10^{-5}
Nickel-63	4.1×10^{-5}
Selenium-79	9.2×10^{-8}
Krypton-85	7.0×10^{-2}
Strontium-90/Yttrium-90	7.4×10^{-4}
Zirconium-93	2.6×10^{-9}
Niobium-93m	6.4×10^{-8}
Technetium-99	2.9×10^{-6}
Ruthenium-106/Rhodium-106	1.7×10^{-9}
Cadmium-113m	2.0×10^{-8}
Antimony-125	1.9×10^{-6}
Tellurium-125m	4.4×10^{-7}
Tin-126/Antimony-126m/Antimony-126	6.2×10^{-8}
Cesium-134	1.9×10^{-7}
Cesium-135	5.9×10^{-8}
Cesium-137/Barium-137m	1.1×10^{-2}
Promethium-147	1.6×10^{-7}
Samarium-151	3.9×10^{-4}
Europium-152	1.5×10^{-9}
Europium-154	1.1×10^{-7}
Europium-155	1.1×10^{-7}
Uranium-234	2.0×10^{-8}
Uranium-235/Thorium-231	6.5×10^{-10}
Uranium-236	5.1×10^{-10}
Uranium-237	1.5×10^{-10}
Uranium-238/Thorium-234/Protactinium-234m	4.3×10^{-9}
Plutonium-238	9.5×10^{-8}
Plutonium-239	1.5×10^{-6}
Plutonium-240	3.8×10^{-7}
Plutonium-241	6.1×10^{-6}
Americium-241	6.2×10^{-7}

TABLE D.4. Routine Liquid Release Rates to the Susquehanna River During Cleanup Following Post-Defueling Monitored Storage

Radionuclide	4-Year Release Rate, Ci/yr
Carbon-14	9.5×10^{-2}
Iron-55	4.5×10^{-4}
Cobalt-60	4.5×10^{-4}
Nickel-63	5.7×10^{-4}
Selenium-79	7.4×10^{-5}
Strontium-90/Yttrium-90	9.5×10^{-3}
Zirconium-93	7.4×10^{-5}
Niobium-93m	7.4×10^{-5}
Technetium-99	9.5×10^{-4}
Ruthenium-106/Rhodium-106	3.1×10^{-4}
Cadmium-113m	7.4×10^{-5}
Antimony-125	2.2×10^{-4}
Tellurium-125m	2.2×10^{-4}
Tin-126/Antimony-126m/Antimony-126	7.4×10^{-5}
Cesium-134	8.3×10^{-5}
Cesium-135	7.4×10^{-5}
Cesium-137/Barium-137m	3.8×10^{-3}
Promethium-147	4.5×10^{-3}
Samarium-151	7.4×10^{-5}
Europium-152	3.6×10^{-7}
Europium-154	4.2×10^{-5}
Europium-155	1.0×10^{-4}
Uranium-234	9.5×10^{-6}
Uranium-235/Thorium-231	1.1×10^{-5}
Uranium-236	3.8×10^{-6}
Uranium-237	7.4×10^{-5}
Uranium-238/Thorium-234/Protactinium-234m	1.1×10^{-5}
Plutonium-238	1.1×10^{-5}
Plutonium-239	1.3×10^{-5}
Plutonium-240	1.3×10^{-5}
Plutonium-241	6.2×10^{-4}
Americium-241	1.1×10^{-5}

TABLE D.5. Postulated Accidental Atmospheric Release from a Fire in the Elevator/Stairwell Structure During Post-Defueling Monitored Storage

Radionuclide	Release, Ci
Carbon-14	4.7×10^{-7}
Selenium-79	9.9×10^{-8}
Krypton-85	1.1×10^{-7}
Strontium-90/Yttrium-90	9.2×10^{-4}
Zirconium-93	2.9×10^{-9}
Niobium-93m	1.9×10^{-7}
Technetium-99	3.2×10^{-6}
Ruthenium-106/Rhodium-106	1.8×10^{-3}
Cadmium-113m	5.8×10^{-8}
Antimony-125	3.4×10^{-4}
Tellurium-125m	8.6×10^{-5}
Tin-126/Antimony-126m/Antimony-126	6.7×10^{-8}
Cesium-134	1.7×10^{-4}
Cesium-135	6.5×10^{-8}
Cesium-137/Barium-137m	1.9×10^{-2}
Cerium-144	7.0×10^{-7}
Praseodymium-144	6.5×10^{-7}
Promethium-147	3.3×10^{-5}
Samarium-151	4.9×10^{-4}
Europium-152	4.7×10^{-9}
Europium-154	6.0×10^{-7}
Europium-155	2.1×10^{-6}
Uranium-234	2.1×10^{-8}
Uranium-235/Thorium-231	7.0×10^{-10}
Uranium-236	5.5×10^{-10}
Uranium-237	4.1×10^{-10}
Uranium-238/Thorium-234/Protactinium-234m	4.6×10^{-9}
Plutonium-238	1.2×10^{-7}
Plutonium-239	1.6×10^{-6}
Plutonium-240	4.2×10^{-7}
Plutonium-241	1.8×10^{-5}
Americium-241	3.3×10^{-7}

TABLE D.6. Postulated Accidental Atmospheric Release from a Fire in the Elevator/Stairwell Structure During Cleanup Following Post-Defueling Monitored Storage

Radionuclide	Release, Ci
Carbon-14	4.6×10^{-9}
Selenium-79	9.9×10^{-10}
Krypton-85	3.1×10^{-10}
Strontium-90/Yttrium-90	5.7×10^{-6}
Zirconium-93	2.9×10^{-11}
Niobium-93m	7.0×10^{-10}
Technetium-99	3.2×10^{-8}
Ruthenium-106/Rhodium-106	1.8×10^{-11}
Cadmium-113m	2.1×10^{-10}
Antimony-125	2.1×10^{-8}
Tellurium-125m	4.7×10^{-9}
Tin-126/Antimony-126m/Antimony-126	6.7×10^{-10}
Cesium-134	2.1×10^{-9}
Cesium-135	6.5×10^{-10}
Cesium-137/Barium-137m	1.2×10^{-4}
Promethium-147	1.7×10^{-9}
Samarium-151	4.2×10^{-6}
Europium-152	1.6×10^{-11}
Europium-154	1.2×10^{-9}
Europium-155	1.2×10^{-9}
Uranium-234	2.1×10^{-10}
Uranium-235/Thorium-231	7.0×10^{-12}
Uranium-236	5.5×10^{-12}
Uranium-237	1.5×10^{-12}
Uranium-238/Thorium-234/Protactinium-234m	4.8×10^{-11}
Plutonium-238	1.1×10^{-9}
Plutonium-239	1.6×10^{-8}
Plutonium-240	4.2×10^{-9}
Plutonium-241	6.2×10^{-8}
Americium-241	6.8×10^{-9}

TABLE D.7. Postulated Accidental Atmospheric Release from a HEPA Filter Failure During Cleanup Following Post-Defueling Monitored Storage

Radionuclide	Release, Ci
Carbon-14	5.8×10^{-6}
Selenium-79	1.2×10^{-6}
Strontium-90/Yttrium-90	6.8×10^{-3}
Niobium-93m	8.5×10^{-7}
Technetium-99	4.0×10^{-5}
Ruthenium-106/Rhodium-106	2.3×10^{-8}
Cadmium-113m	2.6×10^{-7}
Antimony-125	2.6×10^{-5}
Tellurium-125m	5.9×10^{-6}
Tin-126/Antimony-126m/Antimony-126	8.4×10^{-7}
Cesium-134	2.6×10^{-6}
Cesium-135	7.9×10^{-7}
Cesium-137/Barium-137m	1.5×10^{-1}
Samarium-151	5.3×10^{-3}

TABLE D.8. Postulated Accidental Atmospheric Release from a Spill of Reactor-Coolant-System Decontamination Solution During Cleanup Following Post-Defueling Monitored Storage

Radionuclide	Release, Ci
Carbon-14	2.5×10^{-10}
Iron-55	4.7×10^{-8}
Cobalt-60	1.9×10^{-6}
Nickel-63	4.9×10^{-6}
Selenium-79	5.3×10^{-11}
Krypton-85	2.7×10^{-9}
Strontium-90/Yttrium-90	5.3×10^{-6}
Zirconium-93	2.5×10^{-10}
Niobium-93m	9.5×10^{-11}
Technetium-99	1.7×10^{-9}
Cadmium-113m	1.2×10^{-11}
Antimony-125	2.8×10^{-10}
Tellurium-125m	2.8×10^{-10}
Tin-126/Antimony-126m/Antimony-126	3.7×10^{-11}
Cesium-134	2.8×10^{-11}
Cesium-135	8.6×10^{-12}
Cesium-137/Barium-137m	1.5×10^{-6}
Promethium-147	1.5×10^{-8}
Samarium-151	2.3×10^{-7}
Europium-152	1.4×10^{-10}
Europium-154	1.1×10^{-8}
Europium-155	1.1×10^{-8}
Uranium-234	1.9×10^{-9}
Uranium-235/Thorium-231	6.3×10^{-11}
Uranium-236	5.0×10^{-11}
Uranium-237	1.4×10^{-11}
Uranium-238/Thorium-234/Protactinium-234m	4.2×10^{-10}
Plutonium-238	9.2×10^{-9}
Plutonium-239	1.4×10^{-7}
Plutonium-240	3.7×10^{-8}
Plutonium-241	6.0×10^{-7}
Americium-241	6.2×10^{-8}

