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DOE/EIS-0275

FINAL ENVIRONMENTAL IMPACT STATEMENT

S1C Prototype Reactor Plant Disposal

Volume 1 of 2

November 1996

Prepared by the
U. S. Department of Energy
Office of Naval Reactors

Printed on recycled paper

United States Government — Department of Energy

memorandum

DATE: November 1, 1996 R#96-11519

REPLY TO ATTN OF: NE-60

SUBJECT: FINAL ENVIRONMENTAL IMPACT STATEMENT ON THE DISPOSAL OF THE S1C PROTOTYPE REACTOR PLANT; REQUEST FOR CONCURRENCE

TO: W. Dennison, GC-51

Pursuant to DOE Order 451.1 and the DOE NEPA regulations, this memorandum requests concurrence with issuance of the attached Final Environmental Impact Statement. Concurrence also is requested with the attached Notice of Availability.

Thank you for your assistance. If you have any questions, please call me at 703-603-2167 or Jeff Steele at 703-603-5103.

Richard A. Guida

Richard A. Guida
Associate Director
for Regulatory Affairs
Office of Naval Reactors

Attachments

Billing Code 6450-01-P

DEPARTMENT OF ENERGY

NOTICE OF AVAILABILITY OF THE FINAL ENVIRONMENTAL IMPACT STATEMENT ON THE DISPOSAL OF THE S1C PROTOTYPE REACTOR PLANT

AGENCY: Department of Energy

ACTION: Notice of availability

SUMMARY: The Department of Energy (DOE) Office of Naval Reactors (Naval Reactors) has completed and filed with the U.S. Environmental Protection Agency the Final Environmental Impact Statement on the Disposal of the S1C Prototype Reactor Plant. The Final Environmental Impact Statement was prepared in accordance with the National Environmental Policy Act (NEPA) of 1969; Council on Environmental Quality regulations implementing NEPA (40 CFR Parts 1500-1508); and DOE NEPA Implementing Procedures (10 CFR Part 1021). The Final Environmental Impact Statement and its supporting references will be available to the public at the Windsor, Connecticut Public Library. The Final Environmental Impact Statement is also available by mail upon request.

SUPPLEMENTARY INFORMATION:

Background

The S1C Prototype reactor plant is located on the 10.8 acre Windsor Site in Windsor, Connecticut, approximately 5 miles north of Hartford. The S1C Prototype reactor plant first started operation in 1959 and served for more than 30 years as both a facility for testing reactor plant components and equipment and for training Naval personnel. As a result of the end of the Cold War and the downsizing of the Navy, the S1C Prototype reactor plant was shut down in 1993. Since then, the S1C Prototype reactor plant has been defueled, drained, and placed in a stable protective storage condition.

Alternatives Considered

1. Prompt Dismantlement

This alternative would involve the prompt dismantlement of the reactor plant. All structures would be removed from the Windsor Site, and the Windsor Site would be released for unrestricted use. To the extent practicable, the resulting low-level radioactive metals would be recycled at existing commercial facilities that recycle radioactive metals. The remaining low-level radioactive waste would be disposed of at the DOE Savannah River Site in South Carolina. The Savannah River Site currently receives low-level radioactive waste from Naval Reactors sites in the eastern United States. Both the volume and radioactive content of the S1C Prototype reactor plant low-level waste fall within the

projections of Naval Reactor waste provided to the Savannah River Site which in turn are included in the Savannah River Site Waste Management Final Environmental Impact Statement dated July 1995.

2. Deferred Dismantlement

This alternative would involve keeping the defueled S1C Prototype reactor plant in protective storage for 30 years before dismantling it. Deferring dismantlement for 30 years would allow nearly all of the cobalt-60 radioactivity to decay away. Nearly all of the gamma radiation within the reactor plant comes from cobalt-60.

3. No Action

This alternative would involve keeping the defueled S1C Prototype reactor plant in protective storage indefinitely. Since there is some residual radioactivity with very long half lives such as nickel-59 in the defueled reactor plant, this alternative would leave this radioactivity at the Windsor Site indefinitely.

4. Other Alternatives Considered

These alternatives include permanent on-site disposal. Such on-site disposal could involve building an entombment structure over the S1C Prototype reactor plant or developing a below ground disposal area at the Windsor Site. Another alternative would be to remove the S1C Prototype reactor plant as a single large reactor compartment package for offsite disposal. Each of these alternatives was considered but eliminated from detailed analysis.

Public Comments on Draft Environmental Impact Statement

Naval Reactors held a public hearing on the Draft Environmental Impact Statement in Windsor, Connecticut. Comments from 28 individuals and agencies were received in either oral or written statements at the hearing or comment letters. Nearly all of the commenters expressed a preference for the prompt dismantlement alternative. Most comments resulted in either no changes or minor clarifications in the final environmental impact statement. The comments which resulted in the more significant changes are discussed briefly below. All of comments and the Naval Reactors responses are included in an appendix to the Final Environmental Impact Statement.

Some comments requested additional detail on the process, surveys, and criteria identified in the draft environmental impact statement for unrestricted release of the site under either the prompt dismantlement or deferred dismantlement alternatives. In response to these comments, appendices are included in the final environmental impact statement which provide additional details on these matters.

Several comments questioned whether the cost and volume of radioactive waste generated for each alternative included site remediation as well as reactor dismantlement. The draft environmental impact statement discussed the overall site remediation impacts as part of the cumulative impacts, however the quantitative cost and waste volume discussions focused on reactor plant dismantlement, which is where essentially all of the radioactivity is located. The final environmental impact statement includes impacts from all efforts anticipated from the time of the record of decision until completion of each alternative (in the cases of prompt and deferred dismantlement, this is through transfer of the property to another owner). The most significant changes are cost, volume (but not number of shipments) of radioactive waste, and the volume and number of shipments of non-radioactive, non-hazardous solid waste. These changes did not alter the relative ranking of the alternatives on these measures nor did they significantly change the estimated impact of the alternatives on the environment or the health and safety of the workers or the public.

Preferred Alternative

Since the occupational radiation exposure risk to the workers would be small, since prompt dismantlement would result in unrestricted release of the Windsor Site at the earliest time, and the impacts associated with prompt dismantlement have a higher degree of certainty, Naval Reactors has identified prompt dismantlement as the preferred alternative.

Availability of Copies of the Final Environmental Impact Statement

The Final Environmental Impact Statement has been distributed to interested Federal, State, and local agencies, and to individuals who have expressed interest. Copies of the Final Environmental Impact Statement and its supporting references are available for inspection at the Windsor Public Library at 323 Broad Street, Windsor, CT 06095. Requests for copies of the Final Environmental Impact Statement should be directed to Mr. C. G. Overton, Chief Windsor Field Office, Office of Naval Reactors, U.S. Department of Energy, P.O. Box 393, Windsor, CT 06095; telephone (860) 687-5610.

Issued at Arlington, VA this __th day of November 1996.

F. L. Bowman
Admiral, U.S. Navy
Director, Naval Nuclear Propulsion Program

COVER SHEET

PROPOSED ACTION: Determine a disposal strategy for the defueled S1C Prototype reactor plant.

TYPE OF STATEMENT: Final Environmental Impact Statement

RESPONSIBLE AGENCY: U.S. Department of Energy, Office of Naval Reactors

FOR FURTHER INFORMATION: For further information on this Final Environmental Impact Statement, call or contact:

Mr. Christopher G. Overton, Chief
Windsor Field Office, Office of Naval Reactors
U.S. Department of Energy
PO Box 393
Windsor, CT 06095
Telephone (860) 687-5610

ABSTRACT: This Final Environmental Impact Statement evaluates in detail three alternatives for the disposal of the S1C Prototype reactor plant. These alternatives include: prompt dismantlement and disposal of the entire S1C Prototype reactor plant; deferred dismantlement, which allows for decay of some radioactivity prior to dismantlement; and "no action," which means continuing surveillance and monitoring for an indefinite period of time.
The evaluations conclude that the environmental and socioeconomic impacts for all of the disposal alternatives would be small.

Naval Reactors received written comments on the Draft Environmental Impact Statement during a 45-day public comment period lasting from July 5, 1996 to August 19, 1996. Oral comments were received during a public hearing held on August 7, 1996. This Final Environmental Impact Statement includes copies of all written and oral comments that Naval Reactors received on the Draft Environmental Impact Statement. All comments were taken into consideration during preparation of this Final Environmental Impact Statement.

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TABLE OF CONTENTS

VOLUME 1 of 2

LIST OF FIGURES

LIST OF TABLES

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SUMMARY

The U.S. Department of Energy Office of Naval Reactors (Naval Reactors) is currently evaluating alternatives for disposal of the S1C Prototype reactor plant, located at the Knolls Atomic Power Laboratory Windsor Site in Windsor, Connecticut (Windsor Site). A key element of Naval Reactors' decision making is a thorough understanding of the environmental impacts associated with each alternative. The National Environmental Policy Act requires Federal agencies to analyze the potential environmental impacts (both positive and negative) of their proposed actions to assist them in making informed decisions. In following this process, Naval Reactors prepared a Draft Environmental Impact Statement to assess various alternatives and to provide necessary background, data and analysis to help decision makers and the public understand the potential environmental impacts of each alternative. Following consideration of public comments, Naval Reactors prepared this Final Environmental Impact Statement. The Naval Reactors decision will be presented in a Record of Decision to be issued thirty days after publication of the Final Environmental Impact Statement.

National Environmental Policy Act: A Federal law passed in 1969, which requires all Federal agencies to consider in their decision making processes potential environmental effects before implementing any major action, and established the Council on Environmental Quality within the Office of the President.

Alternatives: The range of reasonable options considered in evaluating and selecting an approach to meet the need for agency action.

Environmental Impact Statement: A detailed environmental analysis for a proposed action that could significantly affect the environment. A tool for decision making, it describes the positive and negative environmental effects of the alternatives.

Record of Decision: A concise public record of the agency's decision, which discusses the alternative selected. The discussion will include whether all practicable means to avoid or minimize environmental harm from the selected alternative were adopted (and if not, why they were not).

The S1C Prototype reactor plant was permanently shut down in March 1993, reflecting the end of the Cold War and projected downsizing of the U.S. Naval fleet. All spent nuclear fuel was removed from the S1C Prototype reactor and has been shipped off-site. Management of spent nuclear fuel has been addressed in a separate Department of Energy evaluation, Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement, (Reference 1-1).

The S1C Prototype reactor plant is located within a reactor compartment. The reactor compartment is shielded and serves as a containment structure. All S1C Prototype reactor plant systems have been drained, deenergized and placed in a safe, stable condition. However, the S1C Prototype reactor plant systems still contain radioactive materials such as activated metals and corrosion products.

The disposal alternatives examined in detail include:

(1) **Prompt Dismantlement (Preferred Alternative)** - The S1C Prototype reactor plant would be promptly dismantled and materials would be disposed of or recycled. Low-level radioactive waste would be shipped to the Department of Energy Savannah River Site for disposal. All structures would be removed from the Windsor Site and the Windsor Site would be released for unrestricted use. Under this alternative, the Windsor Site could be made available for other uses as early as possible, currently estimated for the year 2001.

(2) **Deferred Dismantlement** - The S1C Prototype reactor plant would be left in a drained, deenergized, stable condition and monitored for a period of 30 years to allow for radioactive material decay prior to dismantlement. This alternative would not change the amount of material handled as low-level radioactive waste due to the presence of long lived radionuclides. Deferred dismantlement would prevent the release of the Windsor Site for more than 30 years.

(3) **No Action** - The S1C Prototype reactor plant would be left in a drained, deenergized, stable condition and monitored for an indefinite period of time. This alternative would prevent the Windsor Site from being released for unrestricted use for an indefinite period.

The alternative of removing and disposing of the entire S1C Prototype reactor compartment in one piece (analogous to ongoing submarine reactor compartment disposal) was considered but eliminated from detailed analysis as infeasible, due to numerous transportation interferences such as load-limited bridges and width and height restrictions. The alternatives of entombment and on-site disposal were eliminated to avoid creation of a new radioactive and hazardous waste disposal site.

A comparison of the three alternatives examined in detail is provided in Table S-1. No new legislation would be required to implement any of these alternatives. The environmental effects from each alternative are small, as are the health effects and risks. Recycling and volume reduction services of commercial enterprises would be used to minimize the volume of low-level radioactive waste. The cost of the deferred dismantlement alternative is significantly higher than the prompt dismantlement alternative, and the no action alternative also would eventually result in higher costs.

From an environmental perspective, no single alternative stands out in this comparison. Deferred dismantlement has the advantage of minimizing occupational radiation exposure while still providing for eventual unrestricted release of the Windsor Site. Prompt dismantlement has the advantage of not requiring long term commitment of the land for surveillance and maintenance of the S1C Prototype reactor plant. The occupational radiation exposure associated with the prompt dismantlement alternative is comparable in magnitude to the radiation exposure routinely received during operation and maintenance of Naval prototype reactors. Also, the impacts associated with the prompt dismantlement alternative have a higher degree of certainty than those associated with actions thirty or more years in the future. Because prompt dismantlement would result in unrestricted release of the Windsor Site at the earliest time with little occupational radiation exposure risk to the workers, and given that the impacts associated with prompt dismantlement have a higher degree of certainty, Naval Reactors has identified prompt dismantlement as the preferred alternative.

Table S-1: Comparison of Alternatives

	Prompt Dismantlement Preferred Alternative	Deferred Dismantlement Alternative	No Action Alternative
Timing	Prompt start, 2-year dismantlement duration	30-year deferment, 2-year dismantlement duration	Indefinite deferment, no dismantlement
Number of Radioactive Material Shipments [1]	23	23	0
Number of Nonradioactive Material Shipments [2]	1600	1600	0
Additional Latent Fatal Cancer Risks or Fatal Injury Risks [3]			
Occupational [4] (Radiological)	4.3×10^{-2} to 7.9×10^{-2}	1.7×10^{-3} to 2.4×10^{-3}	8.4×10^{-4}
Occupational [5] (Nonradiological)	6.7×10^{-2}	7.4×10^{-2}	6.9×10^{-3}
Public [6] (Radiological)	9.7×10^{-4} to 2.6×10^{-3}	3.8×10^{-5} to 7.1×10^{-5}	1.6×10^{-5}
Public [7] (Nonradiological)	2.2×10^{-2} to 3.0×10^{-2}	2.2×10^{-2} to 3.0×10^{-2}	0
Estimated Cost [8]	$51,000,000	$64,800,000	$13,800,000 total for 30 years of caretaking

1. Data represents a conservatively high number of radioactive material shipments consisting of 19 miscellaneous waste package shipments and 4 major component package shipments such as the reactor pressure vessel, pressurizer and steam generators. As discussed in Sections 5.1.13 and 5.2.13, approximately 10 of the shipments (approximately 110 cubic meters) would be low-level radioactive waste requiring disposal. The other 13 shipments would be to commercial vendors for recycling and volume reduction processing.

2. Data represents the number of shipments of nonradioactive waste and recyclable materials from S1C Prototype dismantlement and Windsor Site demolition activities. Data also includes deliveries of fill and topsoil for Windsor Site restoration activities.

3. Values listed include latent fatal cancer risks due to incident-free activities and accident scenarios as well as fatal injury risk from accidents. For the public, the numbers provide a range since the value strongly depends on the distance to the disposal site. For the purpose of bounding the transportation related impacts, the disposal site was assumed to be either the Department of Energy Savannah River Site in South Carolina or the Department of Energy Hanford Site in Washington State. The occupational risk values do not strongly depend on distance to the disposal site.

4. Occupational (Radiological) risks apply to the on-site worker and transportation worker population. Occupational latent fatal cancer risks are calculated by multiplying occupational exposure in rem for the total on-site worker and transportation worker population by 0.0004 additional latent fatal cancers per rem. The range provided for the prompt and deferred dismantlement alternatives reflects the uncertainty in occupational exposure estimates during the dismantlement of the reactor plant. Individual worker exposure would be limited to two rem per year. Two rem results in a risk of 0.0008 additional latent fatal cancers.

5. Occupational (Nonradiological) risks result from transportation and industrial worker accidents.

6. Public (Radiological) data accounts for effects on the general public from activities associated with on-site work and transportation of radioactive recyclable material and waste. Public latent fatal cancer risks are calculated by multiplying general population exposure in rem by 0.0005 additional latent fatal cancers per rem.

7. Public (Nonradiological) data accounts for effects on the general public from nonradiological causes related to transportation vehicle exhaust emissions and accidents. The No Action alternative does not involve any transport of materials.

8. Estimated costs are presented in 1996 dollars. Taking into consideration the eventual need for a permanent disposal decision, the no action alternative would ultimately result in a higher figure.

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CHAPTER 1
PURPOSE AND NEED FOR AGENCY ACTION

1.0 PURPOSE AND NEED FOR AGENCY ACTION

Naval Reactors is currently evaluating alternatives for disposal of the S1C Prototype reactor plant, located at the Knolls Atomic Power Laboratory Windsor Site in Windsor, Connecticut. The function of the Windsor Site and the S1C Prototype was to train Navy personnel and test propulsion plant equipment. As a result of the end of the Cold War and the downsizing of the Navy, the S1C Prototype reactor plant was permanently shut down in March 1993. Because the S1C Prototype reactor plant is the only activity at this small site and there is no further need for this plant, a decision is needed on its disposal.

1.1 THE PROPOSED ACTION

Naval Reactors proposes to determine and implement a disposal strategy for the defueled S1C Prototype reactor plant.

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CHAPTER 2
S1C PROTOTYPE AND WINDSOR SITE DESCRIPTION

2.0 S1C PROTOTYPE AND WINDSOR SITE DESCRIPTION

The following sections describe the Windsor Site facilities and characterize the S1C Prototype reactor plant and associated reactor compartment structure.

2.1 Windsor Site - General Description

Owned by the Department of Energy, the Windsor Site was established in 1957. It is situated on 10.8 acres of land, in the Town of Windsor, Hartford County, Connecticut (approximately five miles north of the City of Hartford). See Figures 2-1 and 2-2. The Windsor Site is currently operated by KAPL, Inc., a Lockheed Martin company, under contract with the U.S. Department of Energy.

The Windsor Site mission was to train Navy personnel in the operation and maintenance of Naval nuclear propulsion plants for the Navy fleet and to test Naval nuclear propulsion plant equipment. The Windsor Site includes one pressurized-water Naval nuclear propulsion plant, known as the S1C Prototype, and miscellaneous support facilities. Most of the remaining support facilities are located within a fenced security area as shown in Figure 2-3. Parking lots are located outside the security fence. Historical information regarding the Windsor Site is contained in the Windsor Site Environmental Summary Report (Reference 2-1).