TABLE D.9. Postulated Accidental Liquid Release from a Ruptured Storage Tank During Cleanup Following Post-Defueling Monitored Storage

Radionuclide	Release, Ci
Carbon-14	4.2×10^{-3}
Iron-55	2.0×10^{-5}
Cobalt-60	2.0×10^{-5}
Nickel-63	2.5×10^{-5}
Selenium-79	3.3×10^{-6}
Strontium-90/Yttrium-90	4.6×10^{-3}
Zirconium-93	3.3×10^{-6}
Niobium-93m	3.3×10^{-6}
Technetium-99	4.2×10^{-5}
Ruthenium-106/Rhodium-106	1.4×10^{-5}
Cadmium-113m	3.3×10^{-6}
Antimony-125	9.7×10^{-5}
Tellurium-125m	9.7×10^{-5}
Tin-126/Antimony-126m/Antimony-126	3.3×10^{-6}
Cesium-134	3.7×10^{-5}
Cesium-135	3.3×10^{-6}
Cesium-137/Barium-137m	1.6×10^{-3}
Promethium-147	2.0×10^{-4}
Samarium-151	3.3×10^{-6}
Europium-152	1.6×10^{-8}
Europium-154	1.8×10^{-6}
Europium-155	4.6×10^{-6}
Uranium-234	4.2×10^{-7}
Uranium-235/Thorium-231	5.0×10^{-7}
Uranium-236	1.7×10^{-7}
Uranium-237	3.3×10^{-6}
Uranium-238/Thorium-234/Protactinium-234m	5.0×10^{-7}
Plutonium-238	5.0×10^{-7}
Plutonium-239	5.9×10^{-7}
Plutonium-240	5.9×10^{-7}
Plutonium-241	2.7×10^{-5}
Americium-241	5.0×10^{-7}

TABLE D.10. Routine Atmospheric Release Rates During Immediate Cleanup

Radionuclide	4-Year Release Rate, Ci/yr
Carbon-14	4.3×10^{-7}
Manganese-54	1.0×10^{-7}
Iron-55	7.0×10^{-5}
Cobalt-60	2.3×10^{-4}
Nickel-63	4.7×10^{-5}
Selenium-79	9.2×10^{-8}
Krypton-85	2.5×10^{-1}
Strontium-90/Yttrium-90	1.2×10^{-3}
Zirconium-93	2.6×10^{-9}
Niobium-93m	1.8×10^{-7}
Technetium-99	2.9×10^{-6}
Ruthenium-106/Rhodium-106	1.7×10^{-3}
Cadmium-113m	5.4×10^{-8}
Antimony-125	3.2×10^{-4}
Tellurium-125m	8.1×10^{-5}
Tin-126/Antimony-126m/Antimony-126	6.2×10^{-8}
Cesium-134	1.6×10^{-4}
Cesium-135	5.9×10^{-8}
Cesium-137/Barium-137m	1.8×10^{-2}
Cerium-144	6.3×10^{-7}
Praseodymium-144	6.1×10^{-7}
Promethium-147	3.0×10^{-5}
Samarium-151	1.1×10^{-6}
Europium-152	4.2×10^{-9}
Europium-154	5.6×10^{-7}
Europium-155	1.9×10^{-6}
Uranium-234	2.0×10^{-8}
Uranium-235/Thorium-231	6.5×10^{-10}
Uranium-236	5.1×10^{-10}
Uranium-237	3.7×10^{-10}
Uranium-238/Thorium-234/Protactinium-234m	4.3×10^{-9}
Plutonium-238	1.1×10^{-7}
Plutonium-239	1.5×10^{-6}
Plutonium-240	3.8×10^{-7}
Plutonium-241	1.6×10^{-5}
Americium-241	3.0×10^{-7}

TABLE D.11. Routine Liquid Release Rates to the Susquehanna River During Immediate Cleanup

Radionuclide	4-Year Release Rate, Ci/yr
Carbon-14	9.5×10^{-2}
Manganese-54	3.8×10^{-5}
Iron-55	4.5×10^{-4}
Cobalt-60	4.5×10^{-4}
Nickel-63	5.7×10^{-4}
Selenium-79	7.4×10^{-5}
Strontium-90/Yttrium-90	9.5×10^{-3}
Zirconium-93	7.4×10^{-5}
Niobium-93m	7.4×10^{-5}
Technetium-99	9.5×10^{-4}
Ruthenium-106/Rhodium-106	3.1×10^{-4}
Cadmium-113m	7.4×10^{-5}
Antimony-125	2.2×10^{-4}
Tellurium-125m	2.2×10^{-4}
Tin-126/Antimony-126m/Antimony-126	7.4×10^{-5}
Cesium-134	8.3×10^{-5}
Cesium-135	7.4×10^{-5}
Cesium-137/Barium-137m	3.8×10^{-3}
Cerium-144	1.7×10^{-3}
Praseodymium-144	7.4×10^{-5}
Promethium-147	4.5×10^{-3}
Samarium-151	7.4×10^{-5}
Europium-152	3.6×10^{-7}
Europium-154	4.2×10^{-5}
Europium-155	1.0×10^{-4}
Uranium-234	9.5×10^{-6}
Uranium-235/Thorium-231	1.1×10^{-5}
Uranium-236	3.8×10^{-6}
Uranium-237	7.4×10^{-5}
Uranium-238/Thorium-234/Protactinium-234m	1.1×10^{-5}
Plutonium-238	1.1×10^{-5}
Plutonium-239	1.3×10^{-5}
Plutonium-240	1.3×10^{-5}
Plutonium-241	6.2×10^{-4}
Americium-241	1.1×10^{-5}

TABLE D.12. Postulated Accidental Atmospheric Release from a Fire in the Elevator/Stairwell Structure During Immediate Cleanup

Radionuclide	Release, Ci
Carbon-14	4.7×10^{-9}
Selenium-79	9.9×10^{-10}
Krypton-85	1.1×10^{-9}
Strontium-90/Yttrium-90	9.2×10^{-6}
Zirconium-93	2.9×10^{-11}
Niobium-93m	1.9×10^{-9}
Technetium-99	3.2×10^{-8}
Ruthenium-106/Rhodium-106	1.8×10^{-5}
Cadmium-113m	5.8×10^{-10}
Antimony-125	3.4×10^{-6}
Tellurium-125m	8.6×10^{-7}
Tin-126/Antimony-126m/Antimony-126	6.7×10^{-10}
Cesium-134	1.7×10^{-6}
Cesium-135	6.5×10^{-10}
Cesium-137/Barium-137m	1.9×10^{-4}
Cerium-144	7.0×10^{-9}
Praseodymium-144	6.5×10^{-9}
Promethium-147	3.3×10^{-7}
Samarium-151	4.9×10^{-6}
Europium-152	4.7×10^{-11}
Europium-154	6.0×10^{-9}
Europium-155	2.1×10^{-8}
Uranium-234	2.1×10^{-10}
Uranium-235/Thorium-231	7.0×10^{-12}
Uranium-236	5.5×10^{-12}
Uranium-237	4.1×10^{-12}
Uranium-238/Thorium-234/Protactinium-234m	4.6×10^{-11}
Plutonium-238	1.2×10^{-9}
Plutonium-239	1.6×10^{-8}
Plutonium-240	4.2×10^{-9}
Plutonium-241	1.8×10^{-7}
Americium-241	3.3×10^{-9}

TABLE D.13. Postulated Accidental Atmospheric Release from a HEPA Filter Failure During Immediate Cleanup

Radionuclide	Release, Ci
Carbon-14	5.7×10^{-6}
Selenium-79	1.2×10^{-6}
Strontium-90/Yttrium-90	1.1×10^{-1}
Niobium-93m	2.4×10^{-6}
Technetium-99	3.9×10^{-5}
Ruthenium-106/Rhodium-106	2.3×10^{-2}
Cadmium-113m	7.2×10^{-7}
Antimony-125	4.3×10^{-3}
Tellurium-125m	1.1×10^{-3}
Tin-126/Antimony-126m/Antimony-126	8.3×10^{-7}
Cesium-134	2.1×10^{-3}
Cesium-135	7.9×10^{-7}
Cesium-137/Barium-137m	2.4×10^{-1}
Samarium-151	6.1×10^{-3}