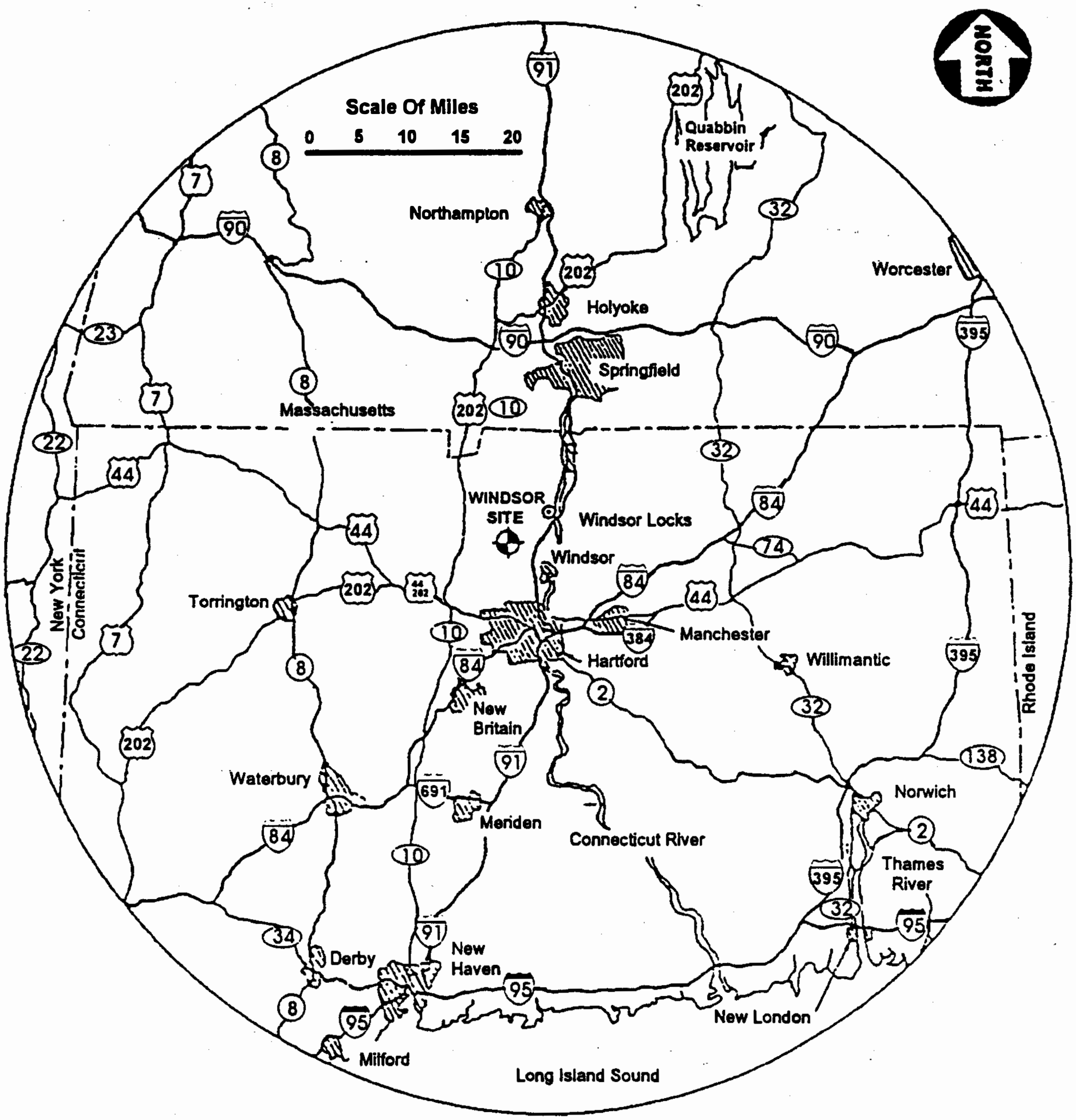


Figure 2-1 Eighty Kilometer (50 Mile) Assessment Area Map for the Windsor Site

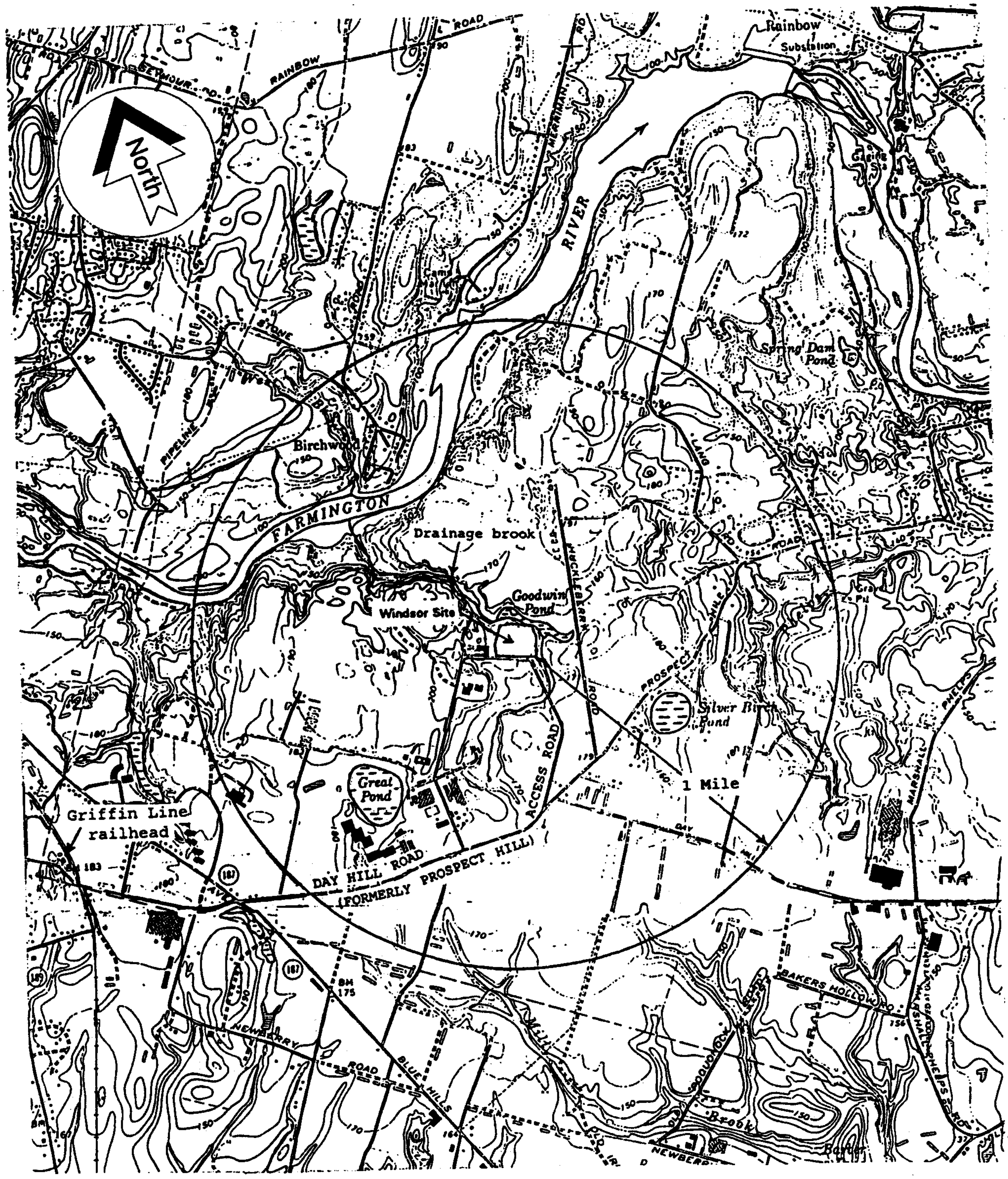
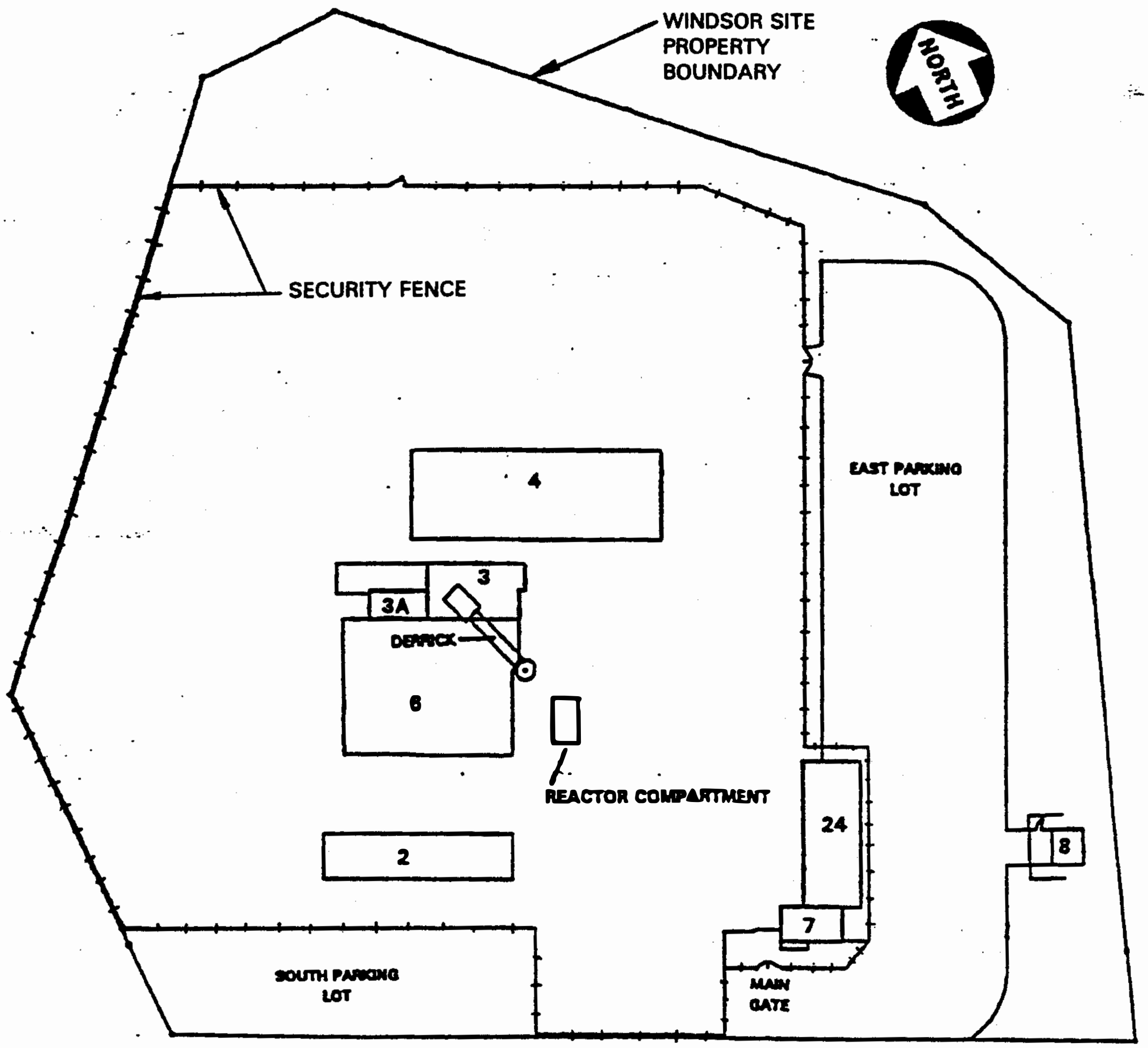


Figure 2-2: Proximity of Windsor Site to the Farmington River



2	Auxiliary Shops	6	Operations Building
3	Power Plant Building	7	Guard House
3A	Battery Room	8	Well Pump House
4	Warehouse	24	Offices

Figure 2-3: Projected Windsor Site Layout at the Start of Each Alternative

2.2 S1C Prototype - General Description

The S1C Prototype was placed in operation in 1959. In addition to its use as a training platform, the S1C Prototype served as a test facility for propulsion plant equipment. Removal of the spent nuclear fuel from the reactor (defueling) was completed in February 1995 and the spent fuel has been removed from the Windsor Site. Management of spent fuel are addressed in a separate Environmental Impact Statement, Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement (Reference 1-1).

Defueling removed all of the fuel assemblies which contained uranium and fission products. Defueling removed more than 95% of the radioactive material inventory from the S1C Prototype reactor plant. After defueling, S1C Prototype reactor plant systems were drained and placed in a stable protective storage condition.

Figure 2-4 provides a sketch of the S1C Prototype reactor compartment. The hull construction duplicates as completely as possible the comparable section in a seagoing submarine. The S1C Prototype reactor compartment is a horizontal cylinder (approximately 24 feet diameter by 23 feet long) formed by a section of the prototype's pressure hull and provides the containment structure for the reactor plant. Stiffened steel bulkheads separate the reactor compartment from the remainder of the prototype. The reactor compartment bulkheads are shielded.

Appendix A provides additional general information for a typical Naval prototype reactor compartment.

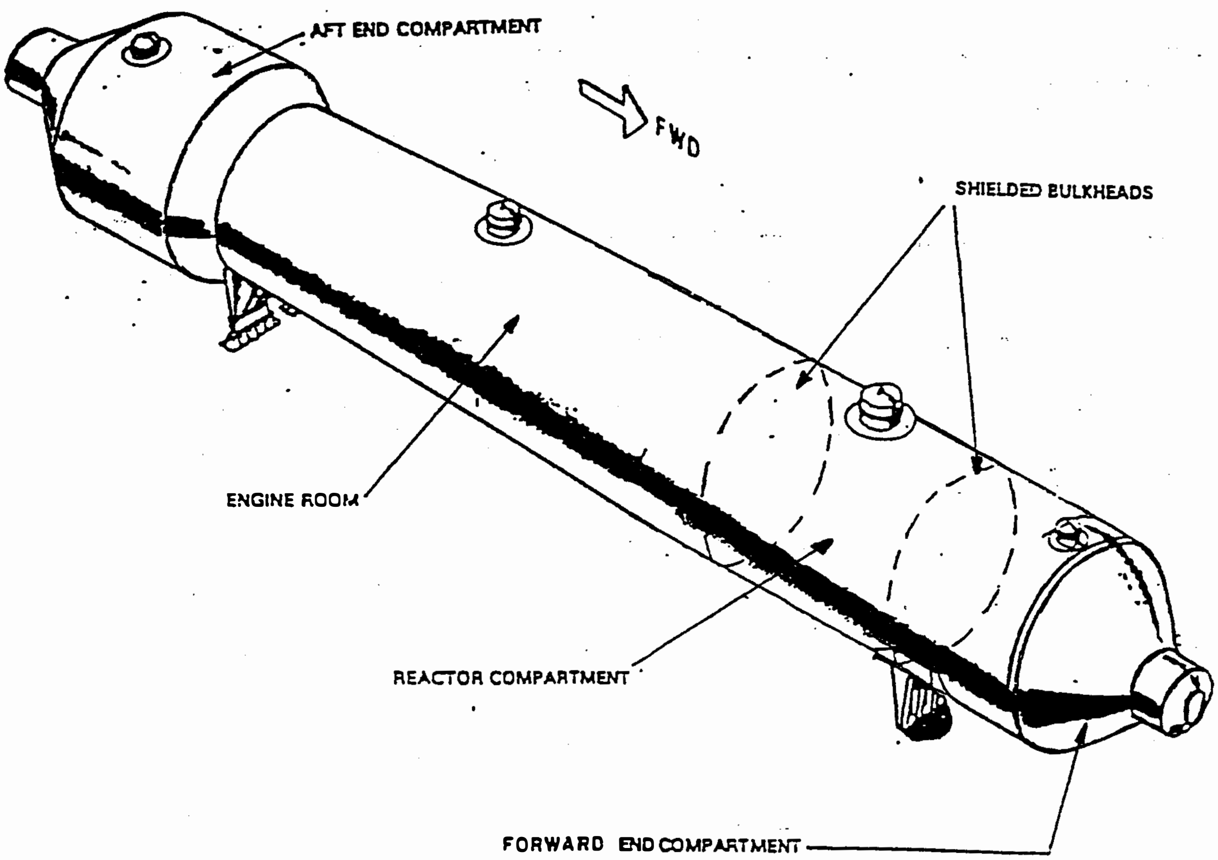


Figure 2-4: S1C Prototype Reactor Compartment

2.3 Radiological Characterization of the S1C Prototype Reactor Plant

Table 2-1 lists the radionuclide inventory that is expected in the defueled S1C Prototype reactor plant at various times after shutdown. Data for four years after shutdown represent the radiological conditions expected for the Prompt Dismantlement Alternative. Data for thirty-four years after shutdown represent radiological conditions expected for the Deferred Dismantlement Alternative.

Cobalt-60 is one of the predominant radionuclides in activated corrosion and wear products within the reactor plant systems. Gamma radiation from cobalt-60 is the major source of radiation exposure in the defueled S1C Prototype reactor plant. Cobalt-60 has a 5.27-year half-life and emits beta and penetrating gamma radiation.

While iron-55 is one of the predominant radionuclides at the time of shutdown in terms of numbers of curies, it is not significant for disposal considerations. Iron-55 has a relatively short half-life of 2.73 years and emits nonpenetrating, low energy x-ray radiation. Iron-55 is not a major source of radiation exposure because the low energy x-rays emitted by iron-55 are stopped within the reactor plant piping and structure.

Some of the radionuclides included in the Table 2-1 list have long half-lives. Examples of long half-life radionuclides include nickel-63 (100 years, beta radiation), carbon-14 (5730 years, beta radiation), niobium-94 (20,000 years, beta and gamma radiation) and nickel-59 (76,000 years, weak x-ray). Nickel-63 and carbon-14 are not major sources of radiation exposure since the beta radiation they emit is stopped within the prototype reactor plant piping. Radiation from nickel-59 is also stopped within the prototype reactor plant piping. Niobium-94 is present in small quantities and would be the only measurable gamma radiation dose emitter after cobalt-60 and all of the other short half-life radionuclides have decayed away.

Appendix A provides additional general information on radioactivity and health effects from radiation exposure.

Table 2-1: Radioactivity by Individual Radionuclide Present in the Defueled S1C Prototype Reactor Plant Four Years and Thirty-Four Years After Final Reactor Shutdown

Radio-nuclide [1]	Half-life [2] (Years)	Radiation Emitted [2]	Radioactivity Four Years After Reactor Shutdown [3] (Curies)	Radioactivity 34 Years After Reactor Shutdown [3] (Curies)
Fe-55	2.73	X-ray	$9.83 \times 10^{+03}$	4.86
Co-60	5.27	beta and gamma	$4.19 \times 10^{+03}$	$8.14 \times 10^{+01}$
Ni-63	100	beta	$9.16 \times 10^{+02}$	$7.45 \times 10^{+02}$
Ni-59	76,000	X-ray	7.68	7.68
C-14	5,730	beta	2.26	2.25
H-3	12.3	beta	4.39×10^{-01}	8.10×10^{-02}
Mn-54	0.85	X-ray and gamma	5.89×10^{-02}	0.00
Nb-94	20,000	beta and gamma	3.79×10^{-02}	3.79×10^{-02}
Mo-93	3,500	X-ray and gamma	3.64×10^{-02}	3.62×10^{-02}
Tc-99	213,000	beta and gamma	8.91×10^{-03}	8.91×10^{-03}
Ba-137m [3]	0.000005	gamma	5.50×10^{-03}	2.76×10^{-03}
Sr-90	29.1	beta	5.50×10^{-03}	2.69×10^{-03}
Y-90 [4]	0.01	beta and gamma	5.48×10^{-03}	2.68×10^{-03}
Cs-137	30.2	beta and gamma	5.47×10^{-03}	2.77×10^{-03}
Pu-241	14.4	alpha, beta and gamma	1.23×10^{-03}	2.93×10^{-04}
Am-241	432.7	alpha and gamma	5.19×10^{-05}	4.97×10^{-05}
Cm-244	18.1	alpha and gamma	4.48×10^{-05}	1.42×10^{-05}
Co-58	0.19	X-ray, beta and gamma	3.82×10^{-05}	0.00
Pu-238	87.7	alpha and gamma	3.62×10^{-05}	2.85×10^{-05}
Zr-93	1,500,000	beta and gamma	3.54×10^{-05}	3.54×10^{-05}
Pu-239	24,100	alpha and gamma	7.76×10^{-06}	7.75×10^{-06}
		TOTALS:	**$1.50 \times 10^{+04}$**	**$8.41 \times 10^{+02}$**

1. The radionuclides listed were considered in facility and transportation accident evaluations in Appendices B and C, respectively. The amounts of radioactivity for each nuclide represent a combined total from activated metals (inseparable from the base metal) and activated corrosion products (which could potentially be released in the event of an accident). More than 99% of the remaining radioactivity in the defueled S1C Prototype reactor plant is an inseparable part of the metal components.

2. Chart of Nuclides, 14th Ed.

3. Ba-137m exists in equilibrium with its parent Cs-137.

4. Y-90 exists in equilibrium with its parent Sr-90.

2.4 Hazardous Materials Contained in the S1C Prototype Reactor Plant

The S1C Prototype reactor plant contains several types of hazardous materials, with lead being the most significant in quantity. The S1C Prototype reactor plant contains more than 100 tons of lead. Most of the lead is encased with welded steel sheets. The encased lead is permanently installed as radiation shielding in the form of panels. The lead inside the panels is either layered sheets, bricks or poured in place. Although the lead used for permanently installed shielding is highly refined, the lead contains a small amount of impurities. The lead closest to the prototype reactor was exposed to a neutron flux which caused the impurities to become activated, resulting in a mixed hazardous and radioactive material. Approximately 30% of the lead in the S1C Prototype reactor plant is estimated to contain activated impurities. There are a variety of other hazardous materials that may be present in small quantities in the S1C Prototype reactor plant. These hazardous materials are usually elemental metals such as lead, chromium and cadmium. Hazardous elements are sometimes found in equipment construction materials and as constituents in paint, leaded glass, adhesives, and brazing alloys.

In addition to hazardous materials, some S1C Prototype reactor plant components may contain regulated concentrations of polychlorinated biphenyls (greater than 50 parts per million). Examples of materials that could contain polychlorinated biphenyls as a constituent include paint, adhesives, electrical cable coverings and rubber items manufactured before the mid-1970s. In these examples, polychlorinated biphenyls are usually tightly bound in the composition of the solid material. While the amount of polychlorinated biphenyls is small by weight, its use as a constituent in paint could affect a large number of components. Painted surfaces in the S1C Prototype reactor plant include the hull, decking support structures, pipe hangers, equipment foundations and thermal insulation.

Some items in the S1C Prototype reactor plant are insulated with asbestos-containing materials, typical of piping systems constructed before the mid-1970s. Thermal insulation that contains asbestos is installed on the steam generators, pressurizer and some piping. Miscellaneous items may also include asbestos-containing materials. Examples of miscellaneous items that could contain asbestos include electrical cable insulation, small components in electrical equipment, and gaskets in mechanical systems.

2.5 Applicable Regulatory Considerations

This section provides a general discussion of the environmental statutes and regulations that are applicable to Windsor Site activities.

2.5.1 Federal Environmental Statutes and Regulations

Applicable Federal statutes for the Windsor Site activities include:

National Environmental Policy Act of 1969, as amended (42 USC §4321 et seq.)

The National Environmental Policy Act establishes a national policy promoting awareness of the environmental consequences of human activities and promoting consideration of the environmental impacts during planning and decision making stages of a project. This law requires all Federal agencies to prepare a detailed statement on the environmental effects of proposed major Federal actions that may significantly affect the quality of the human environment. This environmental impact statement has been prepared in accordance with the Council on Environmental Quality regulations for implementing the procedural provisions of the National Environmental Policy Act (40 CFR Parts 1500-1508) and Department of Energy National Environmental Policy Act Implementing Procedures (10 CFR Part 1021).

Atomic Energy Act of 1954, as amended (42 USC §2011 et seq.)

This law authorizes the Department of Energy to establish standards to protect health or minimize dangers to life or property with respect to activities under its jurisdiction. Through a series of Department of Energy orders, an extensive system of standards and requirements has been established to ensure safe operation of facilities.

Clean Air Act, as amended (42 USC §7401, et seq.)

The Clean Air Act is intended to protect and enhance the quality of the Nation's air resources and to promote the public health and welfare and the productive capacity of its population. The Act requires each Federal agency to comply with all Federal, state, interstate, and local requirements with regard to the control and abatement of air pollution to the same extent as any non-governmental entity. The Clean Air Act established the National Ambient Air Quality Standards program for criteria pollutants. Criteria pollutants include sulfur dioxide, nitrogen oxides, particulate matter, carbon monoxide, ozone and lead. Sources of air pollution are subject to regulation through limitations contained in U.S. Environmental Protection Agency approved State Implementation Plans. The Clean Air Act also addresses specific pollution problems such as hazardous air pollutants and visibility impairment. New and modified sources are regulated to more stringent controls based on available pollution technology.

The State of Connecticut Department of Environmental Protection has been delegated authority to implement and enforce Federal standards and other requirements for some emissions of hazardous air pollutants (such as asbestos) pursuant to Section 112 of the Clean Air Act. Notwithstanding these delegations, the Environmental Protection Agency retains authority for enforcing any rule, standard or requirement established under Section 112.

Clean Water Act, as amended (33 USC §1251 et seq.)

The Clean Water Act was enacted to restore and maintain the chemical, physical and biological integrity of the nation's water. The Act requires each Federal agency to comply with all Federal, state, interstate, and local requirements with regard to any activity that might result in the discharge or runoff of pollutants to surface waters in the same manner and to the same extent as any non-governmental entity. The National Pollutant Discharge Elimination System program is administered by the Water Management Division of the U.S. Environmental Protection Agency. The State of Connecticut Department of Environmental Protection, Water Management Bureau, has regulatory authority for the National Pollution Discharge Elimination System program in the State of Connecticut. Storm water drainage associated with the Windsor Site is also regulated by the State of Connecticut Department of Environmental Protection, Water Management Bureau.

Resource Conservation and Recovery Act, as amended (42 USC §6901 et seq.)

The treatment, storage, or disposal of hazardous and nonhazardous waste is regulated under the Solid Waste Disposal Act, as amended by the Resource Conservation and Recovery Act, the Hazardous and Solid Waste Amendments of 1984, and the Federal Facility Compliance Act. The U.S. Environmental Protection Agency regulations implementing the Resource Conservation and Recovery Act are found in 40 CFR Parts 260 - 280. These regulations define hazardous wastes and specify hazardous waste transportation, handling, treatment, storage, and disposal requirements. The regulations imposed on a generator or a treatment, storage and/or disposal facility vary according to the type and quantity of materials or wastes involved. The U.S. Environmental Protection Agency has granted final authorization to the State of Connecticut Department of Environmental Protection to operate its hazardous waste program, subject to the authority retained by the U.S. Environmental Protection Agency in accordance with the Hazardous and Solid Waste Amendments of 1984. As a result, there is a dual State and Federal regulatory program in Connecticut. To the extent the authorized State program is unaffected by the Hazardous and Solid Waste Amendments, the State program operates in lieu of the Federal program. Where Hazardous and Solid Waste Amendments apply, the U.S. Environmental Protection Agency administers and enforces these provisions until the State receives authorization to do so.

Comprehensive Environmental Response, Compensation, and Liability Act (42 USC §9601 et seq.)

This Act provides a statutory framework for the clean up of waste sites containing hazardous substances and - as amended by the Superfund Amendments and Reauthorization Act - provides an emergency response program in the event of a release (or threat of a release) of a hazardous substance to the environment. Using the Hazard Ranking System, Federal and private sites are ranked and may be included on the National Priorities List. The Act requires Federal facilities having such sites to undertake investigations and remediation as necessary. The Act includes requirements for reporting releases of certain hazardous substances in excess of specified amounts to State and Federal agencies. Section 120(h) of the Act establishes Federal Agency notification requirements for selling or transferring Federal property where any hazardous substance was either stored for one year or more, or known to have been released, or known to have been disposed of on the property. The Environmental Protection Agency regulations implementing property transfer requirements are found in 40 CFR Part 373.

An environmental evaluation of conditions at the Windsor Site, called a Preliminary Assessment, was conducted in 1988 in accordance with the Act. As a result of the Preliminary Assessment, the Environmental Protection Agency placed the Windsor Site in 1990 in the category of No Further Remedial Action Planned within the Federal Superfund program.

Toxic Substances Control Act (15 USC §2601 et seq.)

This Act requires that the health and environmental effects of all new chemicals be reviewed before they are manufactured for commercial purposes. The Act authorizes the U.S. Environmental Protection Agency to secure information on all new and existing chemical substances and to control any of these substances determined to cause an unreasonable risk to public health or the environment. Regulated controls include the manufacture, use, distribution in commerce, and disposal of chemical substances, including polychlorinated biphenyls, and abatement of asbestos and lead.

Federal Facility Compliance Act (42 USC §6921 et seq.)

This Act amended the Resource Conservation and Recovery Act and requires the Department of Energy to prepare plans for developing the required treatment capacity for mixed waste stored or generated at each facility. The Site Treatment Plan for Mixed Wastes Generated at the Windsor Site was approved by the U.S. Environmental Protection Agency. The State of Connecticut Department of Environmental Protection reviewed and commented on the Site Treatment Plan and remains an active participant in related matters. The U.S. Environmental Protection

Agency, Region I, issued a Consent Agreement and Order regarding the Site Treatment Plan for Mixed Waste Generated at the Windsor Site that became effective on October 6, 1995.

Endangered Species Act, as amended (16 USC §1531 et seq.)

This Act is intended to prevent the further decline of endangered and threatened species and to restore these species and habitats. The Act is jointly administered by the U.S. Departments of Commerce and the Interior. The Act requires consultation with the U.S. Fish and Wildlife Service to determine whether endangered and threatened species or their critical habitats are known to be in the vicinity of the proposed action, and whether an action will adversely affect listed species or designated critical habitats.

Safe Drinking Water Act, as amended (42 USC §300f et seq.)

The Safe Drinking Water Act (SDWA) was enacted to protect potable water resources and ensure potable water quality. Among other things, the Act requires each Federal agency and department that owns or operates a public water system to comply with all Federal, State and local safe drinking water requirements. The Environmental Protection Agency has promulgated the SDWA regulations at 40 CFR Parts 140 - 149. The State of Connecticut Department of Public Health has primary enforcement responsibility for the regulations implementing the potable water quality requirements. The State of Connecticut Department of Environmental Protection has primary enforcement responsibility for the regulations implementing the protection of potable water resources.