TABLE D.14. Postulated Accidental Atmospheric Release from a Spill of Reactor Coolant System Decontamination Solution During Immediate Cleanup

Radionuclide	Release, Ci
Carbon-14	2.5×10^{-10}
Manganese-54	1.2×10^{-8}
Iron-55	8.3×10^{-6}
Cobalt-60	2.7×10^{-5}
Nickel-63	5.6×10^{-6}
Selenium-79	5.2×10^{-11}
Krypton-85	1.0×10^{-8}
Strontium-90/Yttrium-90	8.5×10^{-6}
Zirconium-93	2.5×10^{-10}
Niobium-93m	2.5×10^{-10}
Technetium-99	1.8×10^{-9}
Ruthenium-106/Rhodium-106	5.9×10^{-7}
Cadmium-113m	3.1×10^{-11}
Antimony-125	4.7×10^{-8}
Tellurium-125m	4.7×10^{-8}
Tin-126/Antimony-126m/Antimony-126	3.6×10^{-11}
Cesium-134	2.3×10^{-8}
Cesium-135	8.6×10^{-12}
Cesium-137/Barium-137m	2.5×10^{-6}
Cerium-144	6.1×10^{-8}
Praseodymium-144	6.0×10^{-8}
Promethium-147	2.9×10^{-6}
Samarium-151	2.6×10^{-7}
Europium-152	4.1×10^{-10}
Europium-154	5.3×10^{-8}
Europium-155	1.9×10^{-7}
Uranium-234	1.9×10^{-9}
Uranium-235/Thorium-231	6.3×10^{-11}
Uranium-236	5.0×10^{-11}
Uranium-237	3.6×10^{-11}
Uranium-238/Thorium-234/Protactinium-234m	4.2×10^{-10}
Plutonium-238	1.1×10^{-8}
Plutonium-239	1.4×10^{-7}
Plutonium-240	3.7×10^{-8}
Plutonium-241	1.6×10^{-6}
Americium-241	3.0×10^{-8}

TABLE D.15. Postulated Accidental Liquid Release from a Ruptured Storage Tank During Immediate Cleanup

Radionuclide	Release, Ci
Carbon-14	4.2×10^{-3}
Manganese-54	1.7×10^{-6}
Iron-55	2.0×10^{-5}
Cobalt-60	2.0×10^{-5}
Nickel-63	2.5×10^{-5}
Selenium-79	3.3×10^{-6}
Strontium-90/Yttrium-90	4.6×10^{-3}
Zirconium-93	3.3×10^{-6}
Niobium-93m	3.3×10^{-6}
Technetium-99	4.2×10^{-5}
Ruthenium-106/Rhodium-106	1.4×10^{-5}
Cadmium-113m	3.3×10^{-6}
Antimony-125	9.7×10^{-5}
Tellurium-125m	9.7×10^{-5}
Tin-126/Antimony-126m/Antimony-126	3.3×10^{-6}
Cesium-134	3.7×10^{-5}
Cesium-135	3.3×10^{-6}
Cesium-137/Barium-137m	1.6×10^{-3}
Cerium-144	7.6×10^{-5}
Praseodymium-144	3.3×10^{-6}
Promethium-147	2.0×10^{-4}
Samarium-151	3.3×10^{-6}
Europium-152	1.6×10^{-8}
Europium-154	1.8×10^{-6}
Europium-155	4.6×10^{-6}
Uranium-234	4.2×10^{-7}
Uranium-235/Thorium-231	5.0×10^{-7}
Uranium-236	1.7×10^{-7}
Uranium-237	3.3×10^{-6}
Uranium-238/Thorium-234/Protactinium-234m	5.0×10^{-7}
Plutonium-238	5.0×10^{-7}
Plutonium-239	5.9×10^{-7}
Plutonium-240	5.9×10^{-7}
Plutonium-241	2.7×10^{-5}
Americium-241	5.0×10^{-7}

APPENDIX E

CALCULATION OF RADIATION DOSES FROM WATERBORNE AND AIRBORNE PATHWAYS

APPENDIX E

CALCULATION OF RADIATION DOSES

FROM WATERBORNE AND AIRBORNE PATHWAYS

This appendix contains the methods, assumptions, and parameters used in the calculation of the radiation exposure to the public. The pathways are divided into two groups: waterborne pathways from the TMI site and airborne pathways from the TMI site.

E.1 WATERBORNE PATHWAYS

The public radiation doses resulting from the release of accident-generated water to the Susquehanna River were generated by the NRC's LADTAP II computer code (Strenge, Peloquin, and Whelan 1986). The LADTAP II code generates 50-year dose commitments based on a 1-year release. For the cases where the release occurs for a period longer than 1 year, the 50-year dose commitment for a 1-year release was multiplied by the number of years over which the release extends. Doses were determined for the maximum individual, for the population within a 50-mile (80-kilometer) radius of the power plant, and for the population that consumes shellfish harvested from Chesapeake Bay.

The pathways considered for doses to the maximally exposed individual and the population were drinking water obtained from the Susquehanna River, the consumption of fish from the river, rivershore activities, and boating and swimming in the river. The irrigated farm product/food pathway was not applied to the dose calculations because the river water is not commonly used for irrigation purposes.

The affected population within the 50-mile (80-kilometer) radius for 1989 was assumed to be 2.2 million people with age-group distributions as follows: 71 percent, adults; 11 percent, teenagers; and 18 percent, children. Of the 2.2 million people, 300,000 were assumed to obtain their drinking water from the river. The affected population within the 50-mile (80-kilometer) radius was estimated for the year 2009 to number 3.2 million people with the same age-group distribution assumed for 1989. Only 440,000 of the 3.2 million people were assumed to obtain their drinking water from the river.

Table E.1 contains the consumption and usage rates by the maximally exposed individual for the various pathways. Table E.2 lists the consumption rates for drinking water and river fish used for the population dose calculations. Additional parameters used for the population doses for 1989 are as follows:

- shoreline usage - 83,000 person-hours/yr
- swimming - 120,000 person-hours/yr
- boating - 520,000 person-hours/yr
- sport fishing (edible) yield - 150,000 lb/yr (68,000 kg/yr)
- commercial fishing yield - none assumed

Additional parameters used for the population doses for 2009 are as follows:

- shoreline usage - 126,000 person-hours/yr
- swimming - 180,000 person-hours/yr
- boating - 790,000 person-hours/yr
- sport fishing (edible) yield - 227,000 lb/yr (103,000 kg/yr)
- commercial fishing yield - none assumed

TABLE E.1. Consumption and Usage Rates for the Maximally Exposed Individual

Pathway	Target	Rate
Fish	Infant	0 lb/yr (0 kg/yr)
	Child	15 lb/yr (6.9 kg/yr)
	Teenager	35 lb/yr (16 kg/yr)
	Adult	46 lb/yr (21 kg/yr)
Drinking Water	Infant	87 gal/yr (330 L/yr)
	Child	140 gal/yr (510 L/yr)
	Teenager	140 gal/yr (510 L/yr)
	Adult	190 gal/yr (710 L/yr)
Shoreline Use	Infant	0 h/yr
	Child	14 h/yr
	Teenager	67 h/yr
	Adult	12 h/yr
Boating	All	0 h/yr
Swimming	All	0 h/yr

TABLE E.2. Consumption Rates for Population Dose Calculations

Pathway	Target	Rate
Fish	Child	4.8 lb/yr (2.2 kg/yr)
	Teenager	12 lb/yr (5.2 kg/yr)
	Adult	15 lb/yr (6.9 kg/yr)
Drinking Water	Child	69 gal/yr (260 L/yr)
	Teenager	69 gal/yr (260 L/yr)
	Adult	98 gal/yr (370 L/yr)

In addition to the doses discussed above, doses to the population that consumes shellfish harvested from Chesapeake Bay were also calculated. An annual shellfish harvest of 72 million pounds (33 million kilograms) was assumed for 1989 and 108 million pounds (49 million kilograms) was assumed for 2009. Assuming an edible fraction of 1/2, the total shellfish consumption would be 36 million pounds (16 million kilograms) for 1989 and 54 million pounds (24 million kilograms) for 2009. The shellfish consumption rates for the average individual are listed in Table E.3, but the harvest was more than could be consumed by the population living within 50 miles (80 kilometers) of the power plant. Therefore, the population dose from shellfish consumption is applied to the entire population consuming Chesapeake Bay shellfish. A fraction of this dose (approximately 12 percent) is assumed to be received by the persons within the 50-mile (80-kilometer) radius that consume Chesapeake Bay shellfish. The remainder is received by persons outside the 50-mile (80-kilometer) radius.