2.5.2 Executive Orders

Executive Order 12344 (Naval Nuclear Propulsion Program)

Executive Order 12344, enacted as permanent law by Public Law 98-525 (42 USC §7158) prescribes the authority and responsibility of the Naval Nuclear Propulsion Program, a joint Navy/Department of Energy organization, for matters pertaining to Naval nuclear propulsion. These responsibilities include all environmental and occupational safety and health aspects of the program.

2.5.3 Department of Energy Regulations and Orders

The Department of Energy regulations are generally found in Title 10 of the Code of Federal Regulations. Department of Energy Orders generally set forth policy and programs and internal procedures for implementing department policies. These regulations address such areas as administrative requirements and procedures, general environmental protection, radiation protection of the public and the environment, radioactive waste management, and occupational health and safety. Department of Energy Orders are implemented by Naval Reactors under authority of Executive Order 12344.

2.5.4 Hazardous and Radioactive Materials Transport Regulations

Transportation of hazardous and radioactive materials, substances, and wastes are governed by the U.S. Department of Transportation, the U.S. Environmental Protection Agency, and the Nuclear Regulatory Commission regulations (49 CFR Parts 171-178 and Parts 383-397, 40 CFR Part 262, and 10 CFR Part 71, respectively).

Department of Transportation regulations contain requirements for identifying a material as hazardous or radioactive. These regulations interface with those of the Nuclear Regulatory Commission or the U.S. Environmental Protection Agency for identifying material, but the Department of Transportation hazardous material regulations govern the hazard communication (such as marking, hazard labeling, vehicle placarding, and emergency response telephone number) and shipping requirements (such as required entries on shipping papers or waste manifests).

Nuclear Regulatory Commission regulations applicable to the transportation of larger quantities of radioactive materials are found in 10 CFR Part 71, which includes detailed packaging design requirements and package certification testing requirements. Complete documentation of design and safety analysis testing results for these shipments is submitted to the Nuclear Regulatory Commission to certify the package for use.

2.5.5 State Environmental Statutes and Regulations

State of Connecticut environmental laws which apply to Windsor Site activities are found in Connecticut General Statutes, Title 22a - Environmental Protection. The laws are implemented in accordance with requirements contained in the Regulations of Connecticut State Agencies.

State of Connecticut Air Regulations

Regulations of Connecticut State Agencies, Title 22a, Chapter 174, provides applicable air pollution control standards. Regulated air pollutants include criteria pollutants such as carbon monoxide, sulfur dioxide, oxides of nitrogen, particulate matter, ozone, lead, hazardous air pollutants, odors and volatile organic compounds.

State of Connecticut Water Pollution Control Regulations

Regulations of Connecticut State Agencies, Title 22a, Chapter 430, provides applicable water pollution control standards. The State of Connecticut regulates the discharge of water (including waste water, sanitary system discharges and storm water runoff from industrial facilities), and the discharge of substances or material into the waters of the State. Waters of the State include all surface waters and ground waters. Discharges to waters of the State are regulated by permits.

State of Connecticut Hazardous Waste Regulations

Regulations of Connecticut State Agencies, Title 22a, Chapter 449c, Parts 100 through 110, define the State of Connecticut's hazardous waste management program. These regulations incorporate by reference and adopt Federal regulations pursuant to Subtitle C of the Resource Conservation and Recovery Act, with certain specified differences. Differences occur where the State regulations are more stringent than the Federal regulations. The State regulations include standards applicable to generators of hazardous waste, standards for owners and operators of hazardous waste treatment, storage and disposal facilities, and provide the requirements for applicable permits. Other solid waste management regulations are provided in the Regulations of Connecticut State Agencies, Title 22a, Chapter 209.

State of Connecticut Property Transfer Program

Sections 22a-134 through 22a-134e of the Connecticut General Statutes, as amended by Public Act 95-183, establish the State's property transfer law which must be complied with whenever an establishment is transferred. The program requires disclosure of environmental conditions through the filing of one of four available forms. The form executed and submitted is dependent upon the environmental history and condition of the property. When the transfer involves an establishment where there has been a release of hazardous waste, certification is required that the property has been investigated and cleaned up.

State of Connecticut Water Resources Regulations

Regulations of Connecticut State Agencies, Title 25, Chapter 128, Parts 56 and 57, provide applicable requirements for abandonment (closure) of wells. Requirements include plugging wells to prevent the entrance of surface water or any other process that could contaminate or pollute ground water resources. Regulations require well closure actions to be performed or directed by a registered well drilling contractor.

State of Connecticut Standards for Quality and Adequacy of Public Drinking Water

Regulations of Connecticut State Agencies, Title 19, Chapter 13, Part B102 provides the standards for quality of public drinking water supplies including applicable requirements for protection of public water supplies. Protection of drinking water requirements include a cross-connection control program and the use and maintenance of backflow prevention devices.

CHAPTER 3
ALTERNATIVES

3.0 ALTERNATIVES

The following sections discuss in detail three alternatives for the defueled S1C Prototype reactor plant - Prompt Dismantlement, Deferred Dismantlement and the No Action Alternative. Several other alternatives are also considered in limited detail.

3.1 Prompt Dismantlement Alternative (Preferred Alternative): Promptly Dismantle the S1C Prototype Reactor Plant; Recycle or Dispose of Materials

In this alternative, dismantlement of the defueled and drained S1C Prototype reactor plant would begin promptly after the Record of Decision for this Environmental Impact Statement is issued. The project would be completed as soon as possible. Upon completion of reactor plant dismantlement and shipment of recyclable materials and wastes, the Windsor Site property would be released for unrestricted use in accordance with applicable local, State and Federal regulations. Low-level radioactive waste from dismantlement would be shipped to the Department of Energy Savannah River Site for disposal.

3.1.1 Dismantlement Operations

Dismantlement operations would involve mechanical disassembly of all S1C Prototype reactor plant systems and the reactor compartment structure. In general, dismantlement would be sequenced based on a removal strategy that focuses on major reactor components. Major reactor components include the reactor vessel, steam generators, and pressurizer. Prior to the removal of each major component, interferences would be removed. Examples of typical interferences include electrical cables, reactor system piping, pumps, deckplates, and bulkhead sections. Disassembly techniques would include proven methods such as remote machine cutting of piping, grinding, sawing, flame cutting, and plasma arc cutting. Cutting techniques and radiological controls would vary depending on the application, location and radiological status of the affected component.

Operations on radiologically contaminated piping and components would be performed using appropriate personnel protection equipment and environmental protection measures to prevent the spread of radioactivity. The protective measures would adhere to the same standards and practices that were used to successfully control maintenance evolutions during plant operations. These protective measures include, but are not limited to:

- personnel training on mockups and practice in dismantlement tasks,
- protective clothing,
- radiation shielding,
- remotely operated tools,
- engineered containment enclosures and ventilation systems equipped with high efficiency particulate air filters,
- National Institute for Occupational Safety and Health approved respirators and breathing apparatus,
- personnel barriers around work zones, with radiation monitoring stations at all exits,
- sealing open ends of pipes, tubes and other components immediately upon disassembly,
- monitoring and appropriate sampling during work activities.

Similar techniques would also be applied to protect personnel and prevent the spread of nonradiological, hazardous materials.

The S1C Prototype reactor compartment and reactor plant components are within the reach and load capacity of the Windsor Site derrick crane. The derrick is located above a support building adjacent to the S1C Prototype hull (see Figure 2-3). Other lifting and handling equipment such as mobile cranes, fork lifts, jacking and blocking gear could also be used.

3.1.2 Waste Streams and Recycling

In order to minimize the volume of waste generated from prototype dismantlement, detailed material segregation efforts would occur. Segregation is a process of identifying and separating materials into different categories, known as waste streams. Dismantlement activities would generate the following waste streams:

- recyclable materials,
- nonhazardous and nonradioactive wastes,
- hazardous and/or toxic wastes,
- low-level radioactive wastes, and
- mixed wastes (radioactive and hazardous).

Emphasis would be placed on recycling as much material as practical. Segregating radioactive and hazardous materials increases the options for recycling. Most of the recyclable materials from dismantlement and demolition activities would be concrete, lead, carbon steel, and corrosion resisting metals. These materials would be recycled through various licensed commercial vendors.

Other than recyclable materials, the primary waste streams include low-level radioactive materials and mixed wastes. Estimated waste volumes are discussed in further detail in Chapter 5. Low-level radioactive waste includes solid, nonhazardous material only. Low-level radioactive waste would be disposed of at a Department of Energy disposal facility. The Savannah River Site in Aiken, South Carolina currently receives low-level radioactive waste from Naval Reactors sites in the eastern United States.

Mixed wastes are radioactive materials that include inseparable hazardous constituents, such as lead. Typically, mixed wastes generated would be a homogeneous solid (such as radiologically activated or surface contaminated lead), a nonhomogeneous solid (such as a radioactive item having an inseparable coating that contains a hazardous constituent), or a solidified liquid (such as a solidified solution that contains radioactive chromates). Mixed wastes are regulated by the Resource Conservation and Recovery Act (40 CFR Parts 260-271), Regulations of Connecticut State Agencies (Title 22a, Chapter 449c, Parts 100 - 110), as well as the Atomic Energy Act (42 USC §2011 et seq.). The processing and treatment of mixed wastes would be in accordance with the Site Treatment Plan for Mixed Wastes Generated at the Windsor Site, which was approved by the U.S. Environmental Protection Agency (Reference 3-2). The Site Treatment Plan includes volume projections of the mixed wastes to be generated during the dismantlement activities. Naval Reactors is currently evaluating recycling options to use radioactive lead in shielding applications in other Naval Reactors or Department of Energy facilities to further reduce estimated volumes of mixed waste.

Other nonradioactive, nonhazardous demolition debris from dismantlement activities that could not be recycled would be disposed of in accordance with all applicable Federal, State and local regulations.

3.1.3 Packaging and Transport of Recyclable Material and Waste

All recyclable material and waste shipments would be properly classified, described, packaged, marked and labeled for normal transportation conditions in accordance with all applicable regulations. Applicable regulations include 49 CFR Parts 171-179 (Transportation of Hazardous Materials), 10 CFR Part 71 (Packaging and Transportation of Radioactive Materials), Department of Energy orders and disposal site waste acceptance criteria.

Dismantlement of the S1C Prototype reactor plant would require an estimated 23 shipments of low-level radioactive recyclable material and waste. The largest shipment by weight, size and radioactive content would be the reactor pressure vessel. The reactor pressure vessel contains more than 99% of the total radioactivity in the S1C Prototype reactor plant that remains after defueling. Nearly all of this radioactivity results from neutron activation of the metal structure of the reactor pressure vessel and is therefore not loose. The reactor pressure vessel would be placed in a large, shielded shipping container for transport and disposal. This package would be moved by a heavy haul truck over public roads to the Griffin Line industrial track railhead, located approximately 1.5 miles west of the Windsor Site. The reactor pressure vessel package would then be transported by railroad to the Department of Energy Savannah River disposal site. In addition to the pressure vessel, one additional shipment by railroad may be necessary in order to ship the primary shield tank in a single large package. The Department of Energy Hanford disposal site is also considered in the transport analyses.

Large components, such as the steam generators and pressurizer, would be shipped to a Department of Energy disposal site or to a Nuclear Regulatory Commission licensed commercial recycle facility to reduce the volume of disposed waste. Highway shipments of all components and miscellaneous low-level radioactive recyclable material and waste would comply with Department of Transportation regulations and disposal site waste acceptance criteria.

3.1.4 Windsor Site Restoration and Site Release

The prompt dismantlement alternative would include the following actions to support restoration and site release:

- All Windsor Site systems would be completely removed, including all systems that are located below grade. With the exception of the building that housed the Windsor Site's former water supply well and pump system (Building 8, discussed below), all buildings located within the Windsor Site property boundary will be removed to at least four feet below grade. Paved areas, including the parking lots, and Windsor Site security fencing will be removed. The access roadway leading to the U.S. Government owned property will be left intact.

- Electrical service would be terminated. Light posts and associated wiring leading up to the Windsor Site which are located along the access road would be left in place.

- The municipal water supply to the Windsor Site would be shut off. Building 8 (see Figure 2-3) contains piping and backflow protection for the municipal water supply. This piping would be drained and laid-up and Building 8 would remain.

- A voluntary facility assessment (described in Chapter 4 and Appendix F) addressing the potential for environmental chemical contamination would be completed to support Windsor Site inactivation and future release of the property. Following completion of all sample collecting and analytical work, a report would be prepared and provided to the U.S. Environmental Protection Agency, Region I and the State of Connecticut Department of Environmental Protection. The report would summarize findings and would provide recommendations for any additional investigation or cleanup required to support the goal of unrestricted release of the Windsor Site.

- A final radiological survey of the Windsor Site would be performed to confirm that radioactivity levels in soils are below release criteria for future unrestricted uses of the property. Appendix G provides details on the final radiological survey of the Windsor Site, including the timing for performing the surveys. The action of confirming that applicable release criteria are met ensures that any future occupant at the Windsor Site would receive less radiation exposure than limits specified in Department of Energy Order 5400.5 (Reference 3-3) as well as the draft regulations under consideration by the Nuclear Regulatory Commission (Reference 3-4) and the Environmental Protection Agency (Reference 3-5). Draft regulations include a maximum exposure limit of 15 millirem per year from all sources of which a maximum of 4 millirem per year can be from ingestion of radioactivity in water. The final radiological survey would be conducted following a comprehensive strategy that measures radioactivity levels at the ground surface and takes systematic soil samples for analysis. Final survey results would be documented and reported to appropriate Federal and State regulatory agencies. Federal and State regulators would be invited to comment on the reports and to perform verification surveying and sampling.

Upon completion of reactor plant dismantlement, recyclable material and waste shipments, any necessary cleanup activities, verification sampling, and completion of any required Windsor Site restoration, the Windsor Site property would be released for unrestricted use in accordance with applicable local, State and Federal regulations. If there is no other use for the property by the U.S. Government, the land would be offered for sale. The Windsor Site property deed grants Combustion Engineering, Inc. a first right of refusal to acquire the property through the year 2010.

3.1.5 Cost

The cost of prompt dismantlement is estimated at $51,000,000 (1996 dollars). This estimate is a rough order of magnitude based on experience, engineering concepts, and comparison to similar Department of Energy and commercial projects. The principal dismantlement costs included in this estimate are preparation of engineering procedures, procurement or rental of special equipment, direct labor, support labor, waste disposal, utilities, and the voluntary facility assessment process. The highest single expense is the removal and disposal of the pressure vessel.

3.2 Deferred Dismantlement Alternative: Dismantle the S1C Prototype Reactor Plant After a 30-Year Caretaking Period to Allow Radioactive Decay; Dispose of or Recycle Materials

This alternative postpones S1C Prototype reactor plant dismantlement for a defined period of time (known as a caretaking period) to allow for radioactive decay of radioactive materials. The caretaking period selected for this alternative is 30 years. After completion of deferred reactor plant dismantlement and shipments of recyclable material and waste, the Windsor Site property would be released for unrestricted use as stated under the prompt dismantlement alternative.

Although similar analyses sponsored by the Nuclear Regulatory Commission have considered deferment periods of 50 or 60 years for commercial nuclear power plant dismantlements (Reference 3-6), Naval Reactors considers 30 years is appropriate for the S1C Prototype reactor plant for the reason explained below. Nearly all of the gamma radiation within the defueled S1C Prototype reactor plant comes from cobalt-60, which has an approximately 5.27-year half-life. Deferring dismantlement for 30 years would allow cobalt-60 to decay to less than 2% of the radioactivity levels present in 1997. Unlike the S1C Prototype reactor plant, defueled commercial nuclear power plants have a substantial amount of fission products cesium-137 and strontium-90 which have approximately 30-year half-lives. Due to the high integrity of Naval nuclear fuel assemblies, these fission products are not present in significant quantities after defueling of a Naval plant. Even after a longer deferment period of 60 years, commercial power plants would still contain approximately 25% of the cesium-137 and strontium-90 levels present at shutdown. Because cobalt-60 decays relatively quickly, further deferment beyond 30 years for the S1C Prototype reactor plant would provide little additional benefit in reducing the amount of remaining radioactivity. However, allowing cobalt-60 to decay away would not change the amount of materials handled as low-level radioactive waste, due to the presence of other longer-lived radionuclides.

3.2.1 Caretaking Period Operations

During the caretaking period, the defueled S1C Prototype reactor plant would be periodically monitored. The purpose of this monitoring would be to verify overall physical integrity of the reactor plant and to verify that all radioactivity remains contained.

Periodic radiological surveys of the reactor plant would be performed as part of a comprehensive environmental monitoring program to be maintained during the caretaking period. This monitoring program would be a continuation of the current monitoring program at the Windsor Site and would involve air sampling, the continuous monitoring of radiation levels at Site perimeter locations and at off-site locations, and the routine collection and analysis of water samples, sediment samples, and fish. Details of the current environmental monitoring program at the Windsor Site may be found in the annual Knolls Atomic Power Laboratory Environmental Monitoring Report for calendar year 1994 (Reference 4-10). Monitoring would identify any unexpected changes in radiological conditions in the reactor plant and at the Windsor Site. The only expected change would be decay of residual radioactivity.

During the caretaking period, the reactor compartment would be periodically ventilated. Ventilation system exhaust would pass through high efficiency particulate air filters. The system would be tested to verify that it is at least 99.95% efficient at removing potential airborne particulate radioactivity. The reactor compartment exhaust would be continually sampled with a fixed filter air sampler to verify that applicable environmental standards are met (Reference 3-1).

To preserve overall system and compartment integrity, the reactor compartment would be seasonally heated and dehumidified. In addition, visual inspections would be performed inside and outside of the reactor compartment. These inspections would include verification that known hazardous materials remain stable inside the compartment. Maintenance would be performed as necessary to maintain the physical integrity of the reactor plant.

Under this alternative, several buildings would remain at the Windsor Site in an inactive condition. The buildings would be used to support dismantlement operations after the 30-year caretaking period. These buildings would be seasonally heated and dehumidified and routinely inspected. Maintenance would be performed as necessary to sustain their physical integrity. During the caretaking period, access to fenced areas and buildings at the Windsor Site would be controlled by both a staffed security force and a remote alarm system.

3.2.2 Deferred Dismantlement Operations, Recycling, and Waste Disposal

Following completion of the 30-year caretaking period, reactor plant dismantlement would commence. Deferred dismantlement operations are assumed to be identical to dismantlement operations outlined in the prompt dismantlement alternative discussion, Section 3.1.1.

While the radioactive decay of cobalt-60 would substantially reduce occupational exposure associated with deferred dismantlement operations, many of the materials from reactor systems would still be radioactive due to the longer-lived radionuclides which remain after 30 years. The number and types of radioactive shipments associated with deferred dismantlement would be the same as for the prompt dismantlement alternative. Methods for packaging, transport, disposal and recycling would also be the same as discussed in the prompt dismantlement alternative discussion, Sections 3.1.2 and 3.1.3.

3.2.3 Windsor Site Restoration and Site Release

A voluntary facility assessment (described in Chapter 4 and Appendix F) is in process at the Windsor Site and would not be affected by selection of this alternative. Soil within or adjacent to the Windsor Site boundary that exceeds any applicable cleanup standards would be removed at the earliest time practical, consistent with current schedules. The extent of soil remediation, if required, is expected to be very small.

Under this alternative, the Windsor Site would not be released for unrestricted uses until completion of deferred reactor plant dismantlement. After completion of deferred reactor plant dismantlement activities, unnecessary buildings and utility systems would be removed consistent with discussion in Section 3.1.4. Unrestricted Windsor Site release would not occur before 2031.

3.2.4 Cost

The total cost of deferred dismantlement, including caretaking, is estimated at $64,800,000 (1996 dollars). The average caretaking cost is estimated at $477,000 per year (1996 dollars). The principal caretaking costs included in this estimate are direct labor for routine maintenance of the reactor plant and support buildings, radiological and environmental surveys, surveillance and security of the Windsor Site, utilities, and the voluntary facility assessment process. Over the course of 30 years, caretaking costs would total $14,300,000. Deferred dismantlement cost ($50,500,000) is assumed to be the same as prompt dismantlement cost described in section 3.1.5, except that the voluntary facility assessment process cost is included in the caretaking cost as discussed above.

3.3 No Action Alternative: Maintain the S1C Prototype Reactor Plant In Place and Monitor for an Indefinite Period of Time

The primary goal of this alternative would be to maintain the defueled S1C Prototype reactor plant in a stable condition for an indefinite period of time. This alternative involves no dismantlement operations or waste shipments of any kind. This alternative does not provide permanent disposal of the S1C Prototype reactor plant. Disposal of the S1C Prototype reactor plant would be required at some time in the future.

3.3.1 Caretaking Period Operations

Caretaking period operations for this alternative would be identical to caretaking period operations described in the deferred dismantlement alternative (section 3.2.1), except that the voluntary facility assessment process (described in Chapter 4 and Appendix F) and the radiological survey process (discussed in Appendix G) and any associated remediation activities would not be completed. Also, this alternative would have no defined end date.

3.3.2 Cost

The annual cost of this alternative is estimated at $460,000 (1996 dollars), which is the same as the annual caretaking cost for the deferred dismantlement alternative with the exception of the costs associated with the voluntary facility assessment process, which would not be completed. The total cost over a 30-year period is estimated to be $13,800,000. However, the no action alternative does not provide a permanent solution for S1C Prototype reactor plant disposal. Taking into consideration the eventual need for a permanent disposal decision, the no action alternative would ultimately result in a higher figure.

3.4 Other Alternatives

Other alternatives were also considered for this Environmental Impact Statement, but were eliminated from detailed evaluation for various reasons. These alternatives are described in the following sections.

3.4.1 One-Piece Reactor Plant Disposal Off-Site

This alternative is based on the submarine reactor compartment disposal program currently in use for dismantling decommissioned U.S. Navy submarines. Defueled reactor compartments are packaged in their entirety at the Puget Sound Naval Shipyard. The packaged reactor compartments are then sent by barge and special ground transport for disposal at the Department of Energy Low-Level Waste Burial Grounds, Hanford Site, State of Washington. As a single package, the S1C Prototype reactor compartment would measure approximately 23 feet in length, 24 feet in diameter and would weigh approximately 400 tons.

Unlike Puget Sound Naval Shipyard, the Windsor Site is not located adjacent to navigable water. Transport of the S1C Prototype reactor compartment in its entirety to the nearest barge facility on the Connecticut River is considered impractical due to many highway interferences and load-limiting bridges. Transport of the S1C Prototype reactor compartment in one piece by rail was also ruled out due to load-limiting bridges, interferences with bridge underpasses, and tunnels along routes to navigable water or potential disposal sites.

3.4.2 Entombment Alternative

The entombment alternative involves leaving the S1C Prototype reactor plant permanently at the Windsor Site within a strong, durable structure. There are many possible designs for a suitable entombment structure, ranging from the prototype reactor compartment, that currently contains the reactor plant, to additional massively reinforced concrete enclosures. The entombment structure could be located either above grade or below grade. Entombment structures are typically designed to last between several hundred to thousands of years to ensure containment of very long-lived radionuclides that remain in the reactor plant.

The Windsor Site has never been used for burial or permanent storage of radioactive or hazardous waste materials. In addition to radioactivity, the S1C Prototype reactor compartment contains a significant quantity of lead shielding - a hazardous material. Entombment alternatives would require regulatory approval to change the land use criteria of the Windsor Site property. Entombment alternatives would prevent unrestricted release of the Windsor Site property. Based on the small size of the property, the fact that it has had no historical use as a waste burial site, and given the land disposal restrictions for radioactive materials, entombment alternatives were not examined further.

3.4.3 On-Site Disposal

On-site disposal alternatives involve placing the S1C Prototype reactor compartment and contained reactor plant underground and covering it with a series of impervious materials and earth. Like the entombment alternative, based on the small size of the Windsor Site property, the fact that it has had no historical use as a waste burial site, and land disposal restrictions, on-site disposal was not examined further.

CHAPTER 4
AFFECTED ENVIRONMENT

4.0 AFFECTED ENVIRONMENT

This chapter provides baseline environmental conditions pertaining to the Windsor Site property (described in Chapter 2) and surrounding areas.