The flow rate of the river was assumed to be 34,000 ft^3/sec (963 m^3/sec) for all except one of the calculations. The exception was the calculation of dose to the maximally exposed individual from the consumption of fish. For this calculation, a flow rate of 3150 ft^3/sec (89 m^3/sec) was used to correspond to the flow rate of the narrow channel near TMI. The fish caught by the maximally exposed individual were assumed to be caught from this channel. The transport time from the plant discharge point to the maximum individual or the population was neglected during the dose calculations.

TABLE E.3. Average Shellfish Consumption Rates

Target	Rate
Child	0.73 lb/yr (0.33 kg/yr)(a)
Teenager	1.6 lb/yr (0.75 kg/yr)(a)
Adult	2.2 lb/yr (1.0 kg/yr)(a)
Maximum Adult	97 lb/yr (44 kg/yr)(b)

(a) NRC 1977.
(b) Rupp, Miller, and Baes 1980.

E.2 AIRBORNE PATHWAYS AT THREE MILE ISLAND

Radiation doses to the public resulting from atmospheric releases from the TMI site during cleanup operations were calculated using the GASPAR II computer code (Strenge, Bander, and Soldat 1986). The GASPAR code generated 50-year dose commitments based on 1 year of inhalation or ingestion. For those cases where the release extends for more than 1 year, the 50-year dose commitment was calculated for 1 year of exposure and multiplied by the number of years over which the release occurs.

Doses were determined for the maximally exposed individual and for the 2.2 million people (age-group distribution: 71 percent, adults; 11 percent, teenagers; and 18 percent, children) assumed to be living within a 50-mile (80-kilometer) radius of the power plant in 1989. The population within a 50-mile (80-kilometer) radius during 2009 was assumed to be 3.2 million persons with the same age-group distribution. The pathways considered for both the maximally exposed individual and the population doses were inhalation, consumption of agricultural products, and external exposure. The dose (attributable to the TMI-2 cleanup) to the population outside the 50-mile (80-kilometer) radius was also calculated due to inhalation, external exposure, and consumption of agricultural products exported from within the 50-mile (80-kilometer) radius.

The parameters used as input for the calculations include the consumption rates for individual members of the population. The assumed consumption rates were as follows: 434 lb/yr (197 kg/yr) of vegetables, 35 gal/yr (131 L/yr) of milk, and 179 lb/yr (81 kg/yr) for meat. Total annual agricultural production for the 50-mile (80-kilometer) area surrounding the site is 1.2×10^8 pounds (5.32×10^7 kilograms) of vegetables, 1.4×10^8 gallons (5.27×10^8 liters) of milk, and 1.2×10^8 pounds (5.44×10^7 kilograms) of beef. Specific exposure pathway fractions are provided in Table E.4.

TABLE E.4. Airborne Exposure Pathway Fractions

Exposure Pathway	Fraction
Leafy vegetables from garden	0.5
Other edibles from garden	1.0
Fraction of time milk cows are on pasture	0.6
Fraction of time beef cattle are on pasture	1.0
Fraction of time milk goats are on pasture	1.0
Milk cow intake from pasture	1.0
Beef cattle intake from pasture	0.8
Milk goat intake from pasture	1.0

The 1989 population distributions were based on an internal NRC document by A. Sinisgalli, "1981 Residential Population Estimates 0-80 Kilometers for Nuclear Power Plants." The 2009 population distributions were obtained from a letter from F. R. Standerfer to the NRC, February 3, 1988, "Post-Defueling Monitored Storage Environmental Evaluation." $\bar{\chi}/Q'$ values were obtained from Appendix W of the PEIS (NRC 1981). The $\bar{\chi}/Q'$ values for the hypothetical maximally exposed individual [assumed to be a child located at the site boundary full time, 0.34 miles (0.55 kilometers) west of the site, who consumes goat milk from that site] was 3×10^{-5} sec/m^3 for the ground release. In addition, the absolute humidity for the site is 8.0 g/m^3. No credit for enhanced dilution from building wakes was taken.

Exposure parameters for the calculations that are not specified above are contained in the GASPAR II code.

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APPENDIX F

WASTE VOLUME ESTIMATES AND WASTE TRANSPORTATION IMPACTS

APPENDIX F

WASTE VOLUME ESTIMATES AND WASTE TRANSPORTATION IMPACTS

This appendix contains the methodologies, assumptions, and parameters used in the calculation of the waste volume resulting from post-defueling cleanup operations and the impacts of transporting this waste to a disposal site.

Section F.1 describes the waste volume estimates, waste classification, and radiation dose rate estimates that were used as input in calculating the transportation impacts. Section F.2 describes the calculation of the routine radiation exposure from transportation of the waste, the radiological accident risks, the nonradiological accident risks, and transportation costs. Waste volume estimates and waste transportation impacts were calculated for both the licensee's proposal of delayed cleanup and the alternative of immediate cleanup.

F.1 WASTE VOLUME, CLASSIFICATION, AND DOSE RATES

Cleanup activities, whether performed as delayed cleanup or as immediate cleanup, would generate waste. The principal waste-producing activities would include decontamination of the reactor coolant system, removal of contaminated portions of the reactor vessel head and control rod drive mechanisms, removal of the stairwell/elevator structure in the basement, removal of concrete surfaces (primarily in the basement), and removal of temporary shielding that has been placed in the reactor building. These activities would also generate secondary waste consisting of disposable protective clothing, tools, etc. The approximate volumes and classes of waste that would be generated are shown in Tables F.1 and F.2 for delayed cleanup and immediate cleanup, respectively.

Radioactive wastes are classified according to 10 CFR 61 criteria. Class A waste would contain designated radionuclides below the concentrations shown in Table F.3. It would consist mostly of compacted trash, slightly contaminated tools, contaminated equipment from upper elevations, and shielding that was placed in the building to facilitate defueling and cleanup operations. For the volume estimates in Tables F.1 and F.2, it was assumed that compactable material would be compacted. Class A waste would be shipped for offsite burial at a licensed low-level waste disposal facility. All Class A waste was assumed by the staff to be transported in commercially available, Class-A-approved, 217-cubic-foot (6.1-cubic-meter) casks that provided shielding equivalent to 2.73 inches (6.9 centimeters) of lead. Exposure rates 6.6 feet (2 meters) from such a cask, loaded with TMI-2 Class A waste, would average 0.04 mR/h, as calculated using the computer code SIMPLE (Reece et al. 1987). This exposure rate was used to assess the transportation impacts as discussed in Section F.2.

TABLE F.1. Waste from Delayed Cleanup

Waste Classification	Total Volume ft^3	Total Volume m^3
Preparation for PDMS		
Class A waste	100 to 200	2.8 to 5.7
During PDMS		
Class A dry radioactive waste	600 to 2000	17 to 57
Class B or C air filters	0 to 1250	0 to 41
Class A, B, or C residue from liquid waste treatment	100 to 400	2.8 to 11
Following PDMS		
Class A waste	3,400 to 7,800	98 to 220
Class C waste	19,000 to 33,000	540 to 930
Class A, B, or C waste	9,600 to 29,000	270 to 810
Greater than Class C	Some possible	Some possible

TABLE F.2. Waste from Immediate Cleanup

Waste Classification	Total Volume ft^3	Total Volume m^3
Class A waste	3,400 to 7,800	98 to 220
Class C waste	19,000 to 33,000	540 to 930
Class A, B, or C waste	9,600 to 29,000	270 to 810
Greater than Class C	Some possible	Some possible

Both Tables F.1 and F.2 list waste of unspecified class (waste that may be either Class A, B, or C). This waste would include insulation and equipment from the basement, some of the apparatus from the reactor vessel head, and other such equipment. Although the class of the waste cannot be predicted at this time, all of the waste would require measurement and classification before it was packaged and shipped to ensure that transportation

TABLE F.3. Class A Waste Limits for Isotopes Present at TMI-2[a]

Radionuclide	Maximum Concentration
Carbon-14	0.8 Ci/m^3
Cobalt-60	700 Ci/m^3
Nickel-63	3.5 Ci/m^3
Strontium-90	0.04 Ci/m^3
Cesium-137	1.0 Ci/m^3
Plutonium-241	350 nCi/g
Alpha[b]	10 nCi/g

(a) To determine the classification of wastes that contain a mixture of radionuclides, the concentration of each radionuclide is divided by the corresponding limit for that radionuclide (for the classification being determined). These fractional limits are summed and the sum must be less than 1. Radionuclides not listed are either not present at TMI-2 or may be present in any concentration in Class A waste.

(b) Alpha means alpha-emitting radionuclides with a half-life greater than 5 years. The following radionuclides discussed in Section 2.0 of this supplement fit this category: uranium-234, uranium-235, uranium-236, uranium-238, plutonium-238, plutonium-239, and americium-241. (A total of 18.3 curies of these radionuclides are assumed to be present at the completion of defueling.)

and disposal regulations were complied with. Some of the unspecified waste may be Class B waste; however, the quantity is expected to be small. It is assumed that most of the unspecified waste would be Class C waste. Class B waste is waste that exceeds the Class A limits for cobalt-60 or nickel-63 or that contains between 1 Ci/m^3 and 44 Ci/m^3 of cesium-137 or between 0.004 Ci/m^3 and 150 Ci/m^3 of strontium-90. The maximum concentration limits for Class C waste are shown in Table F.4. The rule in footnote (a) of Table F.3 for determining the waste classification for a mixture of radionuclides also applies to Classes B and C waste. For this analysis all of the unspecified waste was assumed to be Class C waste.

Waste that is clearly Class C would result from scabbling the basement walls and floors, from the removal of the enclosed stairwell/elevator shaft, from reactor coolant system decontamination, as well as waste generated during other cleanup activities. All of the Class C waste would require shipping in shielded transport casks. Commercially available 142-cubic-foot (4.0-cubic-meter) casks, which are approved for Class C waste and provide shielding equivalent to 4.5 inches (11.4 centimeters) of lead, were assumed to be used for the Class C and unspecified waste shipments.

The volume of wastes that would be generated (as summarized in Tables F.1 and F.2) was estimated on the basis of knowledge of the TMI-2 facility and assumptions that were made regarding the decontamination tasks to be performed. The volume of waste from scabbling the basement floor was estimated by modeling the reactor building basement floor as a circular area. For high volume estimates it was assumed that 2 inches (5 centimeters) of floor surface would be removed. For low volume estimates it was assumed that 1 inch (2.5 centimeters) of the surface would be removed. For both estimates it was assumed that the waste volume would be three times the volume of the poured concrete floor to the removed depth.

TABLE F.4. Class C Waste Limits(a)

Radionuclide	Maximum Concentration
Carbon-14	0.8 to 8.0 Ci/m^3
Strontium-90	150 to 7000 Ci/m^3
Cesium-137	44 to 4600 Ci/m^3
Plutonium-241	350 to 3500 nCi/g
Alpha(b)	10 to 100 nCi/g

(a) For mixtures of radionuclides footnote (a) in Table F.3 applies.
(b) Alpha has the same meaning as that in Table F.3.

Basement walls were assumed to consist of 400 to 500 lineal feet (122 to 153 meters) of wall that would be scabbled to a height of 8 feet (2.4 meters) on both sides. Although samples from the walls indicate that the initial 1/8th inch (0.32 centimeter) of concrete contains most of the activity, the staff assumed 1/4 inch (0.64 centimeter) would be removed by scabbling. The volume was assumed to triple to give between 400 and 500 cubic feet (11.3 to 14.2 cubic meters) of waste. Assuming that the waste produced from scabbling contains 8 percent of the cesium-137 (1680 curies) and 67 curies of strontium-90, it would be Class C waste. The radiation exposure rate 6.6 feet (2 meters) from a cask of this waste would be less than 0.01 mR/h as calculated using the computer code SIMPLE (Reece et al. 1987). This value was used to assess the transportation impacts as discussed in Section F.2.

The concrete block and other components from the stairwell and elevator shaft would also constitute a significant quantity of waste. The concrete block of this structure has an installed volume of approximately 1100 cubic feet (31 cubic meters). This is based on calculations used by Munson and Harty (1985), assuming that the entire 21 feet (6.4 meters) of stairwell would be removed. For the low estimates of waste volume it was assumed that the volume would double when removed. For the high estimates it was assumed that it would triple. Based on the curie estimates in Section 2, the concrete block waste would average not more than approximately 300 Ci/m^3 of cesium-137 and 14 Ci/m^3 of strontium-90 along with other radionuclides. This would constitute Class C waste. The radiation exposure rate 6.6 feet (2 meters) from a cask of this waste during shipment is expected to be not more than 0.004 mR/h. The radiation exposure rate was calculated using the computer code SIMPLE (Reece et al. 1987). This value was used to assess the transportation impacts as discussed in Section F.2. The metal doors, trolley, loading platform, and other components would contribute an additional volume. A volume of 21 feet (6.4 meters) by 3 feet (0.9 meters) by 1 foot (0.3 meters) was assumed for the stairs, and the volume of the doors, trolley, etc., was assumed to equal that of the stairs. Although some of this waste might be Class A or B, it was all assumed to be Class C waste.

Estimates of waste volume resulting from reactor coolant system decontamination were taken directly from the PEIS. Decontamination of the reactor coolant system was anticipated in the PEIS to generate approximately 9230 cubic feet (261 cubic meters) of radioactive waste. Waste from decontamination is assumed to contain the great majority of the fuel debris identified in Table 2.3. It will also contain some of the activation products. The following maximum concentrations were calculated:

- 73 nCi/g of mixed alpha (density of 1.1 assumed)
- 670 nCi/g of plutonium-241
- 1.4 Ci/m^3 of strontium-90
- 1.6 Ci/m^3 of cesium-137

This waste is expected to be Class C, although verification would be required before shipment. The radiation exposure rate 6.6 feet (2 meters) from a cask of this waste was calculated using the computer code SIMPLE (Reece et al.

1987) to average 1.5 mR/h. This value was used in assessing the transportation impacts as discussed in Section F.2.

The remainder of the Class C waste is expected to come from removal of insulation from the basement, the cleanup of the upper elevations, removal of control rod drives, and contaminated tools and equipment

For the purpose of this analysis the low-level waste disposal facility operated by U.S. Ecology near Richland, Washington, was assumed as the location for the disposal of all waste except that generated during final cleanup in the delayed cleanup alternative. As discussed in Section 2.3.4 of this supplement, regional low-level radioactive waste disposal facilities are expected to be available after 1992. Although no site has yet been designated, for this analysis a generic site 500 miles (800 kilometers) from TMI was assumed. The environmental impact of permanent waste storage in the disposal sites is considered to be outside the scope of this supplement and is the subject of a separate licensing action in connection with the site.

It is possible that some of the waste generated, especially from the basement, could exceed maximum Class C limits, in which case it could not be accepted by a licensed burial site. The licensee, however, has a unique arrangement with the U.S. Department of Energy that allows such wastes to be transferred to the DOE on a cost-reimbursement basis. It is under this agreement, known as the Memorandum of Understanding, that the fuel is being transferred to the DOE Idaho Falls Site.(a)

F.2 WASTE TRANSPORTATION IMPACTS

The transportation impacts estimated in this section include routine radiation doses, radiological accident risks, nonradiological accident risks, and transportation costs from the transport of cleanup wastes from TMI to licensed low-level waste (LLW) disposal facilities. The transportation impacts are examined for delayed cleanup and immediate cleanup. For delayed cleanup, wastes generated during preparations for PDMS and during PDMS are assumed to be transported by truck to the licensed LLW disposal facility operated by U.S. Ecology, Inc., near Richland, Washington. Wastes generated during cleanup following PDMS are assumed to be transported to a regional LLW disposal facility that does not currently exist. For the immediate cleanup alternative, all wastes are assumed to be transported by truck to the licensed LLW disposal facility near Richland, Washington.

The following sections discuss the routine radiological impacts, radiological accident impacts, nonradiological accident impacts, and transportation costs.

(a) Memorandum of Understanding Between the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy, Concerning the Removal and Disposition of Solid Nuclear Wastes from Cleanup of the Three Mile Island Unit 2 Nuclear Plant, March 15, 1982.

F.2.1 Routine Radiological Impacts

The routine radiation doses resulting from the transportation of waste during delayed and immediate cleanup were estimated using the RADTRAN III computer code (Madsen et al. 1983, 1986). A brief description of the RADTRAN III computer code and the bases and assumptions used in this analysis are provided and the results of the routine radiological impact calculations are discussed.

In routine (i.e., incident-free) transport, the package of radioactive material arrives at its destination without releasing its contents. Routine radiation doses include the direct external radiation dose emitted by the radioactive material package as the shipment passes by. Even though the shipping packages are provided with radiation shields, some radiation penetrates the package and exposes the nearby population to a low dose rate. After the shipment passes by, no further exposure occurs.

The population groups exposed to radiation include those exposed on an incidental basis and those exposed as a result of their occupation. Truck crew members are exposed as a result of their occupation. The general public is exposed on an incidental basis. The general public includes bystanders at truck stops, persons living or working along the route, and nearby travelers (moving in the same and opposite directions).