4.1 Land Use

There are two categories of industrial use areas established in the Town of Windsor zoning regulations (Reference 4-24). I-1 Industrial Zones allow for low intensity industrial uses and I-2 Industrial Zones provide for general, higher intensity industrial uses. Industrial zoning regulations allow for higher noise generation compared to other zoning categories. The Windsor Site property and the surrounding Combustion Engineering, Inc. property are located within an area classified by the Town of Windsor as an I-2 Industrial Zone (Reference 4-1). Three other small properties that border the east side of the Windsor Site access road are located within an area zoned for I-1 industrial use. Currently, only one small office building is located on the adjacent properties east of the access road.

In general, the broader surrounding area includes a mixture of residential, agricultural and industrial uses. The nearest residential areas are located about 0.25 miles to the northeast and southeast of the Windsor Site, and about 0.6 miles northwest of the Windsor Site, across the Farmington River (see Figure 2-2). Agricultural areas are mixed in with the residential areas, though located mainly along the floodplains of the Farmington River. Agricultural areas in the vicinity of the Windsor Site consist mostly of tobacco and shrub farms, but other crops include sweet corn and potatoes. Land about 0.5 miles north of the Windsor Site is zoned public and quasi-public, and includes the Windsor-Bloomfield Landfill and the recreational Northwest Park. Bradley International Airport is located about three miles north-northeast of the Windsor Site.

4.2 Ecological Resources

4.2.1 Terrestrial Ecology

The Windsor Site is a small, developed area that is located within a broad basin of gently rolling terrain called the Connecticut River Valley. The natural state of the land was changed during S1C Prototype and Windsor Site construction more than thirty years ago. Almost all of the Windsor Site property has been developed and is covered with tarmac, concrete or crushed stone. The area within the Windsor Site property boundary has no ecological resources of significance. Plant and animal species sensitive to disturbance by human activities have not been observed at the Windsor Site. The area surrounding the Windsor Site is covered with vegetation.

4.2.2 Wetlands

A U.S. Department of the Interior National Wetlands Inventory map shows numerous wetlands dotting the region surrounding the Windsor Site (Reference 4-2). However, there are no wetlands located on the Windsor Site property. The wetland nearest the Windsor Site is associated with Goodwin Pond (see Figure 2-2). The access road to the Windsor Site crosses the southern portion of the wetland at one point. The wetland is not being impacted by current activities at the Windsor Site.

4.2.3 Aquatic Ecology

The principal game fishes in the surrounding area are Atlantic salmon, brown trout, northern pike, bass (largemouth and smallmouth), and shad. Shad is an important commercial fish as well. Information on game fish takes in the area is not available. Shad fishing is concentrated between the community of Wilson and the Enfield Dam on the Connecticut River. Some shad are also caught in the Farmington River, between the Interstate 91 crossing and the Connecticut River. The season typically runs from mid-April to early June. The State of Connecticut maintains an active program of fish stocking. Nongame fish found in the Farmington River include perch, catfish, sunfish, carp, herring, shiner, and eel. Fish found in Goodwin Pond include bluegill, perch and bass.

4.2.4 Critical Habitats and Endangered Species

The Windsor Site is located in an ecoregion known as the North-Central Lowlands. The State of Connecticut Department of Environmental Protection lists several significant biological habitats and rare plant species characteristic to this ecoregion (Reference 4-3); however, there are none found on the Windsor Site property. According to a State of Connecticut Department of Environmental Protection review of the Natural Diversity Data Base, there are no known existing populations of Federal or State endangered, threatened or special concern species currently present at or in the vicinity of the Windsor Site (Reference 4-4).

4.3 Water Resources

4.3.1 Surface Water

The Windsor Site property does not include any bodies of open surface water. The nearest body of open surface water, Goodwin Pond, is located immediately north and east of the Windsor Site. The pond is manmade and predates Windsor Site construction. The pond drains northwest about 3600 feet along a drainage brook to the Farmington River (see Figure 2-2). A hydrogeologic evaluation performed in 1982 estimated the mean annual discharge of the brook into the Farmington River at 1.8 cubic feet per second (Reference 4-5). The State of Connecticut Department of Environmental Protection has classified Goodwin Pond and its drainage brook as suitable for "fish and wildlife habitat, recreational use, agricultural use, industrial supply and other legitimate uses including navigation" (References 4-6 and 4-7).

Goodwin Pond currently only serves as a fish and wildlife habitat with no recreational, agricultural, industrial or navigation uses.

The Farmington River is regulated by the Rainbow Reservoir and continues from there to join the Connecticut River. The U.S. Geological Survey has estimated the mean annual flow downstream of the Windsor Site at the Rainbow Reservoir to be about 1,100 cubic feet per second (Reference 4-8). The State of Connecticut Department of Environmental Protection has classified the Farmington River and Rainbow Reservoir as suitable for "fish and wildlife habitat, recreational use, agricultural use, industrial supply, and other legitimate uses including navigation" (Reference 4-6 and 4-7). Principal recreational activities on the Farmington River are swimming, fishing, and boating limited to small boats and canoes.

The Windsor Site is not located in a floodplain. The Flood Insurance Rate Map for the Town of Windsor shows that the Windsor Site property is in an area of minimal flooding and is above the 500-year flood boundary (Reference 4-9). There are no records of flooding on the Windsor Site property.

4.3.2 Ground Water

Geologic and aquifer test data suggest that two overburden aquifer systems underlie the Windsor Site: an upper relatively fine-grained unconfined aquifer and a lower (at least on the east side of the Windsor Site) coarse-grained semi-confined aquifer. Depth to the water table is typically 25 to 30 feet below grade. Ground water within the upper aquifer has been generally interpreted to flow easterly into the southwest portion of the Windsor Site and then more northeasterly and northerly toward Goodwin Pond and the drainage brook. The State of Connecticut Department of Environmental Protection has classified ground water at the Windsor Site as suitable as "industrial process and cooling water," but "presumed not suitable for direct human consumption without treatment" (References 4-6 and 4-7).

The Windsor Site has an ongoing ground water monitoring program that measures a variety of organic and inorganic parameters from four monitoring wells. These wells monitor the upper aquifer in order to determine the impacts from industrial activities. Details of the Windsor Site's ground water monitoring program and sampling results are described in the annual Knolls Atomic Power Laboratory Environmental Monitoring Report (Reference 4-10). Samples from the four monitoring wells have been taken by Knolls Atomic Power Laboratory personnel since 1984.

The service water production wells are located at the southeast corner of the Windsor Site. The service water production wells pumped ground water from the lower aquifer. The results of Windsor Site service water samples collected from the production wells met all State of Connecticut public health standards for drinking water (Reference 4-10). The production wells were removed from service in March 1994 when the Windsor Site was connected to the Town of Windsor municipal water supply. Since they are no longer needed, the production wells will be closed in accordance with applicable State of Connecticut regulations.

4.3.3 Existing Radiological Conditions - Water Resources

From 1959 to 1978, water containing low concentrations of radioactivity was discharged from the Windsor Site to the shallow, low flow, drainage brook between Goodwin Pond and the Farmington River (References 2-1 and 4-10). The drainage brook is located on property owned by Combustion Engineering, Inc. and is not readily accessible to the public. The concentration of radioactivity in the Windsor Site water discharges never exceeded applicable Federal standards. Since 1979, only nonradioactive water discharges have been released from the Windsor Site into the drainage brook.

The liquid effluent monitoring program at the Windsor Site consists of radiological monitoring of the drainage brook, Goodwin Pond, and Farmington River water. Aquatic life in the vicinity of the Windsor Site is also monitored. The purpose of the monitoring is to determine the effect from operations on the general public and surrounding environment. Analysis results of water and fish collected from the Farmington River have shown no radioactivity attributable to former operations. Conditions in the brook sediment from these discharges are discussed in Section 4.5.4.2.

4.3.4 Existing Nonradiological Conditions - Water Resources

Nonradiological waste water discharges from the Windsor Site included non-contact cooling water, retention tank liquids, and other Windsor Site process waters. Waste water effluent was released through a section of the storm drain system to the drainage brook that flows into the Farmington River. These waste water discharges were permanently secured in October 1995. In February 1996, the State of Connecticut Department of Environmental Protection acknowledged termination of all process discharges from the Windsor Site. The Windsor Site is no longer authorized to discharge any waste water under the previously issued discharge permit (National Pollutant Discharge Elimination System permit number CT0002020). As discussed in the annual Knolls Atomic Power Laboratory Environmental Monitoring Report, the analytical results for chemical constituents present in the Windsor Site liquid effluents at the Windsor Site outfall have been well within applicable standards (Reference 4-10).

Sanitary waste is discharged to an on-site septic system. The septic system includes an auxiliary clarification chamber and a septic tank for anaerobic treatment of the waste. The resultant discharge is released below ground through a leach field located at the north end of the property.

Storm water drainage from the Windsor Site is monitored for compliance with a State of Connecticut Department of Environmental Protection General Storm water Discharge Permit. Under the permit, the Windsor Site is required to annually sample Storm water drainage for parameters such as copper, lead, zinc, and coliform, among others, and to test the biotoxicity of these constituents on aquatic life.

Windsor Site Storm water drainage analysis results have shown higher levels for copper, lead and zinc than guideline values set at the time for this type of effluent. As a result, additional sampling was performed for the affected discharge points and reported to the State as required by the general Storm water permit. On October 1, 1995, the State of Connecticut modified the General Permit for the Discharge of Storm water Associated with Industrial Activity. This modification changed the comparison value for copper from 0.014 milligrams per liter to the current value of 0.200 milligrams per liter. Windsor Site Storm water sampling results from 1996 ranged from none detectable to 0.130 milligrams of copper per liter.

Prior to 1980, water containing chromate compounds was discharged from the Windsor Site to the drainage brook. Chromate compounds were added to Windsor Site cooling water systems to inhibit corrosion and biological growth. Conditions in the brook sediment from these discharges are discussed further in Section 4.5.5.2.

4.4 Air Resources

4.4.1 Climate and Meteorology

The climate in the region of the Windsor Site is typical for a northern temperate climate zone. The prevailing west to east movement of air in the region carries the majority of weather systems into the Windsor area from the west. The location of the Windsor Site, relative to the continent and ocean is noteworthy in that rapid weather changes can result when storms move northward along the Mid-Atlantic coast. The overland air masses produce a frequent passage of low-pressure systems, punctuated by occasional winds from the ocean. The result is a rather variable climate, with cloudy and clear skies alternating as often as twice a week.

Seasonally, weather characteristics vary from the cold and dry continental-polar air of winter to the warm, maritime air of summer. Typical minimum and maximum temperatures are 19°F and 84°F respectively and the average temperature is approximately 50°F. Annual snowfall is approximately 48 inches per year and snow cover is generally present from late December through early March. Precipitation is fairly uniform throughout the year and averages approximately 43 inches per year. Prevailing winds are north to northwest during the winter and south to southwest during the rest of the year (Reference 4-11). Infrequent winds may attain velocities up to 65 miles per hour and are likely to come from the northwest.

4.4.2 Severe Weather Phenomena

The State of Connecticut is subject to about one tornado per year. In contrast, the neighboring States of Massachusetts and New York average 3 and 4 tornadoes a year, respectively. In the period of 1953-1989, 49 tornadoes were recorded in Connecticut, with eight occurring in 1973 (Reference 4-12). All tornadoes have occurred in the summer months (Reference 4-11).

Storms of tropical origin occasionally affect Connecticut during the summer or fall months, as they move on a path well out over the ocean. However, hurricanes have been known to strike areas of Connecticut full force resulting in substantial property damage and loss of life (Reference 4-11).

4.4.3 Air Quality

Air quality in the Windsor area meets the National Ambient Air Quality Standards established for oxides of nitrogen, sulfur dioxide, carbon monoxide, lead, suspended particles, and particulate matter (PM-10). The region has been designated as a serious ozone nonattainment area and is in the Northeast Ozone Transport Region.

4.4.4 Existing Radiological Conditions - Air Resources

Operations having a potential for the release of airborne particulate radioactivity are serviced by monitored exhaust systems and regulated under the National Emission Standard for Emissions of Radionuclides Other Than Radon from Department of Energy Facilities, 40 CFR Part 61 Subpart H, by the U.S. Environmental Protection Agency. Prior to release, the exhaust air is passed through high efficiency particulate air filters to minimize radioactivity content. As reported in the Knolls Atomic Power Laboratory Environmental Monitoring Report for calendar year 1994 (Reference 4-10), the radioactivity contained in exhaust air during 1994 consisted of less than 1×10^{-3} curie of particulate fission and activation products and approximately 9×10^{-3} curie of tritium. The airborne radioactivity was contained in a total air exhaust volume of approximately 8.5×10^{10} liters. The average radioactivity concentration was well below the applicable standards listed in Reference 3-3. The annual radioactivity concentration at the nearest Windsor Site boundary, allowing for typical diffusion conditions, was less than 0.01 percent of the Department of Energy derived concentration guide for effluent released to unrestricted areas (Reference 3-3). Public radiation exposures from airborne radioactivity are calculated using computer models qualified for this specific task. These models conservatively estimate the radiation exposure to the public through many pathways, including radioactivity in surface soil, vegetation and animal pathways from airborne radioactivity sources. The exposures are calculated using computer models because direct measurements results are indistinguishable from naturally occurring background radioactivity levels.

4.4.5 Existing Nonradiological Conditions - Air Resources

There are no longer any regulated sources of nonradiological pollutant air emissions at the Windsor Site. The principal source of industrial air emissions currently at the Windsor Site is from three liquid propane heating units. Nonradiological pollutant emissions from operation of these propane fired heaters is very low and their operation do not require any regulatory permits.

4.5 Terrestrial Resources

4.5.1 Topography

The Windsor Site property is located in the central Connecticut River Valley. Most areas within two miles of the Windsor Site lie between 150 and 250 feet above sea level. The topography of the Windsor Site is generally flat at an elevation of about 180 feet above sea level.

4.5.2 Geology

The Windsor Site is located within the Central Valley landscape of the Newark Terrain rift basin. Borings taken on and near the Windsor Site identified depths to bedrock ranging from 90 to 145 feet. Bedrock underlying the Windsor Site consists of arkose (sedimentary redbeds) with interlayered basalt and diabase. Successive bedrock formations mapped in the vicinity of the Windsor Site (upper to lower layers) include Portland arkose, the Hampden basalt layer (100 to 150 feet thick), the East Berlin formation (about 500 feet thick), Holyoke basalt (about 300 feet thick), Shuttle Meadow formation (100 to 150 feet thick), Talcott basalt (0 to 150 feet thick), and New Haven arkose (Reference 4-13). The soils at the Windsor Site have been mapped as the Merrimac Series (Reference 4-26). The Merrimac Series formed on deltas and nearly level to undulating glacial stream terraces. This soil has been characterized as a well drained to somewhat excessively drained moderately coarse-textured soil (References 4-26 and 4-27). A typical profile consists of generally brown sandy loam from 0 to 22 inches, yellowish-brown loamy sand from 22 to 26 inches, and various colored coarse sand mixed with 10 to 20 percent fine and medium gravel from 26 to 48 inches. Coarse fragments in the surface soil and subsoil make up 3 to 20 percent of the soil volume. There are no known geologic resources at the Windsor Site having economic value.

4.5.3 Seismology

The Windsor Site is located in a stable geological region with no known active faults. According to the U.S. Geological Survey, a geologic fault known as The Great Fault runs in a generally southwest-to-northeast line across Connecticut about 15 miles east of the Windsor Site (Reference 4-14). A complex system of minor faults exists about 15 miles south of the Windsor Site, between the City of Hartford and Middletown. Some very small faults lie about 2 miles west of the Windsor Site. The faults to the south and east of the Windsor Site generally run southwest to northeast. The faults are very old, dating back about 200 million years or more to the development of the Appalachian Mountains. Many are healed and may be stronger than the original structure. Records dating back to the late 1700s indicate the occurrence of earthquakes capable of damage in the vicinity of the Windsor Site are rare (Reference 4-15).

No voids, either natural or man-excavated, are known to be present in the bedrock beneath the Windsor Site. The Windsor Site is located in an area where there is no hazard of surface faulting, subsidence, solution, uplift, collapse, weathering, landsliding or other hazards resulting from natural causes or from mining activity, petroleum recovering, or ground water extraction.

4.5.4 Existing Radiological Conditions of the Windsor Site and Surrounding Areas

As part of a nationwide program to document baseline conditions surrounding energy-related sites of interest to the Department of Energy, aerial radiological surveys were obtained over Windsor Locks, Connecticut. Figure 4-4 (page 4-22) provides aerial radiological survey results for the area in 1982 that includes the Windsor Site and the adjacent Combustion Engineering, Inc, Site (Reference 4-29, Figure A-1).

4.5.4.1 Existing Radiological Conditions on Windsor Site Property

Radioactive materials attributable to Windsor Site operations have never been disposed of or buried on the Windsor Site property (Reference 2-1). However, small amounts of residual radioactivity remain in localized portions of the Windsor Site discharge system from discharges of water containing low concentrations of radioactivity from 1959 to 1978. The Windsor Site discharge system is located below grade, on the west side of the property.

4.5.4.2 Existing Radiological Conditions in the Surrounding Area Relating to S1C Prototype Operations

Due to discharges of water containing low concentrations of radioactivity between 1959 and 1978 from the Windsor Site to the shallow drainage brook that flows from Goodwin Pond to the Farmington River, small amounts of residual radioactivity are present in the brook sediment. The drainage brook is located on property owned by Combustion Engineering, Inc. and is not readily accessible to the public.

A detailed evaluation of radiological conditions at the Combustion Engineering, Inc. site, including the drainage brook, is being performed under the Department of Energy's Formerly Utilized Sites Remedial Action Program (FUSRAP). The Oak Ridge Institute for Science and Education, under contract with the Formerly Utilized Sites Remedial Action Program, performed investigation sampling at the Combustion Engineering, Inc. site and reported sampling results in Reference 4-28. Reference 4-28 provides the most complete radiological description of the drainage brook currently available. The following discussion incorporates figures and data results contained in Reference 4-28.

The drainage brook measures about 3600 feet long and joins the Farmington River at a location northwest of the Windsor Site. The width of the brook bed varies from approximately 6 to 65 feet. The physical characteristics of the brook sediment bed are not uniform. Sediment depth in the brook varies from 0 feet (no sediment, only rocks) to greater than three feet.

During the third quarter of 1991, an extensive survey of the sediments in the brook was performed. Sediment samples were collected by Knolls Atomic Power Laboratory to study cobalt-60 conditions in the drainage brook. Samples were taken at 108 locations as shown on Figure 4-1 (page 4-12). Samples consisted of the top two inches of sediment at all 108 sampling locations. Deeper samples were taken at locations 6, 11, 24, 63, 71, 81, and 93 and consisted of the top six inches of sediment. Samples at locations 16, 42, and 51 consisted of the top twelve inches of sediment. These twelve-inch deep samples were divided into a top sample and a bottom sample. Naval Reactors provided these 1991 samples to the Oak Ridge Institute for Science and Education to assist in their evaluations of Combustion Engineering, Inc. property (including the drainage brook) adjacent to the Windsor Site.

The Oak Ridge Institute for Science and Education analysis results for these sediment samples are shown in Table 4-1, starting on page 4-13 (Reference 4-28, Table 7). The analyses were performed on dried samples and indicated the presence of cobalt-60 and uranium isotopes above naturally occurring concentrations. The uranium is present in three isotopes: uranium-234, uranium-235, and uranium-238.

The uranium detected in the brook is due to discharges from the Combustion Engineering, Inc. facility adjacent to the Windsor Site and is not attributable to Windsor Site operations. This is clearly shown by the different distributions of uranium and cobalt-60 in the brook. The cobalt-60 is found throughout the entire length of the drainage brook and is found in the highest concentrations close to the Windsor Site outfall and upstream of the Combustion Engineering, Inc. site outfalls into the brook. This is consistent with the cobalt-60 originating from the S1C Prototype reactor plant discharges. The elevated uranium concentrations are found at or downstream of the Combustion Engineering, Inc. site outfalls (and nearby trash piles and a partially buried barrel as discussed later in this section). Uranium concentrations in samples taken upstream of the Combustion Engineering, Inc. site outfalls, but downstream of the Windsor Site outfall, are at natural background levels.

In addition to the clear inference of this physical data, the S1C Prototype reactor plant only handled uranium in the form of high integrity, zirconium alloy clad nuclear fuel. Therefore, there was no dispersible uranium at the S1C Prototype reactor plant which could have been discharged. Combustion Engineering, Inc. on the other hand, manufactured uranium fuel. The Oak Ridge Institute for Science and Education report (Reference 4-28) shows uranium contamination at several locations on the Combustion Engineering, Inc. site and not just at the drainage brook.

Figure 4-2 on page 4-18 shows a map of sampling locations independently selected by the Oak Ridge Institute for Science and Education during their designation survey investigation of the Combustion Engineering, Inc. site (Reference 4-28, Figure 16). Analysis results from these samples are provided in Table 4-2 (Reference 4-28, Table 5). Figure 4-3 on page 4-20 depicts a small area of the Combustion Engineering, Inc. property from which the Oak Ridge Institute for Science and Education collected three soil samples (Reference 4-28, Figure 14). The samples were taken near trash piles and a partially buried barrel located on the drainage brook bank. Analysis results are shown in Table 4-3 (Reference 4-28, Table 6), including one sample result having a total uranium concentration of 24,090 picocuries per gram.

Based on Table 4-1 data, the average cobalt-60 concentration in the drainage brook samples, decay-corrected to the time the samples were collected (September 1991) is approximately 2.2 picocuries per gram. As shown in Table 4-1, the concentrations of cobalt-60 were lower, further from the Windsor Site. Between sampling location 77 and sampling location 108, the average cobalt-60 concentration is less than 0.4 picocuries per gram. Also, the cobalt-60 concentrations are, on average, higher near the top layer of sediment. This conclusion is based on comparison of the deeper six- and twelve-inch deep samples with the two-inch deep samples which were taken at the same locations.

Taking into account the 5.3-year half-life of cobalt-60 and the time which has passed since these samples were obtained (almost five years), the present activity levels would be about one-half the activity levels indicated in Table 4-1. Therefore, the average cobalt-60 concentration present in 1997 is about 1.1 picocuries per gram.

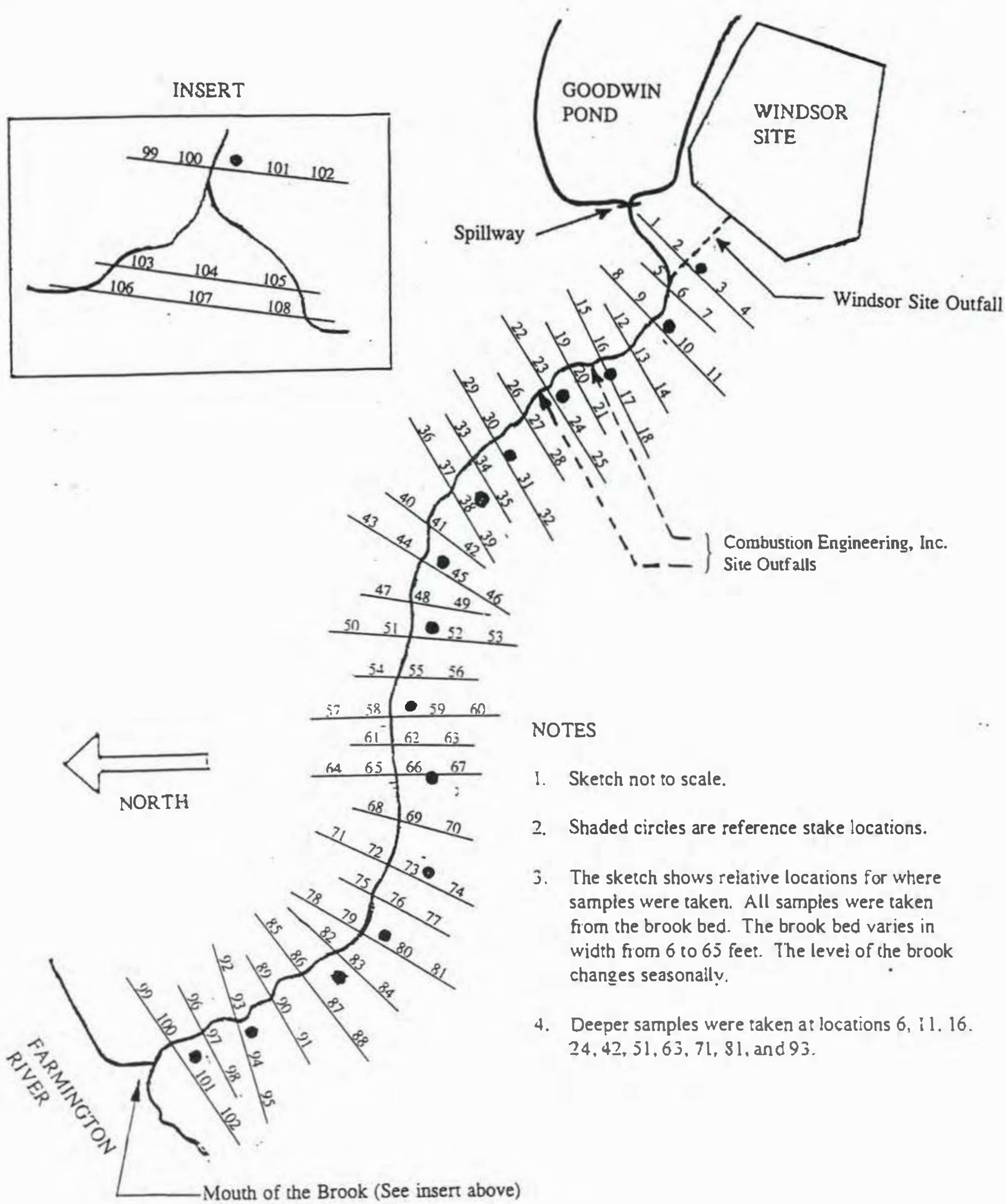
The only other radionuclide from Windsor Site operations that would still be present in the brook sediment is nickel-63 which is a low energy beta emitter and has a half-life of 100 years. Nickel-63 is present in the same insoluble corrosion and wear products as cobalt-60. The Oak Ridge Institute for Science and Education performed nickel-63 analyses on the three samples provided by Naval Reactors which exhibited the highest concentration of cobalt-60. The nickel-63 analysis results were 26.7 ± 1.5 picocuries per gram, 13.2 ± 1.3 picocuries per gram, and 7.2 ± 1.2 picocuries per gram, for samples S7, SC6, and S27, respectively.

The environmental significance of the residual cobalt-60 and nickel-63 in the brook can be evaluated based on a draft report prepared by Argonne National Laboratory and sponsored by the Department of Energy Office of Environmental Restoration (Reference 4-22). The derivation of concentration guideline report is for Combustion Engineering, Inc. property adjacent to the Windsor Site including the drainage brook area. The report describes the derivation of concentration guidelines for both cobalt-60 and nickel-63 which are attributable to Windsor Site operations and uranium isotopes which are not attributable to Windsor Site operations.

Using the conservative assumption that a subsistence farmer moves in and farms the area on top of and immediately surrounding the drainage brook in 1997, an average decay-corrected cobalt-60 concentration of 1.1 picocurie per gram results in a dose of 11 millirem per year to the subsistence farmer. This dose is 3.7% of the dose to an individual from naturally occurring sources. The dose to an industrial worker at a hypothetical industrial facility on top of the brook from a cobalt-60 concentration of 1.1 picocurie per gram is about 3 millirem per year.

The draft concentration guideline for nickel-63 is 3800 times higher than that of cobalt-60. Thus, even though nickel-63 concentration in the brook sediment is higher than the cobalt-60 concentration, the potential dose due to nickel-63 is insignificant.

Additional characterization of the Combustion Engineering, Inc. site adjacent to the Windsor Site will be accomplished as part of the Formerly Utilized Sites Remedial Action Program evaluation of the Combustion Engineering, Inc. Site. Any action taken as a result of the National Environmental Policy Act decision making process for the disposal of the S1C Prototype reactor plant would not affect future evaluation of the drainage brook or any remedial action on the Combustion Engineering, Inc. Site.



NOTES

1. Sketch not to scale.
2. Shaded circles are reference stake locations.
3. The sketch shows relative locations for where samples were taken. All samples were taken from the brook bed. The brook bed varies in width from 6 to 65 feet. The level of the brook changes seasonally.
4. Deeper samples were taken at locations 6, 11, 16, 24, 42, 51, 63, 71, 81, and 93.

Figure 4-1 Drainage Brook Sediment Sample Locations

Table 4-1 Radionuclide Concentrations in the Drainage Brook Sediment Samples Collected by KAPL and Analyzed by the Oak Ridge Institute for Science and Education

KAPL Id[d, e]	ORISE Sample ID	Radionuclide Concentration (pCi/g)					% U-235 Enrichment
		U-235	U-238	Total U[a]	Co-60[b] (1173 keV)	Co-60[b] (1332 keV)	
S1	390S117	0.00	0.73 ± 0.76[c]	0.7	0.11	0.00	0.0
S2	390S001	0.00	0.37 ± 0.78	0.4	0.16	0.00	0.0
S3	390S002	0.00	0.63 ± 0.78	0.6	0.17 ± 0.11	0.17 ± 0.11	0.0
S4	390S003	0.00	0.35 ± 0.86	0.4	0.25	0.00	0.0
S5	390S004	0.00	0.80 ± 0.71	0.8	0.37 ± 0.16	0.49 ± 0.14	0.0
S6	390S005	0.00	1.07 ± 1.74	1.1	5.78 ± 0.73	4.77 ± 0.72	0.0
SC6	390S116	0.00	6.19 ± 1.99	6.2	46.67 ± 1.92	46.90 ± 1.88	0.0
S7	390S006	0.00	0.71 ± 2.15	0.7	29.63 ± 1.53	28.81 ± 1.48	0.0
S8	390S007	0.00	1.33 ± 1.92	1.3	5.33 ± 0.68	5.92 ± 0.69	0.0
S9	390S008	0.00	3.72 ± 2.79	3.7	13.43 ± 1.26	13.02 ± 1.17	0.0
S11	390S009	0.30 ± 0.12	4.36 ± 1.53	11.8	1.81 ± 0.31	1.84 ± 0.33	1.04
SC11	390S102	0.00	0.92 ± 0.92	0.9	0.25	0.00	0.0
S12	390S010	0.00	2.61 ± 1.26	2.6	2.23 ± 0.43	2.44 ± 0.39	0.0
S14	390S011	0.00	1.89 ± 1.29	1.9	1.39 ± 0.33	0.87 ± 0.30	0.0
S15	390S012	0.00	2.79 ± 1.73	2.8	5.09 ± 0.82	5.24 ± 0.70	0.0
S16	390S013	0.00	2.76 ± 1.03	2.8	0.88 ± 0.22	1.02 ± 0.18	0.0
SC16B	390S103	0.72 ± 0.10	11.99 ± 1.37	30.0	0.54 ± 0.31	0.74 ± 0.21	0.92
SC16T	390S104	0.00	5.62 ± 1.54	5.6	2.10 ± 0.45	1.69 ± 0.37	0.0
S17	390S014	0.95 ± 0.22	5.67 ± 2.53	29.4	4.11 ± 0.76	4.44 ± 0.81	2.54
S18	390S015	0.25 ± 0.11	2.97 ± 1.47	9.3	8.70 ± 0.69	7.98 ± 0.68	1.32
S19	390S016	4.97 ± 0.29	15.33 ± 2.61	139.6	13.95 ± 1.12	13.12 ± 1.08	4.80
S20	390S017	0.62 ± 0.08	2.16 ± 1.04	17.7	0.29 ± 0.25	0.12 ± 0.12	4.28
S21	390S018	0.79 ± 0.16	7.63 ± 2.07	27.4	1.78 ± 0.47	1.97 ± 0.56	1.59
S22	390S019	1.76 ± 0.32	11.05 ± 2.73	55.2	2.69 ± 0.74	2.49 ± 0.77	2.42
S23	390S020	2.69 ± 0.18	11.33 ± 1.98	78.5	17.88 ± 0.95	16.36 ± 0.97	3.56
SC24	390S105	0.49 ± 0.07	2.25 ± 0.78	14.5	0.20 ± 0.22	0.41 ± 0.16	3.29
S24	390S021	1.80 ± 0.14	7.54 ± 1.59	52.5	1.76 ± 0.37	1.92 ± 0.36	3.58
S25	390S022	1.45 ± 0.09	1.49 ± 0.82	37.8	0.45 ± 0.18	0.41 ± 0.17	13.14

Footnotes located on page 4-17

Table 4-1 Radionuclide Concentrations in the Drainage Brook Sediment Samples Collected by KAPL and Analyzed by the Oak Ridge Institute for Science and Education, (continued)

KAPL Id[d, e]	ORISE Sample ID	Radionuclide Concentration (pCi/g)					% U-235 Enrichment
		U-235	U-238	Total U[a]	Co-60[b] (1173 keV)	Co-60[b] (1332 keV)	
S26	390S023	7.13 ± 0.50	24.39 ± 4.26	202.7	5.74 ± 1.09	7.35 ± 1.20	4.35
S27	390S118	12.93 ± 0.54	11.82 ± 3.76	335.1	34.38 ± 2.08	33.91 ± 1.98	14.53
S28	390S119	16.48 ± 0.23	1.29 ± 1.40	413.3	0.73 ± 0.15	0.59 ± 0.17	66.12
S29	390S024	4.17 ± 0.19	6.67 ± 1.56	110.9	3.03 ± 0.40	2.36 ± 0.40	8.86
S30	390S025	4.42 ± 0.24	6.77 ± 1.75	117.2	2.44 ± 0.42	2.53 ± 0.39	9.22
S31	390S026	1.02 ± 0.06	0.65 ± 0.62	26.1	0.26 ± 0.12	0.15 ± 0.08	19.63
S32	390S027	14.26 ± 0.60	18.62 ± 4.05	375.1	5.13 ± 0.96	5.47 ± 0.94	10.64
S33	390S120	27.34 ± 0.62	5.36 ± 2.64	688.9	4.66 ± 0.82	4.90 ± 0.67	44.11
S34	390S028	5.04 ± 0.19	2.61 ± 1.22	128.6	0.74 ± 0.31	0.40 ± 0.18	23.04
S35	390S029	7.53 ± 0.33	16.94 ± 2.75	205.2	4.89 ± 0.73	4.97 ± 0.71	6.47
S36	390S030	2.05 ± 0.16	3.69 ± 1.10	54.9	1.28 ± 0.35	1.07 ± 0.27	7.95
S37	390S031	2.44 ± 0.14	3.19 ± 1.02	64.2	0.72 ± 0.25	0.70 ± 0.21	10.61
S38	390S032	4.22 ± 0.26	7.93 ± 1.91	113.3	2.07 ± 0.42	2.30 ± 0.42	7.64
S39	390S033	0.56 ± 0.08	1.76 ± 1.16	15.7	0.24	0.00	4.69
S40	390S034	3.87 ± 0.20	1.34 ± 1.17	7.0	0.29	0.00	2.59
S41	390S035	3.87 ± 0.20	4.01 ± 1.51	100.7	1.93 ± 0.46	2.09 ± 0.39	13.04
S42	390S036	0.96 ± 0.07	2.38 ± 0.97	26.4	0.35 ± 0.16	0.32 ± 0.13	5.90
SC42B	390S106	0.00	0.88 ± 0.62	0.9	0.10	0.00	0.0
SC42T	390S107	0.48 ± 0.08	2.01 ± 0.98	14.0	0.23	0.00	3.57
S43	390S037	8.72 ± 0.41	16.27 ± 3.28	234.3	5.15 ± 0.87	5.17 ± 0.77	7.69
S44	390S038	7.53 ± 0.28	12.64 ± 2.23	200.8	3.47 ± 0.56	3.53 ± 0.46	8.47
S45	390S039	0.31 ± 0.06	0.56 ± 0.71	8.3	0.17	0.00	7.94
S46	390S040	0.51 ± 0.09	1.67 ± 1.10	14.3	0.37 ± 0.21	0.37 ± 0.15	4.51
S47	390S041	0.82 ± 0.06	1.62 ± 0.69	22.2	0.44 ± 0.16	0.38 ± 0.13	7.32
S48	390S042	0.80 ± 0.07	1.11 ± 0.69	21.1	0.73 ± 0.18	0.59 ± 0.17	10.10
S49	390S043	0.36 ± 0.07	1.12 ± 0.93	10.2	0.29	0.00	4.81
S50	390S044	0.24 ± 0.08	1.58 ± 0.75	7.5	0.23	0.00	2.30
S51	390S045	0.21 ± 0.05	1.03 ± 0.82	6.2	0.17	0.00	3.06

Footnotes located on page 4-17

Table 4-1 Radionuclide Concentrations in the Drainage Brook Sediment Samples Collected by KAPL and Analyzed by the Oak Ridge Institute for Science and Education, (continued)

KAPL Id[d, e]	ORISE Sample ID	Radionuclide Concentration (pCi/g)					% U-235 Enrichment
		U-235	U-238	Total U[a]	Co-60[b] (1173 keV)	Co-60[b] (1332 keV)	
SC51B	390S108	0.00	2.57 ± 0.61	2.6	0.09	0.00	0.0
SC51T	390S109	0.00	1.89 ± 0.66	1.9	0.11	0.00	0.0
S52	390S046	5.41 ± 0.26	10.81 ± 2.07	146.1	1.85 ± 0.42	2.12 ± 0.39	7.22
S53	390S047	3.12 ± 0.26	5.18 ± 2.28	83.2	2.01 ± 0.54	1.60 ± 0.42	8.56
S54	390S048	2.85 ± 0.16	1.68 ± 1.14	72.9	1.40 ± 0.29	1.16 ± 0.31	20.85
S55	390S049	0.00	0.67 ± 0.83	0.7	0.30	0.00	0.0
S56	390S050	1.11 ± 0.10	1.73 ± 0.97	29.6	0.63 ± 0.18	0.42 ± 0.25	9.09
S57	390S051	0.85 ± 0.08	1.01 ± 0.93	22.3	0.55 ± 0.22	0.40 ± 0.17	11.55
S58	390S052	0.43 ± 0.06	1.38 ± 0.77	12.1	0.25 ± 0.14	0.46 ± 0.17	4.62
S59	390S053	0.48 ± 0.06	1.42 ± 1.01	13.3	0.24 ± 0.19	0.26 ± 0.15	4.97
S60	390S054	0.26 ± 0.06	0.71 ± 0.67	7.2	0.16	0.00	5.37
S61	390S055	0.41 ± 0.07	1.19 ± 0.97	11.5	0.23 ± 0.15	0.30 ± 0.10	5.11
S62	390S056	0.37 ± 0.07	0.92 ± 0.75	10.1	0.28	0.00	5.89
S63	390S057	0.67 ± 0.08	1.43 ± 1.06	18.3	0.43 ± 0.18	0.41 ± 0.18	6.81
SC63	390S110	0.50 ± 0.07	1.72 ± 0.93	14.2	0.23 ± 0.16	0.20 ± 0.12	4.31
S64	390S058	0.00	0.69 ± 0.77	0.7	0.20	0.00	0.0
S65	390S059	0.28 ± 0.03	0.70 ± 0.46	7.6	0.18 ± 0.07	0.11 ± 0.07	5.77
S66	390S060	0.42 ± 0.04	0.81 ± 0.50	11.3	0.24 ± 0.09	0.20 ± 0.07	7.48
S67	390S061	0.42 ± 0.06	1.18 ± 0.83	11.6	0.17	0.00	5.21
S68	390S062	1.29 ± 0.12	1.44 ± 1.14	33.6	0.69 ± 0.24	0.56 ± 0.19	12.21
S69	390S063	0.44 ± 0.08	1.70 ± 1.00	12.7	0.28	0.00	3.88
S70	390S064	6.56 ± 0.26	9.94 ± 2.05	173.8	4.50 ± 0.52	4.20 ± 0.64	9.30
S71	390S065	0.14 ± 0.03	0.79 ± 0.40	4.4	0.11	0.00	2.73
SC71	390S111	0.00	1.20 ± 0.56	1.2	0.07	0.00	0.0
S72	390S066	0.45 ± 0.04	1.00 ± 0.53	12.3	0.37 ± 0.09	0.28 ± 0.10	6.55
S73	390S067	0.26 ± 0.04	0.78 ± 0.42	7.3	0.18 ± 0.08	0.14 ± 0.06	4.99
S74	390S068	0.57 ± 0.08	0.65 ± 0.82	15.0	0.30	0.00	12.10
S75	390S121	40.07 ± 0.55	7.36 ± 2.59	1009.1	1.71 ± 0.36	1.56 ± 0.35	45.69

Footnotes located on page 4-17

Table 4-1 Radionuclide Concentrations in the Drainage Brook Sediment Samples Collected by KAPL and Analyzed by the Oak Ridge Institute for Science and Education, (continued)

KAPL Id[d, e]	ORISE Sample ID	Radionuclide Concentration (pCi/g)					% U-235 Enrichment
		U-235	U-238	Total U[a]	Co-60[b] (1173 keV)	Co-60[b] (1332 keV)	
S76	390S069	1.03 ± 0.11	1.95 ± 1.25	27.6	0.77 ± 0.26	0.75 ± 0.23	7.56
S77	390S070	0.25 ± 0.08	2.15 ± 1.23	8.4	0.29	0.00	1.76
S78	390S071	0.22 ± 0.06	1.48 ± 0.88	7.0	0.19	0.00	2.26
S79	390S072	0.00	1.26	0.0	0.31	0.00	0.0
S80	390S073	0.23 ± 0.05	1.40 ± 0.69	7.1	0.15 ± 0.15	0.20 ± 0.13	2.49
S81	390S074	0.21 ± 0.07	0.67 ± 1.03	5.8	0.36 ± 0.13	0.31 ± 0.17	4.57
SC81	390S112	0.19 ± 0.05	1.33 ± 0.97	6.0	0.16	0.00	2.14
S82	390S075	0.76 ± 0.09	1.64 ± 0.76	20.6	0.39 ± 0.19	0.25 ± 0.14	6.71
S83	390S076	0.00	0.65 ± 0.85	0.7	0.19	0.00	0.0
SC83	390S113	0.36 ± 0.05	1.22 ± 0.62	10.3	0.24 ± 0.11	0.15 ± 0.09	4.40
S84	390S077	0.12 ± 0.03	0.58 ± 0.37	3.6	0.08	0.00	3.13
S85	390S078	1.78 ± 0.10	1.86 ± 0.77	46.3	0.74 ± 0.19	0.66 ± 0.17	12.97
S86	390S079	0.00	0.32 ± 0.94	0.3	0.24	0.00	0.0
S87	390S080	0.00	0.70 ± 0.51	0.7	0.11	0.00	0.0
S88	390S081	0.20 ± 0.04	1.30 ± 0.70	6.3	0.10	0.00	2.35
S89	390S082	0.23 ± 0.06	1.44 ± 0.97	7.2	0.21 ± 0.11	0.12 ± 0.12	2.41
S90	390S083	0.21 ± 0.04	0.88 ± 0.84	6.2	0.23 ± 0.12	0.11 ± 0.13	3.62
S91	390S084	0.32 ± 0.03	0.80 ± 0.44	8.9	0.15 ± 0.09	0.15 ± 0.06	5.88
S92	390S085	0.46 ± 0.05	0.64 ± 0.58	12.0	0.14	0.00	9.94
S93	390S086	0.25 ± 0.05	1.52 ± 0.84	7.8	0.18 ± 0.12	0.10 ± 0.10	2.52
SC93	390S114	0.31 ± 0.04	1.08 ± 0.62	8.8	0.10	0.00	4.27
S94	390S087	0.00	1.14 ± 0.91	1.1	0.21	0.00	0.0
S95	390S088	0.00	0.70 ± 0.96	0.7	0.22	0.00	0.0
S96	390S089	0.13 ± 0.05	1.23 ± 0.64	4.4	0.14	0.00	1.58
S97	390S090	0.00	0.78 ± 0.85	0.8	0.21	0.00	0.0
S98	390S091	0.24 ± 0.05	1.47 ± 0.86	7.6	0.20 ± 0.12	0.17 ± 0.10	2.51
SC98	390S115	0.23 ± 0.05	1.56 ± 0.91	7.4	0.13	0.00	2.26
S99	390S092	0.00	2.04 ± 1.57	2.0	0.28	0.00	0.0

Footnotes located on page 4-17

Table 4-1 Radionuclide Concentrations in the Drainage Brook Sediment Samples Collected by KAPL and Analyzed by the Oak Ridge Institute for Science and Education, (continued)

KAPL Id[d, e]	ORISE Sample ID	Radionuclide Concentration (pCi/g)					% U-235 Enrichment
		U-235	U-238	Total U[a]	Co-60[b] (1173 keV)	Co-60[b] (1332 keV)	
S100	390S093	0.00	1.47 ± 0.69	1.5	0.15	0.00	0.0
S101	390S094	0.00	1.11 ± 0.75	1.1	0.10	0.00	0.0
S102	390S095	0.00	0.52 ± 0.74	0.5	0.13	0.00	0.0
S103	390S096	0.56 ± 0.10	1.72 ± 1.17	15.6	0.34 ± 0.21	0.28 ± 0.18	4.79
S104	390S097	0.00	1.46 ± 1.14	1.5	0.19	0.00	0.0
S105	390S098	4.34 ± 0.30	6.75 ± 2.46	115.2	3.05 ± 0.69	2.81 ± 0.65	9.09
S106	390S099	2.10 ± 0.14	3.22 ± 1.20	55.6	2.08 ± 0.34	1.77 ± 0.31	9.20
S107	390S100	0.15 ± 0.06	0.28 ± 0.77	4.0	0.22	0.00	7.73
S108	390S101	1.48 ± 0.13	1.96 ± 1.21	38.9	0.88 ± 0.27	0.85 ± 0.23	10.50

[a] Total uranium calculated by multiplying U-235 concentration by 25 (to account for U-234 concentration) and adding U-238 concentration.

[b] All samples were decay-corrected to sample collection date (9/91).

[c] Uncertainties represent the 95% confidence level, based only on counting statistics.

[d] Refer to Figure 4-1.

[e] Samples consisted of the top two inches of sediment. Some deeper samples were collected and are labeled SC. The deeper samples were either six or twelve inches deep. The twelve inch samples were split into a top sample and a bottom sample.

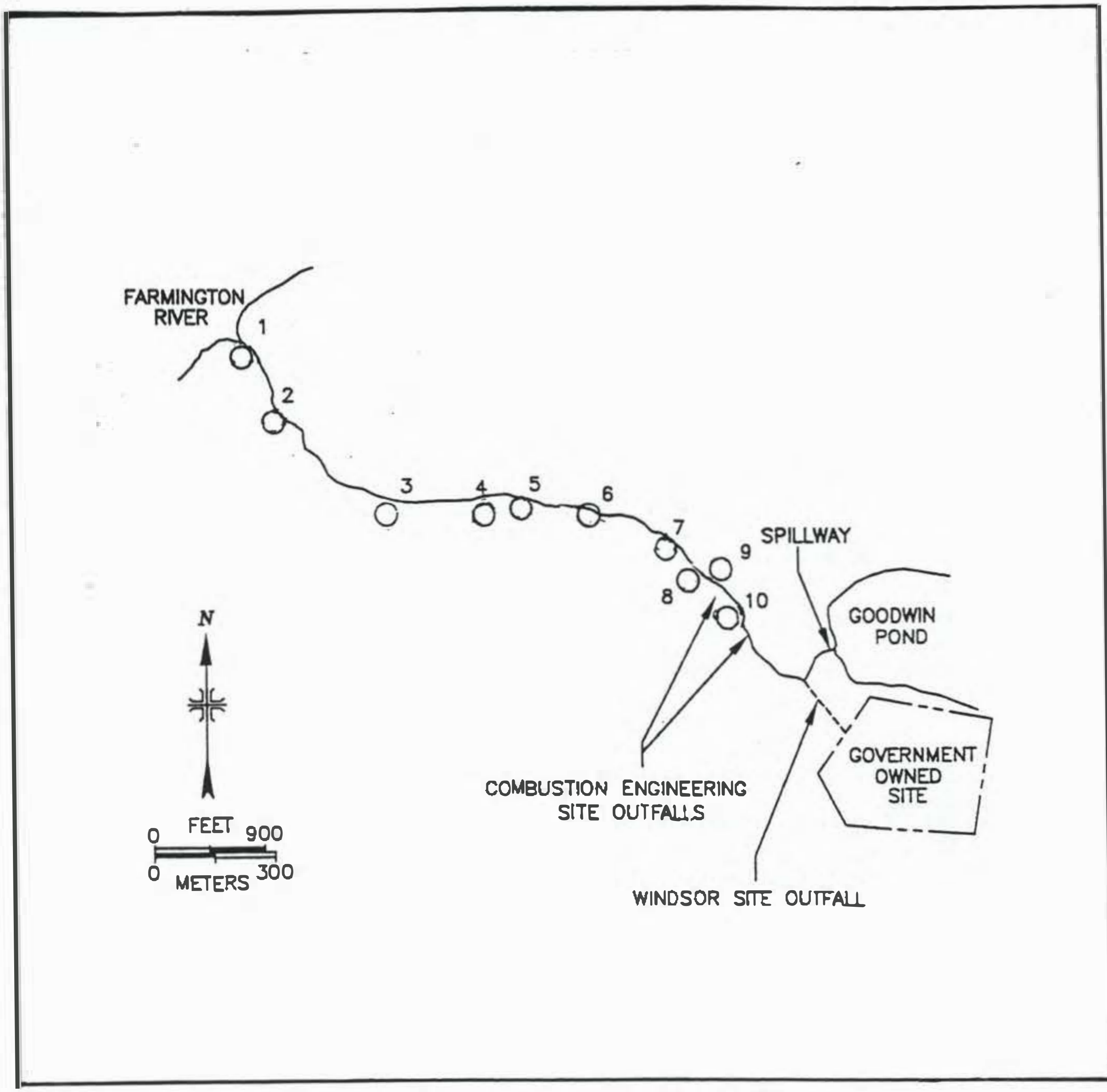


Figure 4-2 Drainage Brook - Oak Ridge Institute for Science and Education Sampled Locations

Table 4-2 Uranium Concentrations in Drainage Brook Sediment Samples Collected by the Oak Ridge Institute for Science and Education

Location [b]	Uranium Concentrations (pCi/g)			% U-235 Enrichment
	U-235	U-238	Total U [a]	
1	<0.1	<1.0	3.5	1.5
2	<0.1	1.2±1.1[c]	3.7	1.3
3	<0.1	1.3±0.8	3.8	1.2
4	0.1±0.1	0.5±0.7	3.0	3.0
5	10.9±0.6	11.3±4.8	280	13
6	1.5±0.1	3.4±1.2	41	6.4
7	16.7±1.0	21±10	440	11
9	2.3±0.2	8.6±2.9	66	4.0
10	1.0±0.1	2.0±1.7	27	7.2

[a]Total uranium concentration based on assumed U-234 to U-235 ratio of 24.
[b]Refer to Figure 4-2. The results for Location #8 are shown in Table 4-3 (page 4-21).
[c]Uncertainties represent the 95% confidence level, based only on counting statistics.