For the assessment of population dose, the packaging is assumed to be a point source of external, penetrating radiation. The point-source approximation is acceptable for distances between the receptor and the radiation source of more than two source-characteristic lengths. Source-characteristic length is defined as the largest physical dimension (length, diameter, etc.) of the radiation source. At shorter distances, the point-source approximation is conservative; that is, the calculated doses tend to be higher than those likely to occur. Derivations of the various equations used for different population groups and transport modes are discussed in detail by Taylor and Daniel (1982) and Madsen et al. (1986). Some of the input data used in this analysis are listed in Table F.5. These data are RADTRAN III default values, except where indicated.

The transportation impacts are influenced by the population densities of the regions across which the shipments must travel. The percent of time that travel occurs in each of three population zones (i.e., rural, suburban, and urban) was taken from Cashwell et al. (1986). The values for shipments to the LLW site in Richland, Washington, are 78 percent in rural areas, 21 percent in suburban areas, and 1 percent in urban areas. For shipments to a regional LLW disposal facility, the values used are 63-percent rural, 36-percent suburban, and 1-percent urban(a) and are believed to be representative of shipments from TMI to most locations within 500 miles (800 kilometers) of the site.

(a) A 1-percent urban travel fraction is assumed even though Cashwell et al. (1986) indicate no travel in urban areas.

TABLE F.5. Input Data for Analysis of Routine Transportation Impacts(a)

Parameter	Value
Number in Truck Crew	2
Distance from Source to Crew, meters	5
Population Densities, persons/km^2	
High-population zone (urban)	3861
Medium-population zone (suburban)	719
Low-population zone (rural)	6
Average Speed of Truck, km/h	
High-population zone (urban)	24
Medium-population zone (suburban)	40
Low-population zone (rural)	88
Traffic Count, one-way vehicles/h	
High-population zone (urban)	2800
Medium-population zone (suburban)	780
Low-population zone (rural)	470
Average Exposure Distance While Stopped, meters	20
Stop Time, h/km	0.011
Number of Persons Exposed While Vehicle Stopped	50
One-Way Shipping Distance, kilometers	
To Hanford LLW disposal facility	4314(b)
To regional LLW disposal facility	805(c)

(a) Values are taken from Madsen et al. (1983) except where otherwise indicated.
(b) Source: Cashwell et al. 1986.
(c) Assumed value used in this study.

The calculated, routine, 50-year committed radiation doses are shown in Table F.6. The routine doses are given in units of person-rem accumulated during the entire shipping program. Doses to the truck crews, persons who live or work in the vicinity of the highway (off-highway), persons sharing the highway with the shipments (on-highway), and bystanders at truck stops are shown separately for each waste type. As shown, the truck crews will receive the largest portion of the routine dose, followed by persons at truck stops. On-highway and off-highway doses are small relative to truck crew and stop doses.

As shown in Table F.6, delayed cleanup results in the lowest routine dose from transportation, approximately 5 to 6 person-rem depending upon the waste volume. The routine dose estimates for shipping wastes from immediate cleanup are estimated to be between about 20 and 25 person-rem. This is because of a shorter travel distance between TMI and the regional repository during cleanup following PDMS than between TMI and Richland during immediate cleanup.

F.2.2 Radiological Accident Risks

The accident analysis considers the potential release of radioactive material from the waste package and its contents. The RADTRAN III computer code was also used to calculate the transportation accident risks. For this study, risk is defined as the frequency of accidents involving radioactive material multiplied by the consequences of an accident. The consequences can be expressed in terms of the radiation dose resulting from a release of radionuclides from the packaging or the exposure of persons to radiation that could result from damaged package shielding.

The frequency of an accident is expressed in terms of the number of accidents per unit distance. The response of the shipping container, and thus the probability of a release or loss of shielding, is related to the severity of the accident. Accidents with severities that exceed the design standards for shipping packages (see 10 CFR 71 and 49 CFR 173, Subpart I) could potentially occur, but the probability is extremely small. Thus, there is a slight possibility that an accident accompanied by a release of radioactive material or reduction of shielding could occur. The accident rates used in this study, which are RADTRAN III default parameters, are given for three population density zones: rural (1.4×10^{-7} accidents/km), suburban (2.7×10^{-6} accidents/km), and urban (1.6×10^{-5} accidents/km). As expected, accident rates in urban areas are significantly higher (i.e., about 100 times higher) than accident rates in rural areas.

RADTRAN III uses four quantities (the release fraction, the severity fraction, the aerosol fraction, and the respirable fraction) to describe a release of radioactive material. These quantities are dependent upon the severity of the accident. The release fraction is the amount of radioactive material of all sizes that could escape from the package in an accident (given as the fraction of the total contents of the package). The release fraction varies with the severity category. Eight severity categories are used in RADTRAN III. Associated with each severity category is a severity

TABLE F.6. Routine Radiological Doses for Transporting TMI Cleanup Wastes to Offsite Disposal Facilities

Alternative	Exposed Population	Doses by Type of Waste, person-rem: Class A Waste	Class C Waste (Reactor Coolant System)	Other Class C Waste	Total(a)
Delayed Cleanup - High waste volume	Truck Crew	1	2	0.7	
	Off-highway(b)	0.04	0.1	0.03	
	On-highway(c)	0.04	0.1	0.03	
	Stops	0.5	1	0.3	
	Total(a)	2	4	1	6
Delayed Cleanup - Low waste volume	Truck Crew	0.4	Same as	0.3	
	Off-highway(b)	0.02	High Waste	0.01	
	On-highway(c)	0.02	Volume	0.01	
	Stops	0.2	Case(d)	0.1	
	Total(a)	0.7	4	0.4	5
Immediate Cleanup - High waste volume	Truck Crew	2	11	3	
	Off-highway(b)	0.06	0.3	0.09	
	On-highway(c)	0.07	0.4	0.1	
	Stops	1	5	1	
	Total(a)	3	17	5	25
Immediate Cleanup - Low waste volume	Truck Crew	1	Same as	1	
	Off-highway(b)	0.03	High Waste	0.03	
	On-highway(c)	0.03	Volume	0.03	
	Stops	0.4	Case(d)	0.5	
	Total(a)	1	17	2	20

(a) The totals may not be exact because of rounding.
(b) "Off-highway" refers to exposures to persons residing or working along a highway.
(c) "On-highway" refers to persons sharing the highway with the waste shipment; includes persons traveling in the same and opposite directions.
(d) Only one waste volume estimate was provided for reactor coolant system decontamination activities.

fraction, that is, the fraction of accidents that occur that would be representative of the accident conditions described by each severity category. The overall accident frequency for each severity category can be obtained by multiplying the severity fraction by the overall accident rate. The aerosol fraction is defined as the fraction of material released that can be entrained in an aerosol (cloud of radioactive material). The respirable fraction accounts for the fraction of aerosolized material that is also respirable (i.e., can be inhaled into the lungs). The release fraction and the severity fractions are presented in Table F.7. The values used for the aerosol fraction and the respirable fraction for each waste type considered in this study are presented in Table F.8.

RADTRAN III evaluates the radiation dose resulting from four pathways: external exposure to radiation from a passing cloud of radioactive material, external exposure from radioactive materials deposited on the ground, inhalation (exposure to radiation from breathing in radioactive materials), and ingestion (exposure from food that has been contaminated as a result of an accidental release of radionuclides and then ingested). The accident dose pathways are illustrated in Figure F.1. RADTRAN III assumes that radioactive materials released from a package in an accident are dispersed according to a standard Gaussian diffusion model. The model predicts downwind airborne radionuclide concentrations and the amount of material deposited on the ground. Radiation doses to human organs are then determined using the

TABLE F.7. Release Fraction and Severity Fractions Used in RADTRAN III Accident Analysis

Severity Category	Release Fraction(a)	Severity Fraction for Truck Shipments		
		Rural	Suburban	Urban
1	0.0	4.6×10^{-1}	4.4×10^{-1}	5.8×10^{-1}
2	0.0	3.0×10^{-1}	2.9×10^{-1}	3.8×10^{-1}
3	1×10^{-6}	1.8×10^{-1}	2.2×10^{-1}	2.8×10^{-2}
4	1×10^{-5}	4.0×10^{-2}	5.1×10^{-2}	6.4×10^{-3}
5	1×10^{-4}	1.2×10^{-2}	6.6×10^{-3}	7.4×10^{-4}
6	1×10^{-3}	6.5×10^{-3}	1.7×10^{-3}	1.5×10^{-4}
7	1×10^{-2}	5.7×10^{-4}	6.7×10^{-5}	1.1×10^{-5}
8	1×10^{-1}	1.1×10^{-4}	5.9×10^{-6}	9.9×10^{-7}

(a) Given as the fraction of the cask contents that are released as a result of an accident. Source: JIO 1986.