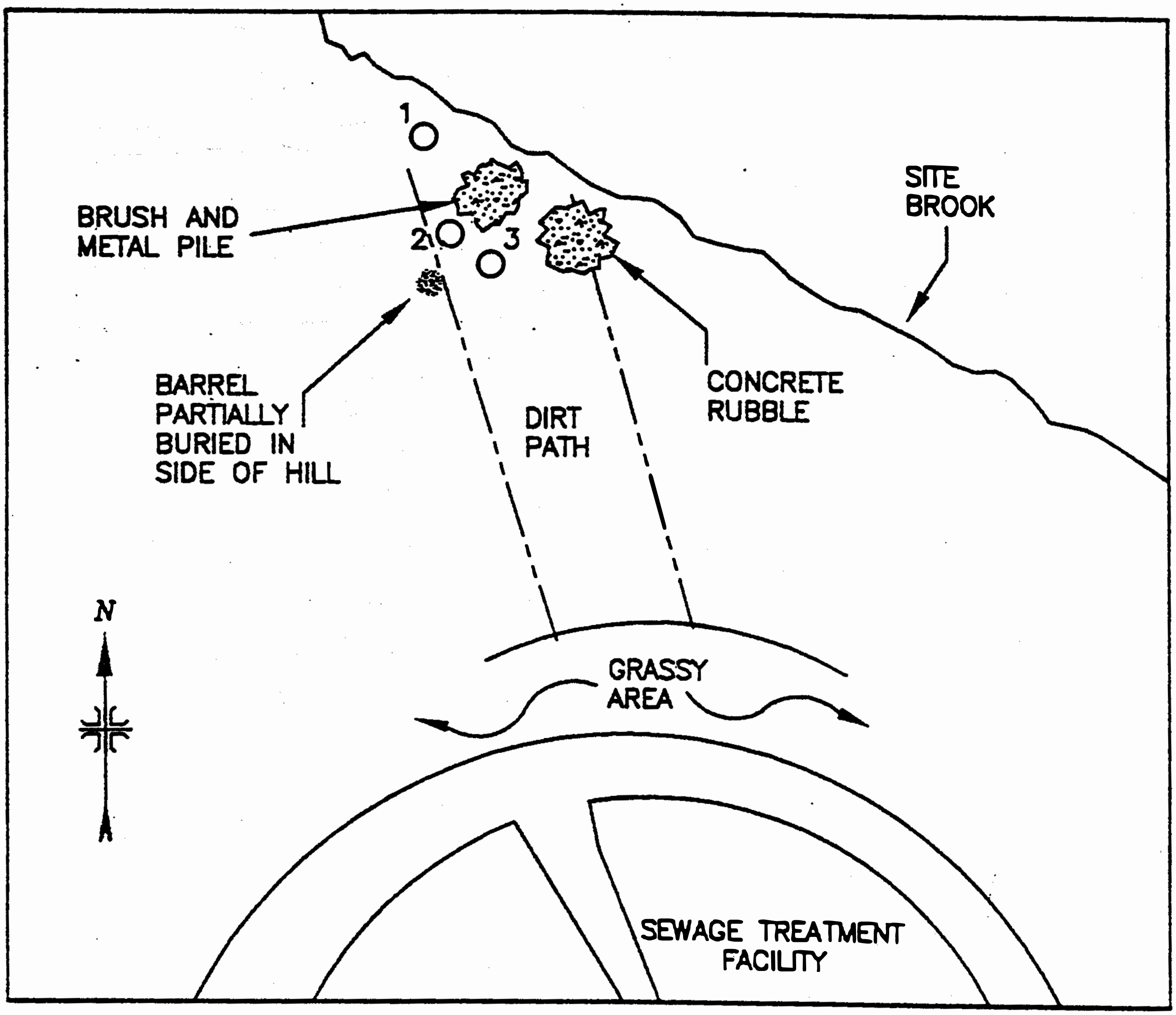


Figure 4-3 Trash Piles on Drainage Brook Bank - Oak Ridge Institute for Science and Education Measurement and Sampling Locations

Table 4-3 Isotopic Uranium Concentrations in Drainage Brook Bank Samples Collected by the Oak Ridge Institute for Science and Education

Location	Figure No.	Uranium Concentration (pCi/g)				%U-235 Enrichment
		U-234	U-235	U-238	Total U[a]	
Site Brook Bank #1	4-3	929±74	37±10	0.9±1.5	967±75	86
Site Brook Bank #2, 0-15 cm	4-3	15,450±320	4,860±200	3,780±160	24,090±410	17
Site Brook Bank #3	4-3	387±33	22.0±3.5	9.1±1.9	418±33	27
Site Brook #8	4-2	16,160±370	525±75	59±22	16,740±380	58

[a]Total uranium concentrations based on the sum of U-234, U-235 and U-238 concentrations.
[b]Uncertainties represent the 95% confidence level, based only on counting statistics.

Map labels have been added and features enhanced for clarity in this Environmental Impact Statement

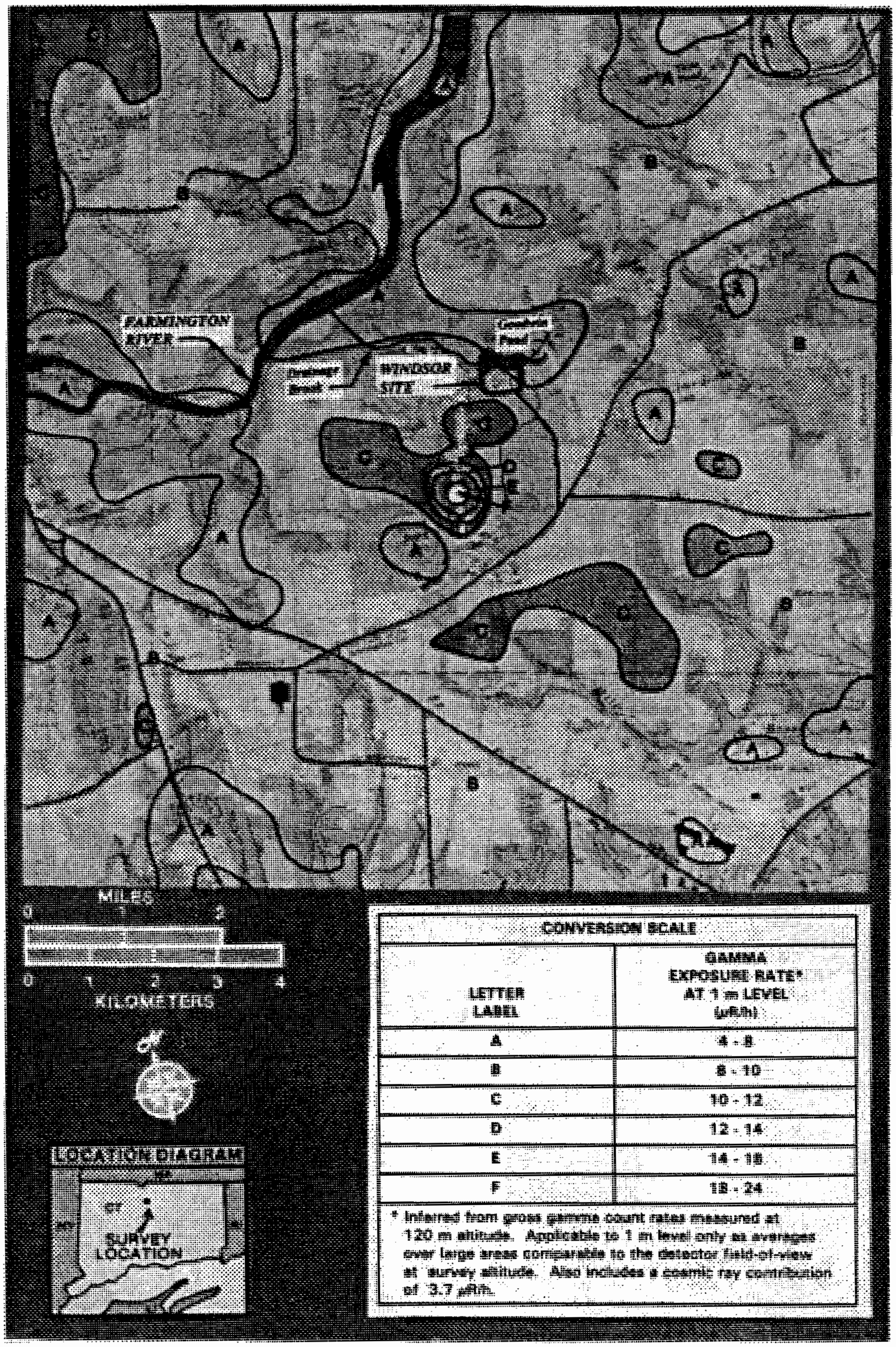


Figure 4-4 Exposure Rate Contour Map for the June 1982 Aerial Survey of the Windsor Locks, Connecticut Area. The elevated levels, D and above, show man-made radiation levels over the Combustion Engineering, Inc. facilities.

4.5.5 Existing Nonradiological Conditions of the Windsor Site and Surrounding Areas

4.5.5.1 Existing Nonradiological Conditions on Windsor Site Property

Only small amounts of chemicals have been disposed of at the Windsor Site. Windsor Site practices have conformed with established rules applicable at the time. These practices included disposal of small amounts of laboratory acids and oxidizers and minute amounts of nonhazardous laboratory analysis chemicals in the Windsor Site septic system. Small amounts of dilute battery acid were disposed of in a dry well and inadvertently into the septic system. At no other time has the Windsor Site been used to bury or otherwise dispose of laboratory or chemical wastes.

As part of ongoing Windsor Site activities, a voluntary facility assessment has been initiated to support Windsor Site inactivation and future release of the property. As a part of the voluntary facility assessment, a work plan for sampling the Windsor Site was developed and provided to the U.S. Environmental Protection Agency and to the State of Connecticut Department of Environmental Protection.

The work plan for sampling was prepared based on interviews with personnel involved in Windsor Site operations over its history, detailed record searches, review of construction drawings, review of historical environmental sampling, hydrogeologic information, published technical literature, and Environmental Protection Agency Resource Conservation and Recovery Act investigation guidance documents. The sampling plan involves investigation of areas within and adjacent to the Windsor Site property to confirm that no significant contamination has occurred resulting from Windsor Site operations. Soil, surface water, ground water and sediment samples will be collected. Samples will be analyzed for specific chemicals of concern based on Windsor Site operating history.

Following completion of all field work, a report will be prepared and provided to the regulatory agencies. The report will summarize findings and will identify the need for any additional investigation or cleanup required to support the goal of unrestricted release of the Windsor Site. Naval Reactors will meet with both regulatory agencies to review report findings and to obtain regulatory agency perspective on achieving the goal of unrestricted release of the Windsor Site.

4.5.5.2 Existing Nonradiological Conditions in the Surrounding Area Relating to S1C Prototype Operations

As discussed in Section 4.3.4, prior to 1980, chromate compounds were added to the Windsor Site cooling water system to inhibit corrosion and biological growth. Samples of Windsor Site discharges, the drainage brook and the Farmington River were taken in 1978. Sample analysis results showed that elevated levels of chromium were present in the brook sediment but indicated that the chromate-containing water discharges from the Windsor Site were not significantly impacting the environment of the Farmington River. Chromium analysis results of twenty sediment samples taken from the brook ranged from 11 to 70 parts per million. Two samples were also taken from Goodwin Pond sediment upstream of the Windsor Site discharge point. Chromium analysis results of the two Goodwin Pond sediment samples were 1.7 and 2.0 parts per million. Subsequent to 1978, a United States Geological Survey reported that chromium levels in soils and other surficial materials in the State of Connecticut range from 30 to 50 parts per million (Reference 4-25).

Although there are no United States Environmental Protection Agency or Connecticut Department of Environmental Protection standards that are directly applicable to sediments, the maximum chromium level detected in the brook sediment (70 parts per million) is less than the most conservative Connecticut Department of Environmental Protection chromium standard (100 parts per million) for direct exposure to residential soils (Regulations of Connecticut State Agencies Section 22a-133k-1 through 22a-133k-3, Remediation Standard, effective January 1996).

The voluntary facility assessment in process includes further investigation of chromium conditions in soils at the Windsor Site and immediately surrounding areas. Surface soil and sediments from the brook and Goodwin Pond will be collected and analyzed for chromium to assess the significance of the 1978 data and the potential for other sources besides the Windsor Site cooling water system.

4.6 Socioeconomics

The population distribution within a 50-mile radius of the Windsor Site, compiled from 1990 Census data, is shown on Figure 4-5. Table 4-4 summarizes the population distribution.

Table 4-4: Population Distribution Within a 50-Mile Radius of the Windsor Site

Miles	People	Cumulative People
0 to 5	56,429	56,429
5 to 10	286,341	342,770
10 to 20	868,651	1,211,421
20 to 30	717,683	1,929,104
30 to 40	532,391	2,461,495
40 to 50	963,795	3,425,290

Table 4-5 presents socioeconomic factors for the Capital Region of north-central Connecticut and for the Town of Windsor based on 1990 Census data. The State of Connecticut Capital Region is made up of 29 towns, including the Town of Windsor.

Table 4-5: Socioeconomic Factors for the Town of Windsor and the Capital Region

	Town of Windsor	Capital Region
Population	27,817	709,404
Civilian Labor Force	15,767	387,360
Percent Non-White [a] Population	21 [d]	18
Percent Hispanic Population	3	8
Average Household Income	$50,228	$49,630 [b]
Percent Living In Poverty	2	8 [c]

a. Includes "Black", "Asian or Pacific Islander", "American Indian or Alaska Native."
b. Average for 29 towns comprising the Capital Region.
c. For Hartford County.
d. According to the Town of Windsor, neighborhoods in the northeastern and southern portions of Windsor had 25 to 30% non-white populations while most other neighborhoods, including the area around the Windsor Site, had 3 to 6% (Reference 4-16).

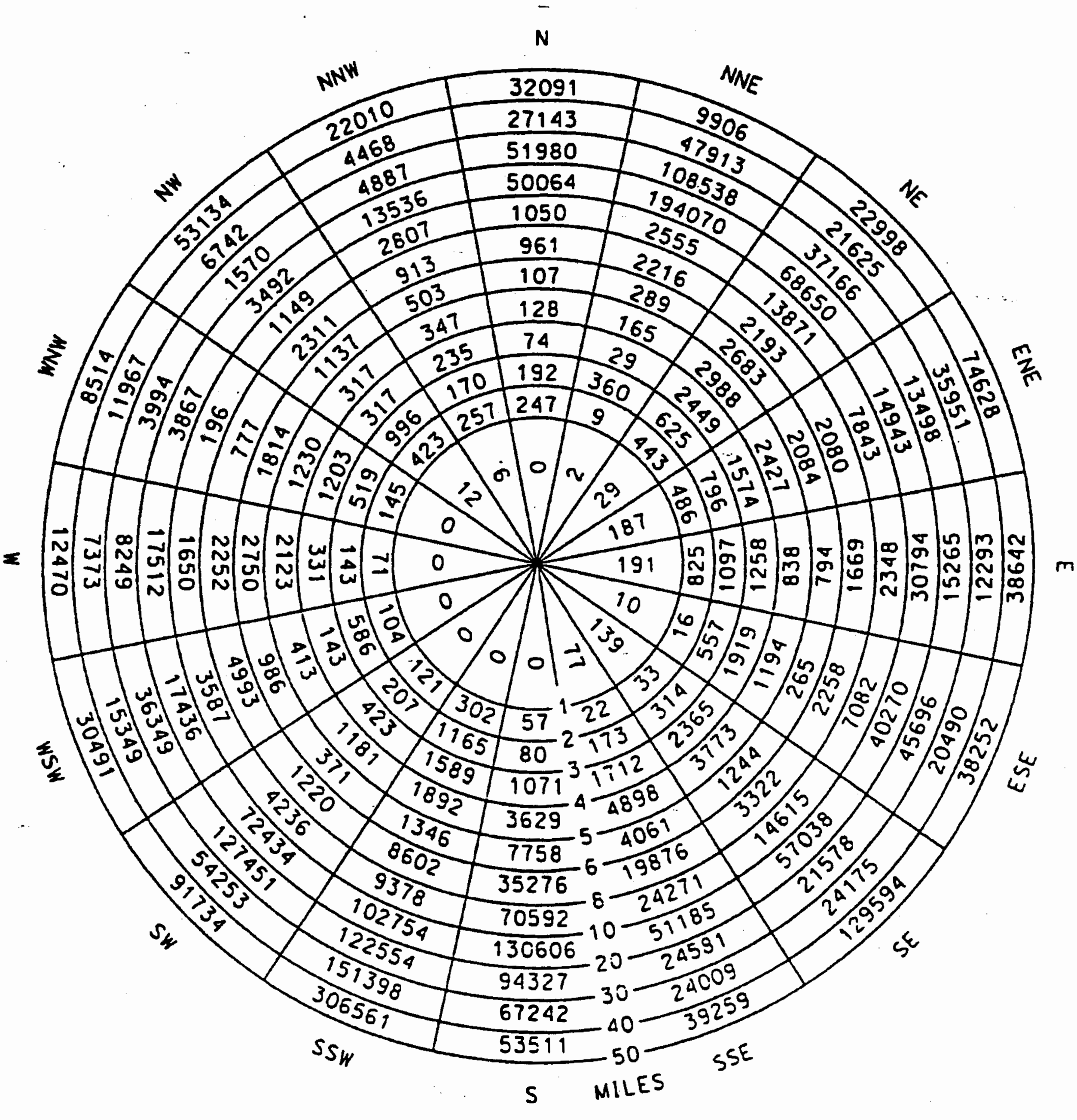


Figure 4-5: 1990 Population Distribution Within a 50-Mile Radius of the Windsor Site

According to the Town of Windsor Plan of Development, nonagricultural employment in the area totals about 19,500 jobs, divided between manufacturing (39%) and nonmanufacturing (61%) (Reference 4-16). The majority of the manufacturing jobs involve fabricating metals, aircraft and machinery. The majority of the nonmanufacturing jobs involve wholesale trade, retail, financial, insurance, real estate, services and government. Based on Reference 4-16, the diversity of jobs helps to mitigate the impacts of heavy layoffs when any one sector suffers economic setbacks. In 1989, Windsor had a surplus of approximately 2,800 jobs. This surplus was an important factor contributing to Windsor's extremely low unemployment rate of about 4% town wide and about 2% for minority groups. For comparison, overall unemployment in Hartford was about 9%. The Windsor Site currently employs about 150 personnel.

4.7 Cultural Resources

According to the Connecticut Historical Commission (Reference 4-17) and the Connecticut Office of State Archaeology (Reference 4-18), there are no known areas or items of archaeological, historical or cultural significance located at or immediately adjacent to the Windsor Site that predate Windsor Site construction in 1957. There are no Native American rights or interests associated with the Windsor Site property.

A Memorandum of Agreement has been executed between the Department of Energy and the State of Connecticut. The Memorandum of Agreement identifies measures that will be carried out to address issues pertaining to the historical significance of the Windsor Site and the S1C Prototype reactor plant. The Memorandum of Agreement, a copy of which is included in Appendix E of this Environmental Impact Statement, has been accepted by the Advisory Council on Historic Preservation.

4.8 Noise, Aesthetic and Scenic Resources

The Windsor Site property is offset from Day Hill Road (formerly Prospect Hill Road) approximately one-half mile and is not visible from public roadways. The land surrounding the Windsor Site property is mostly commercially-owned woodlands. Thus, the public is not exposed to noise generated by Windsor Site activities, typically equivalent to light industrial activity.

4.9 Traffic and Transportation

Two major interstate highway corridors are located in close proximity to the Windsor Site. Approximately two miles east of the Windsor Site, Interstate 91 serves urban traffic north and south of the Hartford metropolitan area. Approximately eight miles south of the Windsor Site, Interstate 84 serves urban traffic east and west of Hartford. Secondary roads bounding the Windsor Site are used for local residential traffic, commuting and delivery routes by a variety of businesses. Secondary roads include Day Hill Road (approximately one-half mile south of the Windsor Site) and State Routes 187 and 189 (approximately two miles west of the Windsor Site). Traffic associated with Windsor Site activities does not contribute noticeably to overall traffic conditions in the area.

There are two rail lines that pass through the Town of Windsor. The New Haven/Springfield line is located approximately five miles east of the Windsor Site, running north-south, and passing through Hartford. The New Haven/Springfield line provides routine passenger and freight service. The Griffin Line is located approximately 1.5 miles west of the Windsor Site. The Griffin Line is a branch line, approximately eight miles in length, that runs in a north-south direction. The Griffin Line connects to the New Haven/Springfield line in the City of Hartford. The Griffin Line was abandoned in place by commercial railroad companies about ten years ago and except for very infrequent use by Naval Reactors for Windsor Site related shipments, the Griffin Line has been unused during the past ten years. The Griffin Line is currently under the purview of the Connecticut Department of Transportation.

Commercial barge traffic occurs on the Connecticut River up to the city of Hartford. Docking facilities are available in Hartford, and full, seagoing facilities exist on Long Island Sound.

Bradley International Airport, approximately three miles north-northeast of the Windsor Site, is the nearest airport with scheduled flights by commercial jet aircraft. Flights from Bradley International Airport also include air cargo, corporate, private, and military aircraft (Connecticut and Army Air National Guard). The Windsor Site is located within the five nautical mile radius air traffic control boundaries used for aircraft approaches and departures. Regular large and small aircraft flight patterns, including designated Federal Aviation Administration (FAA) instrument approaches, have the potential for flying over the Windsor Site. Simsbury Airport, approximately three miles northwest of the Windsor Site, is a small airport used only by light private aircraft. Other small airports in the greater Hartford area are located more than five miles from the Windsor Site.

There are no public roads, highways, railways, or navigable waterways on the Windsor Site property.

4.10 Nonradiological Occupational Hazards

Naval Reactors policy is to maintain a healthful work environment at all its facilities, and utilize Occupational Safety and Health Administration standards where appropriate for all Windsor Site activities. Engineered systems, administrative controls, and employee training are the primary means employed for minimizing potential employee exposure to occupational hazards. If hazards cannot be controlled with engineering or administrative controls, personal protective equipment is used to provide additional protection.

Impact for workplace hazards other than radiation are measured by reportable injury, illness, and fatality rates in the work force. Injury, illness, and fatality rates for construction (demolition) workers are considered separately because of the more hazardous nature of their work. Table 4-6 provides the reportable injury, illness, and fatality rates for the Department of Energy and its contractors. Reportable injury and illness rates related to Naval Reactors work have been consistently lower than the rates reported by private industry and the Department of Energy. For the purposes of this evaluation, overall Department of Energy statistics provide a more representative baseline for dismantlement work. The average rates for private industry in the United States are also provided for comparison.

According to the U.S. Department of Labor, injuries in the workplace are most likely to be sprains and strains, bruises and contusions, cuts and lacerations, and fractures. Injuries are most likely to occur from contact with equipment and other objects, falls, and overexertion. Generally, fatalities in the workplace (non-violence related) are most likely to result from contact with equipment and other objects, falls, and exposure to harmful substances or environments (Reference 4-19).

Table 4-6: Average Occupational Injury, Illness and Fatality Rates

	All labor categories		**Construction workers**	
	Total injuries and illnesses per worker-year	Fatalities per worker-year	Total injuries and illnesses per worker-year	Fatalities per worker-year
Department of Energy and contractors [a]	3.6×10^{-2}	3.0×10^{-5}	6.6×10^{-2}	1.0×10^{-4}
Private industry [b]	8.9×10^{-2}	5.8×10^{-5}	1.2×10^{-1}	2.2×10^{-4}

a. 1989-1993 averages (Reference 4-20).
b. 1990-1994 averages (Reference 4-21).

4.11 Utilities and Energy

Windsor Site electricity is supplied by Connecticut Light and Power. Since March 1994, the Windsor Site water system is supplied from the Town of Windsor municipal water supply (Metropolitan District Commission). Utility usage by the Windsor Site during 1995 averaged about 7,000 kilowatt-hours per day for electricity and about 34,000 gallons per day for water. Monthly fuel use at the Windsor Site during 1995 averaged 120 gallons of gasoline, 130 gallons of diesel fuel, and 480 gallons of liquid propane.

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CHAPTER 5
ENVIRONMENTAL CONSEQUENCES

5.0 ENVIRONMENTAL CONSEQUENCES

This chapter summarizes a wide variety of potential environmental consequences associated with the Prompt Dismantlement, Deferred Dismantlement, and No Action alternatives for S1C Prototype reactor plant disposal. The environmental consequences are determined by comparing calculated impacts (such as hypothetical health risks) to the baseline environmental conditions described in Chapter 4. Detailed analyses of potential impacts on worker and public health are described in Appendix B for facility activities and Appendix C for transportation of materials off-site. In addition, Appendices B and C discuss potential consequences and risks of various accident scenarios. This chapter provides a brief description of analysis methodology, results, and conclusions. A basic, overall understanding of the environmental consequences can be gained without reading Appendices B and C, however, those appendices are cited numerous times to assist the reader in finding additional information on specific topics.

To further assist the reader and decision makers, this chapter is organized by alternatives. All environmental topics of concern are discussed within a section devoted to each alternative.

Hypothetical health effects are expressed in terms of latent fatal cancer risks. The most significant potential health effect from environmental and occupational radiation exposures is the inducement of latent fatal cancers. This effect is referred to as latent because the cancer may take many years to develop. The health risk conversion factors used in this document are taken from the International Commission on Radiological Protection which specifies 0.0005 latent fatal cancers per person-rem of exposure to the public and 0.0004 latent fatal cancers per person-rem for workers (Reference 5-1). These risk estimates were extrapolated from estimates applicable to high doses and dose rates and probably overstate the true lifetime risk at low doses and dose rates. In an assessment of this uncertainty, the National Academy of Sciences pointed out that "the possibility that there may be no risks from exposures comparable to external natural background radiation cannot be ruled out" (Reference 5-3). Appendix A provides additional information on common sources of radiation and health effects.

Detailed analyses discussed in Appendices B and C support the conclusion that public exposure resulting from any of the reasonable alternatives for disposal of the S1C Prototype reactor plant would be negligible.

5.1 Prompt Dismantlement Alternative

This alternative would dismantle the reactor plant promptly, dispose of waste, and recycle materials. Dismantling the reactor plant would be done by removing components individually. Section 3.1 provides a detailed description of the prompt dismantlement alternative. Environmental impacts are discussed below.

5.1.1 Land Use

Prompt dismantlement of the defueled S1C Prototype reactor plant would have no adverse effect on land use. Prompt dismantlement would actually have a positive impact on land use by making the Windsor Site property available for other uses as soon as practical. Prompt dismantlement could allow for the unrestricted release of the Windsor Site as early as 2001.

Dismantlement activities would be confined within the Windsor Site property boundary which is an already developed area. There would be no impact on present or planned use of the surrounding areas. Materials resulting from dismantlement activities would be recycled as much as practical and any wastes would be disposed of off-site at licensed disposal facilities. No land on-site and no additional land off-site would have to be set aside for waste disposal.

5.1.2 Ecological Resources

Prompt dismantlement of the defueled S1C Prototype reactor plant would not impact ecological resources at the Windsor Site or surrounding areas. Since the Windsor Site property is small and mostly developed, ecological resources are essentially nonexistent. There are no woodlands, wetlands, or other significant biological habitats at the property, so there would be no habitat loss due to dismantlement activities. Plant or animal species sensitive to disturbance by human activities are not expected to be present and have not been observed at the Windsor Site. There are no known populations of Federal or State designated endangered, threatened or special concern species existing at or in the vicinity of the Windsor Site (Reference 4-4). Fish populations in Goodwin Pond and the Farmington River, and small game and deer in the surrounding area would be unaffected.

5.1.3 Water Resources

The Windsor Site is not located in a 100 or 500-year floodplain area. Dismantlement activities would not involve major earth moving work. Consequently, floodplains in the vicinity of the Windsor Site would not be affected by prompt dismantlement activities.

PROMPT DISMANTLEMENT ALTERNATIVE

5.1.3.1 Radiological Consequences for Water Resources

Prompt dismantlement activities would not involve any discharges of radioactive liquid effluents. Ground water under the Windsor Site and surface water in the surrounding environment would not be affected.

5.1.3.2 Nonradiological Consequences for Water Resources

During prompt dismantlement, liquid discharges from the Windsor Site would be limited to Storm water runoff. Approximate flow of this drainage was incorporated in the State of Connecticut Department of Environmental Protection General Storm water Permit application. These effluents would continue to be monitored and results would continue to be reported annually through completion of this alternative. In the event that excavation activities disturb more than five acres of land, a State of Connecticut Department of Environmental Protection General Permit for the Discharge of Stormwater and Dewatering Wastewater from Industrial Activities will also be obtained. Effluent from the sanitary sewer would continue to be treated in the anaerobic septic system and released below ground through the Windsor Site leach field. Ground water under the Windsor Site and surface water in the surrounding environment would not be affected.

5.1.4 Air Resources

5.1.4.1 Radiological Consequences for Air Resources

Dismantlement operations on radiologically contaminated piping and components would be performed using environmental protection measures to minimize the emission of particulate radioactivity to air as discussed in Section 3.1.1. The resulting airborne particulate radioactivity emissions associated with incident-free prompt dismantlement activities were evaluated. The details of the analyses are provided in Appendix B, Section B.2. Analyses for this alternative assumed an airborne particulate radioactivity source term which was derived from radiation levels measured on high efficiency particulate air filters in reactor servicing ventilation systems used during past maintenance activities. The high efficiency particulate air filters have a greater than 99.95% efficiency for removal of airborne particulate radioactivity. It is conservatively estimated that 7.7×10^{-6} curies per year would be discharged during prompt dismantlement (1.5×10^{-5} total curies for the two-year duration of the alternative). As discussed in Chapter 4, Section 4.4.4, and reported in Reference 4-10, the radioactivity contained in exhaust air from calendar year 1994 Windsor Site activities totaled less than 1×10^{-3} curies of particulate fission and activation products and had no environmental impact. Therefore, prompt dismantlement activities would have no significant radiological consequences on air resources.

PROMPT DISMANTLEMENT ALTERNATIVE

5.1.4.2 Nonradiological Consequences for Air Resources

Environmental impacts on air resources from nonradiological emissions were evaluated for several sources, including Windsor Site heating furnaces, airborne dust, and vehicle emissions. There are currently no regulated point sources of nonradiological industrial gaseous emissions at the Windsor Site. Windsor Site heating would be provided by several small, liquid propane fueled, forced hot air furnaces.

Prompt dismantlement activities would include cutting, handling and removal of systems and structures. The presence of materials such as asbestos insulation, lead-based paint, and lead shielding introduce the potential for minor emissions of criteria and hazardous air pollutants from these operations. Since detailed plans have not yet been developed, specific analyses of these emissions were not performed. However, such emissions would be controlled in accordance with applicable State and Federal regulations. Furthermore, these emissions would be transitory and would not be expected to result in the classification of the Windsor Site under the Clean Air Act as a major source of air pollutants based on Program experience. Consequently, no discernible effect on air resources is expected.

Facility demolition and miscellaneous earth moving work could affect air quality through emission of pollutants from diesel and gasoline powered equipment and from the spread of dust. As described in Appendix B, Section B.5, the spread of dust was analyzed using a computer model. The analysis focused on the maximally exposed off-site individual located 100 meters from the center of the dismantlement work area. The calculated dust concentration for the maximally exposed off-site individual during Windsor Site restoration activities was 1.7 milligrams per cubic meter. When this airborne concentration is compared to a Threshold Limit Value - Time Weighted Average concentration for inhalable particulates (10 milligram per cubic meter), it is concluded that dust emissions associated with dismantlement activities would not result in any adverse effects. To reduce the generation of dust, control measures such as using an appropriate level of water spray would be used. Pollutants from diesel and gasoline powered equipment and vehicles would be immediately diluted in the air with no discernible effect on-site or off-site.

Nonradiological consequences of vehicle emissions from transport of dismantlement wastes and recyclable materials off-site is discussed in Section 5.1.10.2. The overall discharge of nonradiological air pollutants from prompt dismantlement activities would be very small and would not have a discernible effect on air resources.

5.1.5 Terrestrial Resources

Prompt dismantlement of the defueled S1C Prototype reactor plant would not adversely affect terrestrial resources at the Windsor Site or surrounding areas. Excavation work in support of reactor plant dismantlement activities would be confined within the fenced security boundary of the Windsor Site. Excavation work would be shallow (limited in depth to a few feet) and would not affect the geological character of the Windsor Site.

PROMPT DISMANTLEMENT ALTERNATIVE

As discussed in Section 3.1.4, following prompt dismantlement of the S1C Prototype reactor plant, shipment of recyclable materials, and waste disposal, actions would be taken to remove all unnecessary buildings, systems, and paved areas from the Windsor Site property. Excavation activities required to remove underground systems would be accomplished in small, limited areas and would not have a permanent affect on the terrestrial resources. After completion of Windsor Site dismantlement activities, the land contour would be restored to natural, nominally flat conditions to support natural reforestation.

5.1.5.1 Expected Final Radiological Conditions of the Windsor Site Property After Prompt Dismantlement

As discussed in Section 3.1.4, after removal of all radioactive material from the Windsor Site, actions would be taken to release the Windsor Site property for future unrestricted use. In other Naval Nuclear Propulsion Program Sites which have been released for unrestricted use, a screening limit of 1 picocurie per gram of soil for cobalt-60 was used to confirm that final conditions in soil were acceptable. Typical soils often contain more than 10 picocuries per gram of naturally occurring radionuclides.

The extent of soil remediation, if any, is expected to be small and has been included in the estimation of radioactive waste to be shipped from the Windsor Site discussed in Section 5.1.13. Soil within or immediately adjacent to the Windsor Site boundary that exceeds applicable radioactive guideline values would be removed. A final radiological survey of the Windsor Site would be performed to confirm radioactivity levels in soils are below release criteria for future unrestricted uses of the property. The action of confirming that applicable release criteria are met ensures that any future occupant at the Windsor Site would receive less radiation exposure than limits specified in Department of Energy Order 5400.5 (Reference 3-3) as well as the draft regulations under consideration by the Nuclear Regulatory Commission and the Environmental Protection Agency. Draft regulations include a maximum exposure limit of 15 millirem per year from all sources of which a maximum of 4 millirem per year can be from ingestion of radioactivity in water. The final radiological survey would be conducted following a comprehensive strategy that measures radiation levels at the ground surface and takes systematic soil samples for analysis. Appendix G provides details on the final radiological survey of the Windsor Site, including the timing for performing the surveys. Final survey results would be documented and reported to appropriate Federal and State regulatory agencies. Federal and State regulators would be invited to comment on these reports and perform verification surveying and sampling.

5.1.5.2 Expected Final Nonradiological Conditions of the Windsor Site Property After Prompt Dismantlement

As discussed in Section 3.1.4, all Windsor Site systems would be completely removed including all systems that are located below grade. Buildings, paved areas, and Windsor Site security fencing would be removed. Soil within or adjacent to the Windsor Site boundary that exceeds any applicable cleanup standards would be removed to support unrestricted release of the property. The extent of soil remediation, if required, is expected to be very small.

PROMPT DISMANTLEMENT ALTERNATIVE

As discussed in Section 4.5.5, a voluntary facility assessment has been initiated to support Windsor Site inactivation and future release of the property. Following completion of all sample collecting and analytical work, a report will be prepared and provided to the U.S. Environmental Protection Agency, Region I and the State of Connecticut Department of Environmental Protection. The report will summarize findings and will provide recommendations for any additional investigation or cleanup required to support the goal of unrestricted release of the Windsor Site. Naval Reactors will meet with both regulatory agencies to review report findings.

5.1.6 Socioeconomics

Current Windsor Site staffing is about 150 personnel. Windsor Site staffing through a reactor plant dismantlement period of approximately 2 years is estimated to remain at 150 construction (demolition) and support personnel. Based on current staffing demographics, an estimated 75% of the Windsor Site labor force would be made up of Connecticut state capital region residents, and the rest would commute from longer distances. Reduction in the Windsor Site work force to zero at the end of the dismantlement period would not significantly affect overall unemployment levels in the Windsor area or greater Hartford region.

Since the Windsor Site is currently owned by the U.S. Government, the Site is not taxable. Under the prompt dismantlement alternative, the Windsor Site could be transferred to a taxpaying entity soon after prompt dismantlement is complete. However, considering the small size of the Windsor Site, the impact on the tax base of the town is not expected to be significant. Consequently, prompt dismantlement would not have any discernible socioeconomic impact.

5.1.7 Cultural Resources

Prompt dismantlement of the S1C Prototype reactor plant would not impact any cultural resources predating Windsor Site construction. Measures that will be carried out by the Department of Energy to address the effects of dismantlement activities on the Windsor Site and the S1C prototype reactor plant are identified in the Memorandum of Agreement with the State of Connecticut, as approved by the Advisory Council on Historic Preservation. A copy of this Memorandum of Agreement is included in Appendix E of this Environmental Impact Statement.

5.1.8 Noise, Aesthetic and Scenic Resources

Prompt dismantlement of the S1C Prototype reactor plant would not have noticeable noise, aesthetic or scenic impacts. The Windsor Site is an existing industrial zoned area characterized by noise from truck and automobile traffic, and operating industrial equipment such as diesel-powered engines, air-operated jackhammers, and other similar equipment. Reactor plant dismantlement activities would not result in an increase in ambient noise levels in occupied areas surrounding the Windsor Site above pre-dismantlement levels.

PROMPT DISMANTLEMENT ALTERNATIVE

Since wooded areas surrounding the Windsor Site block any view from the nearest public roadways, dismantlement activities would have no impact on aesthetic and scenic resources in the vicinity of the Windsor Site. Compared to the existing developed conditions at the property, removal of the S1C Prototype reactor plant and various other buildings, structures and pavement would have a positive environmental effect on visual characteristics.

5.1.9 Traffic and Transportation

Prompt dismantlement of the defueled S1C Prototype reactor plant and Windsor Site restoration activities would not have a noticeable impact on regional and local traffic. Traffic related to dismantlement would include commuting personnel, equipment mobilization, waste shipments, and deliveries of fill and topsoil. On average, there would be about three shipments arriving and departing daily. Transport of the pressure vessel (and possibly the primary shield tank) from the Windsor Site to the Griffin Line industrial track railhead, approximately 1.5 miles west of the Windsor Site, would affect traffic on the western portion of Day Hill Road (formerly Prospect Hill Road) for a short period during the day the shipment leaves the site. The transport of such shipments by heavy hauler would be planned for a time that minimizes the impact. Highway shipments of packages of similar size to the reactor pressure vessel package have occurred between the Windsor Site and the Griffin Line industrial track railhead in the past. Based on past experience for these shipments, local police escorts have allowed traffic to pass to reduce congestion.

5.1.10 Occupational and Public Health and Safety

This section summarizes analysis results for expected incident-free conditions during prompt dismantlement. Section 5.1.12 summarizes analysis results for potential accident conditions during prompt dismantlement and transport of materials.

5.1.10.1 Incident-Free Facility Activities

5.1.10.1.1 Incident-Free Facility Activities - Radiological Consequences

The radiological health risks associated with incident-free facility activities during prompt dismantlement are evaluated in Appendix B, Section B.2. Effects from assumed airborne particulate radioactivity releases and direct radiation exposure were assessed for the worker, maximally exposed off-site individual and the general population. Gamma radiation from cobalt-60 contained within the reactor plant systems is the primary source of direct radiation exposure. For the workers, analyses were based on data from detailed radiation surveys of the S1C Prototype reactor compartment, worker staffing levels, and time in or near the reactor compartment. For the general population, analyses were based on the cumulative exposure to all members of the general population living within a 50-mile radius of the Windsor Site and historical radiation data at the Windsor Site boundary.

PROMPT DISMANTLEMENT ALTERNATIVE

Appendix B, Section B.2 discusses assumptions used in calculations for airborne particulate radioactivity releases. Airborne particulate radioactivity can cause exposures via several different pathways. The air resource impact calculations include the external exposure from the ground and surface water deposition (from fallout of airborne radioactivity), air immersion pathways and internal exposure through the ingestion and inhalation pathways.

The combined health risks for direct radiation exposure and radioactive contamination to the air are summarized in Table B-6. It is estimated that the radiation workers would receive between 94 to 188 person-rem (3.8×10^{-2} to 7.5×10^{-2} additional latent fatal cancer risk) during prompt dismantlement. The larger value of the range represents an estimate based on preliminary plans. The lower value of the range reflects experience that detailed work planning typically results in additional exposure reductions. The annual occupational radiation exposure from prompt dismantlement would be comparable in magnitude to the radiation exposure routinely received during operation and maintenance of Naval nuclear reactor plants. Individual worker exposure would be limited to 2 rem per year even though Federal limits allow exposure up to 5 rem per year. Under the prompt dismantlement alternative, the general population would receive an estimated total of 8.1×10^{-3} person-rem (4.0×10^{-6} additional latent fatal cancer risk) from radiation exposure during prompt dismantlement.

5.1.10.1.2 Incident-Free Facility Activities - Nonradiological Consequences

Naval Reactors policy is to maintain a safe and healthful environment at all facilities, including the Windsor Site. Work practices are designed to minimize exposure to physical and chemical hazards. Employees are routinely monitored during work for exposure to such hazards and, when appropriate, are placed into medical surveillance programs. Dismantlement evolutions requiring the use of specialized equipment or the handling of hazardous materials would only be performed by trained personnel. Personnel exposure to hazardous materials would be maintained within Occupational Safety and Health Administration limits through the use of engineered controls, protective clothing, air supplied respirators, containment tents and filtered ventilation. These controls would also ensure protection of the environment within applicable limits. Nonradiological emissions would be controlled in accordance with applicable State and Federal regulations. Nonradiological effects from facility demolition and miscellaneous earth moving work would be negligible.

PROMPT DISMANTLEMENT ALTERNATIVE

5.1.10.2 Incident-Free Transportation Analyses

Appendix C transportation evaluations assumed all shipments originate at the Windsor Site located in Windsor, Connecticut. The analyses assumed that there would be 1,100 shipments of nonradioactive waste and recyclable materials from dismantlement and demolition activities and 500 incoming shipments of materials such as fill and topsoil. These shipments would occur between the Windsor Site and facilities located an average of 200 kilometers from the Site. Analyses assumed that 23 radioactive material shipments would be made from the Windsor Site consisting of 19 shipments of miscellaneous waste packages and 4 individually packaged major components. As discussed in Section 3.1.3, the reactor pressure vessel would be shipped by heavy haul truck to the Griffin Line industrial railhead and the rest of the trip to the disposal site would be made by rail. The other 22 shipments would be made by truck. In addition to the pressure vessel, one additional shipment by railroad may be necessary in order to ship the primary shield tank in a single large package. In the transportation analyses, two Department of Energy destinations were analyzed for shipments of low-level radioactive materials: the Savannah River disposal site in the State of South Carolina and the Hanford disposal site in the State of Washington. The analyses included additional general assumptions to keep the meaning of the results simple and conservative. For example, the Savannah River disposal site and the Hanford disposal site were examined individually as the destination for all radioactive shipments. The Savannah River disposal site represents a reasonable close location and distance for transportation analyses, and the Hanford disposal site represents a reasonable but significantly more distant location. Combinations of shipping destinations, including available recycling facility locations for radioactive materials, are not examined. This is a conservative simplification because the cumulative mileage of any combination of available destinations would be less than the cumulative mileage of all shipments going cross-country to the Hanford disposal site. Actual disposal of dismantlement materials would utilize multiple shipping destinations with emphasis on recycling as much material as practical. The topic of waste management and recycling is discussed in more detail in Section 5.1.13.

5.1.10.2.1 Incident-Free Transportation - Radiological Consequences

Gamma radiation emanating from cobalt-60 contained within reactor components is the primary source of direct radiation exposure from the low-level radioactive recyclable material and waste shipments. All low-level radioactive recyclable material and waste shipments would be packaged to meet Department of Transportation standards for packaging integrity and dose rate limits.

The potential radiological health risks associated with incident-free transportation of reactor plant components were evaluated using the RADTRAN 4 computer code. Health effects were assessed for the general population, transportation crew, hypothetical maximally exposed individuals in the general population and the maximally exposed individual in the transportation crew. As discussed in Appendix C, Section C.2, a conservative simplification was made in the transportation analyses which assumed that all radioactive recyclable material or waste would be shipped to the same location, either the Savannah River disposal site in South Carolina or the Hanford disposal site in Washington State. Details for the technical approach for assessing incident-free radioactive shipments are provided in Appendix C, Section C.4. Computer model variables and assumptions are provided in Appendix C, Section C.5.

The health risks due to low-level radioactive material shipments from the Windsor Site to the Savannah River disposal site for prompt dismantlement are summarized in Table C-12. Analyses indicate that the general population would receive 1.93 person-rem (9.66×10^{-4} additional latent fatal cancer risk) from shipment of low-level radioactive materials from the Windsor Site to the Savannah River Disposal Site. Transportation workers would receive 6.66 person-rem (2.67×10^{-3} additional latent fatal cancer risk) for the same shipments.

The health risks for shipments of the same packages to the Hanford disposal site are summarized in Table C-13 and are slightly higher due to the greater distance traveled. However, the radiological health risks are still very small. For shipment of low-level radiological materials from the Windsor Site to the Hanford disposal site, analyses indicate the general population would receive 5.11 person-rem (2.55×10^{-3} additional latent fatal cancer risk) and the transportation crew would receive 10.3 person-rem (4.11×10^{-3} additional latent fatal cancer risk).

These results represent conservative estimates of the radiological consequences of incident-free transportation, and are higher than past experience shows for typical Naval Reactors waste shipments.

5.1.10.2.2 Incident-Free Transportation - Nonradiological Consequences

The nonradiological health risks associated with incident-free transportation of waste and recyclable materials, fill and topsoil were evaluated based on methods developed at Sandia National Laboratory. Nonradiological health risks for incident-free transportation would result from vehicle exhaust emissions (air pollutants). Health effects were assessed for the general population. All material shipments were evaluated. The radiological shipment evaluations considered shipment to Savannah River and the Hanford disposal sites. The nonradiological shipment evaluations assumed an average transportation distance of 200 kilometers since the final destination for waste and recyclable materials and the points of origin for fill and topsoil shipments could vary depending on the haulers.

Incident-free transportation analyses are discussed in detail in Appendix C, Section C.3. The nonradiological health risks (due primarily to vehicle exhaust emissions) are presented in Tables C-3, C-12, and C-13. Adding the nonradiological-related health risks for all waste shipments from the Windsor Site results in a 1.8×10^{-2} additional fatality risk to the general population. This risk is small and is approximately the same as the radiological risks discussed in Section 5.1.10.2.1.