TABLE F.8. Aerosol and Respirable Fractions Used in RADTRAN III Accident Analysis

Parameter	Type of Waste		
	Class A Waste	Class C, Reactor Coolant System Waste	Other Class C Waste
Aerosol Fraction	0.05	0.1	0.05
Respirable Fraction	0.05	0.05	0.05

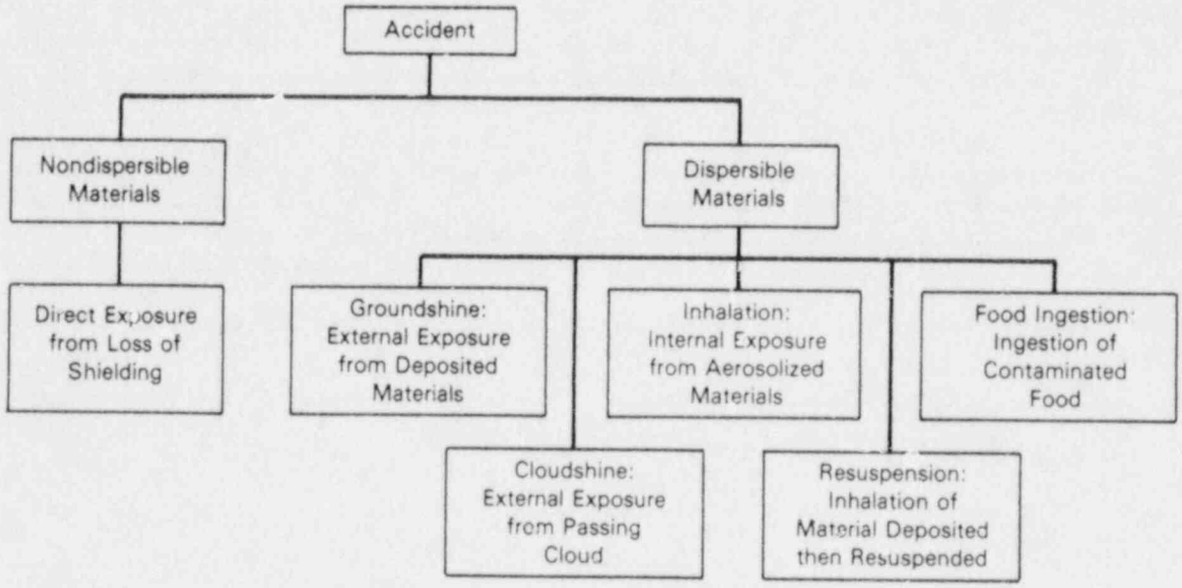

FIGURE F.1. Accident Dose Pathways Included in RADTRAN III

calculated airborne and ground-deposited radionuclide concentrations and standard dosimetric conversion factors. A 50-year dose commitment from radioactive materials deposited on the ground for a single year is calculated for the public. The model assumes that after 1 year the contaminated area will be cleaned up to acceptable residual levels, if needed, or, if the contamination is too great, it is assumed that the area will be fenced off and access prohibited. Radiation doses to emergency response and cleanup personnel are not included. Doses to the general population from ingestion of radioactive material are estimated with the use of radionuclide transfer fractions which relate the amount of radioactive material ingested to the amount deposited on the ground after a potential accident (Ostmeyer 1985).

Calculated transportation accident risks associated with TMI cleanup activities are presented in Table F.9. The results are given in units of population dose (total 50-year committed dose) for the entire shipping program. This can be viewed as the sum of frequencies of a particular accident times the consequence of that accident in person-rem.

As shown in Table F.9, delayed cleanup is expected to result in the lowest transportation risks. This is because of a significant reduction in the shipping distance if cleanup is delayed until regional LLW disposal facilities are available. The largest doses result from shipment of wastes generated during reactor coolant system decontamination activities because the decontamination wastes include higher levels of cobalt-60 than the other waste types and also include some actinides such as plutonium-238, plutonium-239, and americium-241.

TABLE F.9. 50-Year Radiological Dose Commitment from Accidents During Transport of TMI-2 Cleanup Wastes to Offsite Disposal Facilities

		Type of Waste, person-rem			
Alternative	Exposed Population	Class A Waste	Class C, Reactor Coolant System Waste	Other Class C Waste	Total,(a) person-rem
Delayed - High waste volume	Public	<<5 x 10-5	6 x 10-4	1 x 10-4	7 x 10-4
Delayed - Low waste volume	Public	<<5 x 10-5	6 x 10-4(b)	6 x 10-5	7 x 10-4
Immediate - High waste volume	Public	<<5 x 10-5	2 x 10-3	6 x 10-4	2 x 10-3
Immediate - Low waste volume	Public	<<5 x 10-5	2 x 10-3(b)	2 x 10-4	2 x 10-3

(a) The totals may not be exact because of rounding.
(b) Only one waste volume estimate was provided for reactor coolant system decontamination activities.

F.2.3 Nonradiological Accident Impacts

Nonradiological accident risks consist of injuries and fatalities that may result from traffic accidents involving the shipments of TMI-2 cleanup wastes. These risks are in no way related to the radioactive nature of the waste materials being transported. The number of estimated injuries and fatalities would be the same even if the cargo were not radioactive materials. This section uses standard unit risk factors to estimate the nonradiological risks of transporting TMI cleanup wastes to offsite disposal facilities.

The potential for accidents involving shipments of TMI cleanup wastes is assumed to be comparable to that of general truck transport in the United States. Cashwell et al. (1986) used statistics compiled by the U.S. Department of Transportation (DOT 1985) to develop nonradiological risk factors. These risk factors, in units of fatalities/kilometer and injuries/kilometer of travel, are multiplied by the total distance traveled for each type of waste shipment to calculate the expected number of nonradiological injuries and fatalities due to transportation of TMI cleanup wastes. These risk factors are shown in Table F.10. Accident fatality and injury data are available for both transport workers (truck crews) and the general public during travel in three population zones: rural, suburban, and urban (Cashwell et al. 1986). Therefore, the total number of fatalities (or injuries) over the entire shipping program is the sum of the products of the vehicle miles (kilometers) and the fatality or injury rates in each zone.

The total number of traffic accidents involving these shipments was also estimated using a similar approach. The number of accidents was estimated using the accident rates in rural, suburban, and urban areas that were given in Section F.2.1. These rates were multiplied by the total travel distances in these areas.

TABLE F.10. Truck Transportation Accident Risk Factors for All Waste Types

Population Zone	Affected Group	Risk Factors	
		Fatalities/kilometer	Injuries/kilometer
Rural	Truck crew	1.5×10^{-8}	2.8×10^{-8}
	Public	5.3×10^{-8}	8.0×10^{-7}
Suburban	Truck crew	3.7×10^{-9}	1.3×10^{-8}
	Public	1.3×10^{-8}	3.8×10^{-7}
Urban	Truck crew	2.1×10^{-9}	1.3×10^{-8}
	Public	7.5×10^{-9}	3.7×10^{-7}

Source: Cashwell et al. 1986.

The estimated number of traffic accidents and the total estimated fatalities and injuries are shown in Table F.11 for delayed cleanup and immediate cleanup. The total number of nonradiological fatalities was estimated to be less than 1.0 for both delayed and immediate cleanup. The number of injuries occurring as a result of an accident during transport of wastes during delayed cleanup was estimated to be between about 0.3 and 0.6 injuries, depending upon the estimated waste volume. For wastes generated during immediate cleanup, the number of injuries was estimated to be between 1 and 3. The alternative that is anticipated to involve the highest nonradiological impacts is immediate cleanup. This is because of the much longer shipping distances involved for the immediate cleanup alternative (i.e., shipment to Richland, Washington, during immediate cleanup versus a much closer regional LLW disposal facility during delayed cleanup following storage).

TABLE F.11. Estimated Number of Traffic Accidents, Fatalities, and Injuries for Each Alternative

Alternative	Waste Volume Estimate	Total Number of Accidents	Total Fatalities		Total Injuries	
			Truck Crew	Public	Truck Crew	Public
Delayed Cleanup	High	1	0.01	0.04	0.02	0.6
	Low	0.5	0.004	0.02	0.009	0.3
Immediate Cleanup	High	3	0.05	0.2	0.1	3
	Low	1	0.02	0.08	0.05	1

F.2.4 Transportation Costs

Transportation costs are estimated assuming that all transportation services would be provided by commercial companies. It is assumed that a sufficient supply of the shipping containers would be available when needed. This means that capital costs for construction of additional shipping containers would not be necessary. Thus, transportation costs consist of shipping charges and shipping container leasing fees.

Shipping costs are the costs charged by commercial carrier companies for moving waste shipments from TMI to a destination facility and returning the empty container to TMI. Data used to determine shipping costs were taken from McNair et al. (1986). These data are based on published tariffs and include such items as freight rates and detention of drivers and vehicles while shipping containers are being loaded/unloaded. Because of the deregulation of the transportation industry, actual shipping costs cannot be determined until a contract is negotiated between the shippers and carrier companies.

Because shipping containers are owned by commercial companies, a lease is required for the shipper to use the shipping containers. Based on telephone conversations with owners of shipping containers, it is assumed for

this study that the lease fee for the representative shipping containers amounts to $200/day. This rate is multiplied by the number of days the shipping containers are estimated to be used. The number of days was estimated by dividing the shipping distances through rural, suburban, and urban population zones by the average speed in these zones (see Table F.5), summing over all three zones, and then adding time spent at truck stops. Stop time was estimated using a factor of 0.011 hour of stop time per kilometer of travel (see Table F.5). Two days were added to each trip to account for loading (1 day) and unloading (1 day) of shipping containers. Assuming that a two-person driving team travels round-the-clock, shipments from TMI to Hanford and back to TMI would take approximately 11 days. Shipments from TMI to a regional LLW disposal facility and back again are estimated to require about 4 days.

Results of the transportation cost calculations for shipment of TMI cleanup wastes to disposal facilities are shown in Table F.12. The total number of shipments for each waste type and total costs are shown in the table. Although total costs are based on the best available information and are believed to be representative approximations, they are intended for comparison purposes only.

TABLE F.12. Total Transportation Costs

Alternative	Waste Volume Estimate	Waste Type, Number of Shipments: Class A Waste	Class C, Reactor Coolant System Waste	Other Class C Waste	Total Cost, $ Millions(a)
Delayed	High(b)	47	65	384	1.3
	Low(c)	20	65	138	0.5
Immediate	High	36	65	372	3.2
	Low	16	65	137	1.5

(a) Costs are given in 1987 dollars.
(b) Includes 11 shipments of Class A waste and 12 shipments of Class C waste to Richland during pre-PDMS and PDMS phases.
(c) Includes 4 shipments of Class A waste and 1 shipment of Class C waste to Richland during pre-PDMS and PDMS phases.

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APPENDIX G

CALCULATION OF SOCIOECONOMIC IMPACTS

APPENDIX G

CALCULATION OF SOCIOECONOMIC IMPACTS

The direct socioeconomic impacts of both delayed and immediate cleanup were evaluated. The socioeconomic impact of the cleanup operations depends on the size of the work force that would perform the work. Employment at TMI is considered to be "export-base" employment because it involves the sale of products or services outside the economy. The useful rule of thumb for local Pennsylvania economies is that export-base employment sustains 0.5 or more local offsite support-sector jobs for every 1 direct "export-base" job. Thus, the implicit offsite employment multiplier is about 0.5 (0.5 offsite jobs for each onsite job) and the total employment multiplier is about 1.5 (1.5 total jobs, onsite and offsite, for each job onsite).(a) Similarly, a reasonable local offsite income multiplier is about 1.0 and the total local income multiplier is about 2.0.

For delayed cleanup, the licensee estimates that the level of direct employment for the PDMS program would be about 100 to 125 personnel during the transition year following the completion of current defueling activities and about 70 to 75 personnel thereafter until decommissioning or refurbishment begins at an indefinite future date (currently, approximately 1150 personnel are involved in defueling and decommissioning).(b) According to the same source, transition-year direct payrolls would be about $6.2 million, and subsequent annual payrolls for the monitoring and maintenance work force would be about $3.8 million. Assuming total employment and income impact multipliers of 1.5 and 2.0, respectively, the total local economic impact in the transition year would be about 150 to 200 jobs and $12 million to $13 million in local income, and from 100 to 120 jobs and about $7 million to $8 million in annual local income thereafter. These amounts are extremely small in relation to the local economy and cannot be considered significant socioeconomic impacts.

In response to NRC staff inquiries, the licensee has indicated that although detailed planning for immediate cleanup has not been accomplished, it can be assumed for the purpose of evaluating the socioeconomic impact of delayed cleanup that the level of employment during immediate cleanup would not be greater than the current level of approximately 1150 personnel involved in defueling and decontamination. In all likelihood, the required

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(a) The total employment multiplier can be somewhat higher than 1.5 (0.5 offsite plus 1.0 direct jobs) if the export sector jobs are highly paid. (Personal contact with Stan Duobinis, Econometrics Department, Wharton School of Finance and Commerce, University of Pennsylvania, Philadelphia, Pennsylvania, September 22, 1987.)

(b) Letter from F. R. Standerfer, Director, TMI-2, GPU Nuclear Corporation to W. D. Travers, Director, TMI-2 Cleanup Project Directorate, NRC, November 5, 1987. Subject: Post-Defueling Monitored Storage Environmental Evaluation, NRC Comment Response.

work force would be much smaller.(a) Based on the potential occupational exposure levels and activity requirements for immediate cleanup described in Section 3.3, it is assumed for the purpose of the socioeconomic analysis that cleanup would require approximately 3 years with a maximum of 1150 personnel. The licensee estimates that the payroll cost of the immediate cleanup scenario would be about $57.5 million per year at the employment level of 1150 workers. According to data supplied by the licensee, approximately 70 percent of the current TMI work force resides in the Harrisburg-Lebanon-Carlisle labor market area, which consists of Cumberland, Dauphin, Lebanon, and Perry Counties. (About 50 percent reside in Dauphin County.) It can be expected that this labor market area would benefit the most from the continued employment of workers, followed by Lancaster County (with 25 percent of the TMI work force) and York labor market area (consisting of Adams and York Counties and having 5 percent of the TMI labor force).

Assuming that the multipliers discussed above apply and that offsite employment and income impacts are distributed geographically in the areas where the TMI workers reside, total employment and income in nearby local economies temporarily would be higher than they otherwise would be by the amounts shown in Table G.1. As can also be seen from the table, the impact in each labor market area is significant but relatively small (less than 0.5 percent) in comparison to the total local economy. Table G.1 shows maximum annual impacts. If the cleanup work force were significantly smaller, then the impacts would also be significantly smaller.

The only socioeconomic impact associated with the delayed cleanup alternative is the early transition from the current level of project employment of about 1150 to the much lower levels discussed above. Immediate cleanup would temporarily sustain the portion of local jobs and income dependent upon current defueling activities at the reactor. Compared to delayed cleanup, the level of local employment during immediate cleanup could be as much as 1500 more jobs for the 3-year period, and as much as $100 million more to the local area income during the same period. Although the differences between the alternatives are significant, the employment difference is temporary and amounts to less than 0.2 percent of the local baseline employment in 1987.

(a) Letter from F. R. Standerfer, Director, TMI-2, GPU Nuclear Corporation, to W. D. Travers, Director, TMI-2 Cleanup Project Directorate, NRC, November 5, 1987. Subject: Post-Defueling Monitored Storage Environmental Evaluation NRC Comment Response.

TABLE G.1. Annual Local Economic Impact of TMI-2 Employment for Immediate Cleanup(a)

Labor Market Area	Direct Jobs	Total Local Jobs	Direct Payrolls, $ million 1987	Local Income, $ million 1987	Total Local Jobs Impact as a Percent of Current Employment(b)
Harrisburg-Lebanon-Carlisle(c)	1150	1550	$57.5	$ 80	0.5
Lancaster(d)	0	145	0	$ 28	0.1
York(e)	0	30	0	$ 6	0.01
Total	1150	1725	$57.5	$114	0.2

(a) Jobs and direct payroll are reported on a place-of-work basis; local income is reported on a place-of-residence basis. The 0.5 offsite jobs for each direct job and $1 of offsite income per dollar of onsite income are assumed to be generated at the areas where TMI workers reside.

(b) Based on second-quarter 1987 employment as reported in Pennsylvania State University College of Business Administration, Pennsylvania Business Survey, August, 1987.

(c) Includes Cumberland, Dauphin, Lebanon, and Perry Counties. Residence of 70 percent of TMI workers.

(d) Lancaster County only. Residence of 25 percent of TMI workers.

(e) Includes Adams and York Counties. Residence of 5 percent of TMI workers.

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In accordance with the National Environmental Policy Act, the Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Waste from March 28, 1979 Accident Three Mile Island Nuclear Station, Unit 2 (PEIS) has been supplemented. This draft supplement addresses potential environmental impacts associated with the licensee's (GPU Nuclear's) proposal to place the TMI-2 facility in a post-defueling monitored storage mode followed by the completion of cleanup. The NRC staff has concluded, based on this evaluation, that the licensee's proposed plan and the NRC staff-identified alternatives for completion of cleanup are within applicable regulatory limits and could be implemented without significant environmental impact. No alternative was found to be clearly preferable from an environmental impact perspective. The staff concluded that the benefits of cleanup action outweigh the small associated impacts.

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