5.1.11 Utilities and Energy

Prompt dismantlement of the defueled S1C Prototype reactor plant would not result in a large demand on utilities and energy resources. Dismantlement activities would require quantities of fuel, water, and electricity typical of small to medium sized construction or demolition projects. The amount of utilities and energy expected to be consumed would not result in any discernible environmental consequences.

5.1.12 Occupational and Transportation Accidents

Hypothetical accident scenarios were evaluated to estimate the potential for, and effects of, release of radioactive material and toxic chemicals. Appendix B, Section B.3 provides details of hypothetical facility accidents resulting in the release of radioactive materials to the environment. Appendix B, Section B.4 provides analysis of a nonradiological fuel fire. Appendix C, Section C.6 describes the technical approach for assessing radioactive shipment accidents. The results of these analyses are presented in terms of latent fatal cancer risks to facility workers and the public.

5.1.12.1 Facility Accidents

5.1.12.1.1 Radiological Consequences of Facility Accidents

Several hypothetical accident scenarios that would result in uncontrolled release of radioactivity to the environment were evaluated to determine the long-term health risks. The hypothetical releases of airborne radioactivity and exposure to radiation during accident scenarios were assessed for the worker, maximally exposed off-site individual and the general population.

As described in Appendix B, Section B.1.2, accidents were considered if they were expected to contribute substantially to risk. Risk is defined as the product of the probability of occurrence times the consequence of the accident. The four hypothetical accident scenarios evaluated for the dismantlement activities included 1) a large component drop, 2) mechanical damage of a component due to a wind-driven missile, 3) an airplane crash into the reactor plant with damage to several components, 4) and a high efficiency particulate air filter fire. Variables considered in the analyses include source terms, population density, meteorological conditions, affected area and pathways for exposure to radiation (such as external direct exposure and internal exposure from inhalation).

PROMPT DISMANTLEMENT ALTERNATIVE

The details of the analysis are provided in Appendix B, Section B.3. As shown in Table B-16, the accident with the greatest risk for dismantlement activities is an airplane crashing into the reactor plant. The annual risk of a member to the general population developing a latent fatal cancer due to an airplane crash accident during prompt dismantlement is 3.8×10^{-7}. This risk is the product of the probability of the accident occurring (6.6×10^{-7} per year) times the consequence of the accident (0.58 latent fatal cancers). Over the two-year duration of prompt dismantlement, the cumulative risk from an airplane crash accident is 7.6×10^{-7}. This is extremely small compared to other incident-free radiological impacts due to the low probability of an airplane crashing directly into the S1C Prototype reactor plant.

5.1.12.1.2 Nonradiological Consequences of Facility Accidents

For the purposes of comparison with other risks associated with dismantlement and caretaking activities, an evaluation of a diesel fuel oil fire was analyzed in detail in Appendix B, Section B.4. This accident was selected based on the potential duration of the accident, the potential size of the affected area, and the combustion products that would result. A hypothetical accident scenario involving a fire in the Windsor Site's hazardous waste container storage area was considered but eliminated from detailed analysis. The quantity of hazardous wastes that would be stored at any time during dismantlement is expected to be maintained small by routine disposal shipments. The Windsor Site's hazardous waste container storage area is constructed and operated such that the overall environmental risks, including risks from accidents, are insignificant.

The airborne concentrations of the combustion products resulting from the fire were evaluated with respect to the maximally exposed off-site individual. The toxic chemicals that would be generated during the fire due to combustion are carbon monoxide, oxides of nitrogen (90% nitric oxide and 10% nitrogen dioxide), lead and sulfur dioxide. In the event of an accidental fire, Windsor Site safety procedures would be immediately followed to protect the workers and the public.

Nonradiological occupational accidents, such as slips and falls, are expected to occur during the dismantlement activities; however, the rate is not expected to be greater than rates for other construction activities (provided in Table 4-6). Projections of the number of fatalities, injuries or illnesses during prompt dismantlement were calculated based on Table 4-6 Department of Energy rates and are summarized in Table 5-1 for the prompt dismantlement alternative. These results indicate that the overall nonradiological occupational risks are small.

PROMPT DISMANTLEMENT ALTERNATIVE

Table 5-1: Estimated Nonradiological, Occupational Impacts for Prompt Dismantlement

Estimated Windsor Site Staffing Level	150
Estimated average injuries/illnesses per year [a]	9.9
Estimated fatalities per year [a]	1.5×10^{-2}
Total estimated injuries/illnesses [b]	19.8
Total estimated number of fatalities [b]	3.0×10^{-2}

a. Calculated by multiplying Windsor Site staffing level times the Department of Energy rates for construction workers provided in Table 4-6.

b. Total values calculated for a two-year duration of prompt dismantlement.

5.1.12.2 Transportation Accidents

There has never been a major accident nor measurable release of radioactivity to the environment during shipment of Naval Reactors program waste or materials, however, hypothetical accidents were evaluated to determine potential environmental effects.

5.1.12.2.1 Radiological Consequences of Transportation Accidents

Appendix C, Section C.6 provides the technical approach used for assessing hypothetical radioactive shipment accidents. Health effects were assessed for the general population and the hypothetical maximally exposed individual. Analyses assumed that the transportation workers would evacuate the scene of an accident within a relatively short time after the accident occurred. Therefore, the risks of transportation accidents on transportation workers are included in the results for the general population.

Radiological health risks from uncontrolled releases of radioactivity to the environment and direct radiation exposure from damaged packages were evaluated using the RADTRAN 4 and RISKIND computer codes. Variables considered in the analyses include affected areas and pathways for exposure to radiation (such as external direct exposure and internal exposure from inhalation), weather conditions and accident and package release fractions. The major contributor to radiation exposure would be from the ground contamination pathway (more than 90% of total exposure).

PROMPT DISMANTLEMENT ALTERNATIVE

The health risks associated with transportation accidents for shipments from the Windsor Site to the Savannah River disposal site for prompt dismantlement are summarized in Table C-16. Analyses indicate that the general population would receive 7.74×10^{-4} person-rem (3.88×10^{-7} additional latent fatal cancer risk) in this scenario.

The health risks associated with transportation accidents for shipments of the same packages from the Windsor Site to the Hanford disposal site for prompt dismantlement are summarized in Table C-17. Analyses indicate that the general population would receive 9.09×10^{-4} person-rem (4.55×10^{-7} additional latent fatal cancer risk) in this scenario. These results are slightly higher than the Savannah River destination due to the greater distance traveled. However, the radiological health risks are still very small.

When compared to the radiological health risks associated with incident-free radioactive waste shipments (see Section 5.1.10.2) the consequences of hypothetical accidents are less. This is due to the very low probability of a severe accident occurring.

5.1.12.2.2 Nonradiological Consequences of Transportation Accidents

There would be no long-term environmental consequences from an accident in which some waste package containing hazardous or toxic material is breached. Hazardous or toxic constituents such as polychlorinated biphenyls, lead, and chromium would be in a solid (insoluble) state. Asbestos, if present, could be disturbed in an accident and portions of the disturbed asbestos might become airborne or mix with water. Any asbestos would eventually settle out of the air or water and become entrained in soil. Naval Reactors would ensure recovery, as necessary, of any spilled hazardous or toxic materials as part of the accident recovery action.

5.1.13 Waste Management

The S1C Prototype reactor plant is small. The volume of the intact reactor compartment is only approximately 293 cubic meters (10,400 cubic feet); the reactor compartment weighs approximately 400 tons. In comparison, decommissioning of the Shippingport pressurized water reactor plant (a small plant by commercial standards) produced approximately 6,060 cubic meters (214,000 cubic feet) of low-level radioactive waste that weighed approximately 4,200 tons. Even though the S1C Prototype reactor plant is small, emphasis would be placed on recycling as much material as practical. Section 3.1.2 described the various waste streams that would be generated as a result of dismantlement activities.

Waste minimization is achievable through recycling and volume reduction. One existing business in Tennessee recycles low-level radioactive metals by melting them into shield blocks which are then provided to the Department of Energy for reuse in high energy physics applications. Other commercial enterprises are also starting to enter the radioactive metal

PROMPT DISMANTLEMENT ALTERNATIVE

recycling field with alternate recycling uses. Recycling and volume reduction services that would be used in conjunction with this alternative would be selected using the normal competitive bidding process.

Low-level radioactive materials from the S1C Prototype reactor plant that could be recycled include stainless steel piping, pumps, valves, and other components and carbon steel structural materials. Low-level radioactive materials would be candidates for recycling if the radioactivity concentration is less than 2×10^{-3} microcuries per gram. As a rule of thumb, components with radiation levels that measure less than 50 millirem per hour on contact would meet the radioactivity concentration criteria for recycling.

Radioactive components that exceed the criteria for recycling could still be candidates for volume reduction if their radiation levels measure less than 200 millirem per hour on contact. Similar to recycling, S1C Prototype reactor plant materials that would be candidates for volume reduction include stainless steel piping, pumps, valves and other components. Volume reduction savings vary widely depending on component construction. Volume reduction processing of S1C Prototype reactor plant materials could achieve an average volume savings in excess of 40%. It is estimated that thirteen shipments would be required for the removal of low-level radioactive materials that would be recycled or volume reduced and for the removal and treatment of mixed waste (see discussion below).

S1C Prototype reactor plant dismantlement would generate approximately 16.7 cubic meters (approximately 600 cubic feet) of elemental lead weighing more than 100 tons that would require recycling or disposal. Lead that could be released from radiological controls would be recycled; other lead containing radioactive impurities or surface contamination would be treated in accordance with the Site Treatment Plan for Mixed Waste Generated at the Windsor Site (Reference 3-2). The 1996 revision of the Site Treatment Plan estimates that 70% of the lead (approximately 11.7 cubic meters (420 cubic feet)) would be released from radiological controls and recycled. The remaining 30% of the lead (approximately 5.0 cubic meters (180 cubic feet)) would be recycled or treated off-site as mixed waste. Decontamination of the lead containing radioactive impurities may not be practical because the impurity concentrations are very low and essentially inseparable. Naval Reactors is evaluating recycling options to reuse lead containing low levels of radioactive impurities in shielding applications at other Naval Reactors facilities. Decontamination of lead with surface contamination is practical with commercially available technology which would further reduce the volume of mixed waste. Residues from treatment of mixed waste would be disposed of off-site.

In addition to elemental lead, the Site Treatment Plan includes twelve other potential mixed waste streams. The estimated volume of mixed waste that could be generated is 11.9 cubic meters (420 cubic feet). The waste streams that would potentially contribute most of the estimated volume include: inorganic debris and equipment, soils, and sludge. The Site Treatment Plan identifies treatment facilities at the Hanford Site, the Savannah River Site, and the Idaho National Engineering Laboratory for off-site treatment of mixed wastes. In the event that identified facilities are not available in time for treatment of mixed wastes generated at the Windsor Site, the Site Treatment Plan states that other options would be evaluated and an Alternate Measures Plan would be submitted to the Environmental Protection Agency, Region I.

After completion of all segregation, recycling, and volume reduction processing initiatives, S1C Prototype reactor plant dismantlement would generate approximately 76 cubic meters (2700 cubic feet) of low-level radioactive waste that would require disposal at a Department of Energy disposal site. This estimate represents approximately 25% of the volume of the intact reactor compartment. Almost half of the low-level radioactive waste volume (35.7 cubic meters, 1262 cubic feet) is due to the reactor pressure vessel package alone (one shipment). In addition to waste from reactor plant dismantlement, Windsor Site dismantlement activities are estimated to generate 30 cubic meters (1040 cubic feet) of low-level radioactive waste mostly originating from the removal of some support systems inside the radiological support facility. Overall, low-level radioactive wastes would include the pressure vessel, steam generator and pressurizer major components, volume reduced-nonrecycled materials, and miscellaneous waste unsuitable for recycling or volume reduction. Low-level radioactive wastes would comprise an estimated ten shipments.

The Savannah River Site has established radioactivity concentration limits for acceptance of waste based upon site specific analysis. In addition, the Savannah River Site Waste Acceptance Criteria prohibits acceptance of waste exceeding the Nuclear Regulatory Commission Class C limits as defined by 10 CFR Part 61 (Licensing Requirements for Land Disposal of Radioactive Waste). Of the total radioactivity remaining in the S1C Prototype reactor plant listed in Table 2-1, more than 95 percent would be in the single package that contains the reactor pressure vessel and its internal structure. This package would be within the limits of the Savannah River Site Waste Acceptance Criteria. The other radioactive waste packages would have much lower radioactivity concentrations.

While the end of the Cold War may result in some increases in radioactive wastes such as S1C Prototype reactor plant dismantlement waste, there has been a larger decrease in radioactive waste generation due to the earlier-than-projected inactivation of nuclear powered ships and prototype reactor plants. As a result, both the volume and the radioactivity content of the S1C low-level waste fall within the projections of Naval Reactor waste provided to the Savannah River Site for disposal. The impacts of Naval Reactor low-level waste disposal activities at the Savannah River Site are analyzed in the recent Savannah River Site Waste Management Final Environmental Impact Statement (Reference 5-2).

Municipal solid waste, nonradioactive hazardous material, and nonradioactive nonhazardous demolition debris from Windsor Site activities would be recycled or disposed of off-site at permitted facilities using licensed haulers. Emphasis would be placed on recycling as much nonradioactive material as practical. Reusable materials such as concrete, lead, carbon steel, and other metals would be reused or recycled through various licensed commercial vendors. Dismantlement and demolition activities would result in approximately 1,100 shipments of nonradioactive waste and recyclable materials from the Windsor Site.

5.1.14 Irreversible and Irretrievable Commitments of Resources

The prompt dismantlement alternative would not involve any irretrievable or irreversible commitment of environmentally sensitive resources. As discussed previously in this section, this alternative would not contribute to any loss of endangered species, critical habitat, or areas of archeological, historical or cultural value. Prompt dismantlement activities would not require any significant demand on consumable resources such as utilities and energy. No additional land at disposal sites would be required to dispose of dismantlement wastes. This alternative would release the Windsor Site land resource for other unrestricted uses in the shortest time.

5.2 Deferred Dismantlement Alternative

This alternative would dismantle the reactor plant, after a 30-year caretaking period, and dispose of waste and recycle materials at that time.

During the caretaking period, the reactor compartment would be seasonally heated and dehumidified to preserve overall system and structure integrity. Radiological work on contaminated systems or opening of contaminated systems in the reactor compartment would not be expected during the caretaking period. To maintain the temperature and humidity, a hull ventilation system would be used. The ventilation system would discharge filtered, monitored air to the environment. Periodic inspections and radiological surveys would be conducted each year during the caretaking period to confirm the continued integrity of the reactor plant systems and reactor compartment structure. The surveys would be performed both inside and outside of the reactor compartment.

Under this alternative, several buildings would remain at the Windsor Site in an inactive condition. The buildings would be used to support dismantlement activities after the thirty-year caretaking period. These buildings would be seasonally heated and dehumidified and routinely inspected. Maintenance would be performed as necessary to sustain their physical integrity. Dismantling the reactor plant plus disposal and recycling of radioactive and nonradioactive materials would be done in the same manner as the prompt dismantlement alternative. Section 3.2 provides a detailed description of the deferred dismantlement alternative. Environmental impacts are discussed below.

5.2.1 Land Use

The deferred dismantlement alternative would not adversely impact area land use but it would prevent unrestricted release of the Windsor Site until at least the year 2031. During the 30-year caretaking period and subsequent dismantlement period, the U.S. Government would maintain ownership of the property. Access to areas inside the Windsor Site's security fence would continue to be controlled during caretaking and deferred dismantlement.

Caretaking activities and dismantlement activities would be confined within the boundary of the Windsor Site property, which is an already developed area. There would be no impact on present or planned use of the surrounding areas. Materials associated with dismantlement activities would be recycled as much as practical and any wastes would be disposed of off-site at licensed disposal facilities. No land on-site and no additional land off-site would have to be set aside for waste disposal.

DEFERRED DISMANTLEMENT ALTERNATIVE

5.2.2 Ecological Resources

Caretaking activities over a 30-year period, and the subsequent dismantlement of the S1C Prototype reactor plant would not impact ecological resources at the Windsor Site or surrounding areas. Windsor Site caretaking activities would include periodic radiological monitoring, visual inspections and maintenance of the reactor compartment (see Section 3.2.1). Environmental monitoring would be continued throughout this alternative and results would be reported annually. Since deferred dismantlement activities would be similar to prompt dismantlement activities, the discussion on ecological impact in Section 5.1.2 also applies to this alternative.

5.2.3 Water Resources

The Windsor Site is not located in a 100 or 500-year floodplain area. Deferred dismantlement activities would not involve major earth moving work. Consequently, floodplains in the vicinity of the Windsor Site would not be affected by deferred dismantlement activities.

5.2.3.1 Radiological Consequences for Water Resources

Caretaking and deferred dismantlement activities would not involve any discharges of radioactive liquid effluents. Ground water under the Windsor Site and surface water in the surrounding environment would not be affected.

5.2.3.2 Nonradiological Consequences for Water Resources

During the 30-year caretaking period and deferred dismantlement, liquid discharges from the Windsor Site would be limited to Storm water runoff. Approximate flow of this drainage was incorporated in the State of Connecticut Department of Environmental Protection General Stormwater Permit application. In the event that excavation activities disturb more than five acres of land, a State of Connecticut Department of Environmental Protection General Permit for the Discharge of Stormwater and Dewatering Wastewater from Industrial Activities will also be obtained. Effluent from the sanitary sewer would continue to be treated in the anaerobic septic system and released below ground through the existing leach field. No environmental effects would be expected. Liquid effluents would continue to be monitored and results would continue to be reported annually through completion of this alternative.

5.2.4 Air Resources

5.2.4.1 Radiological Consequences for Air Resources

Airborne particulate radioactivity emissions associated with incident-free caretaking and deferred dismantlement activities were evaluated. The details of the analyses are provided in Appendix B, Section B.2. Analyses for this alternative utilized two assumed source terms - one for the 30-year caretaking period and a second one for the two-year deferred dismantlement period. For the caretaking period, the release source term was derived using the minimum detectable airborne radioactivity concentration of 2 x 10^{-14} microcuries per milliliter and the expected volume of ventilation air which would flow through the reactor compartment. For deferred dismantlement activities, the airborne particulate radioactivity source term was derived from radiation levels measured on high efficiency particulate air filters in reactor servicing ventilation systems used during past maintenance activities. The high efficiency particulate air filters have a greater than 99.95% efficiency for removal of airborne particulate radioactivity. It is conservatively estimated that 1.6 x 10^{-6} curies per year would be discharged during the caretaking period (4.8 x 10^{-5} total curies over thirty years) and 7.2 x 10^{-6} would be discharged during deferred dismantlement (1.4 x 10^{-5} for the two-year dismantlement period). Over the thirty-two year duration of this alternative, it is estimated that 6.2 x 10^{-5} curies would be cumulatively discharged into the air. As discussed in Chapter 4, Section 4.4.4, and reported in Reference 4-10, the radioactivity contained in exhaust air from calendar year 1994 Windsor Site activities totaled less than 1 x 10^{-3} curies of particulate fission and activation products and had no environmental impact. Therefore, deferred dismantlement activities would have no significant radiological consequences on air resources.

5.2.4.2 Nonradiological Consequences for Air Resources

During the caretaking period, there would be no regulated point sources of nonradiological industrial gaseous emissions at the Windsor Site. The principal source of nonradiological airborne emissions would be from liquid propane fueled heating units for preservation of remaining Windsor Site buildings. Nonradiological emissions would be approximately the same as current baseline conditions and would have no significant environmental impact. The discussion of nonradiological consequences of prompt dismantlement for air resources in Section 5.1.4.2 applies equally to the deferred dismantlement period.

5.2.5 Terrestrial Resources

Caretaking activities and the subsequent dismantlement of the defueled S1C Prototype reactor plant would not adversely affect terrestrial resources at the Windsor Site or surrounding areas. As discussed in Section 3.2.3, unnecessary buildings, systems, and pavement would be removed early to reduce caretaking and future Windsor Site restoration costs. Excavation activities for system removals would be accomplished in small, limited areas. Windsor Site restoration activities following deferred S1C Prototype reactor plant dismantlement would be the same as described in Section 5.1.5 for prompt dismantlement.

DEFERRED DISMANTLEMENT ALTERNATIVE

5.2.5.1 Expected Final Radiological Conditions of the Windsor Site Property After Deferred Dismantlement

After completion of all deferred S1C Prototype dismantlement activities, including final removal of Windsor Site buildings, systems and pavement, the discussion of Section 5.1.5.1 for unrestricted Windsor Site release conditions would apply similarly.

5.2.5.2 Expected Final Nonradiological Conditions of the Windsor Site Property After Deferred Dismantlement

After completion of all deferred dismantlement activities, including final removal of Windsor Site buildings, systems and pavement, the discussion of Section 5.1.5.2 for unrestricted Windsor Site release conditions would apply similarly.

5.2.6 Socioeconomics

Current Windsor Site staffing is about 150 personnel. This alternative results in a staff reduction for the caretaking period. The labor force needed to support caretaking activities at the Windsor Site is estimated at 8 full-time workers. Deferred dismantlement would require rehiring staff for a relatively short (2 year) period. During deferred dismantlement activities, staffing levels and demographics are expected to be similar to those described in the prompt dismantlement alternative (about 150 total personnel). Staff fluctuations associated with deferred dismantlement would not significantly affect regional unemployment levels.

Since the Windsor Site is currently owned by the U.S. Government, the Site is not taxable. Under the deferred dismantlement alternative, the possible transfer of the Windsor Site to a taxpaying entity would be delayed by approximately thirty years. However, considering the small size of the Windsor Site, the impact on the tax base of the town is not expected to be significant. Consequently, deferred dismantlement would not have any discernible socioeconomic impact on the region.

5.2.7 Cultural Resources

Caretaking activities and deferred dismantlement of the S1C Prototype reactor plant would not impact any cultural resources predating Windsor Site construction. Measures that will be carried out by the Department of Energy to address the effects of dismantlement activities on the Windsor Site and the S1C prototype reactor plant are identified in the Memorandum of Agreement with the State of Connecticut, as approved by the Advisory Council on Historic Preservation. A copy of this Memorandum of Agreement is included in Appendix E of this Environmental Impact Statement.

DEFERRED DISMANTLEMENT ALTERNATIVE

5.2.8 Noise, Aesthetic and Scenic Resources

Caretaking activities and deferred dismantlement of the S1C Prototype reactor plant would not have noticeable noise, aesthetic or scenic impacts. During caretaking, the aesthetic and scenic character of the Windsor Site would be maintained consistent with present conditions. Noise generation above ambient levels would not be expected. Dismantlement activities after the caretaking period would be similar to prompt dismantlement activities. The discussion on noise, aesthetic and scenic impacts in Section 5.1.8 is equally applicable to deferred dismantlement.

5.2.9 Traffic and Transportation

Caretaking activities and deferred dismantlement of the S1C Prototype reactor plant would not have a noticeable impact on regional and local traffic. During caretaking, Windsor Site staffing levels and traffic to and from the Windsor Site would be minimal. After caretaking, Windsor Site staffing and traffic would return to present levels for about a 2-year deferred dismantlement period. Impact on traffic during deferred dismantlement would be similar to that for prompt dismantlement, discussed in Section 5.1.9.

5.2.10 Occupational and Public Health and Safety

This section summarizes analysis results for expected incident-free conditions during a 30-year caretaking period followed by a two-year deferred dismantlement period. Section 5.2.12 summarizes analysis results for potential accident conditions during caretaking, deferred dismantlement and transport of materials.

5.2.10.1 Incident-Free Facility Activities

5.2.10.1.1 Incident-Free Facility Activities - Radiological Consequences

The radiological health risks associated with incident free-facility activities during a 30-year caretaking period and deferred dismantlement of the S1C Prototype reactor plant were evaluated in Appendix B, Section B.2. Effects from assumed airborne particulate radioactivity releases and direct radiation exposure were assessed for the worker, maximally exposed off-site individual and the general population. During a 30-year caretaking period, much of the short half-life radionuclides, primarily cobalt-60, would decay. This decay would result in a reduction factor of 52 for direct radiation exposure to workers compared to the prompt dismantlement alternative.

DEFERRED DISMANTLEMENT ALTERNATIVE

Analyses for radiological exposure during the caretaking period and deferred dismantlement were made using a consistent approach as the analyses for prompt dismantlement, discussed in Section 5.1.10.1.1. Occupational exposure over the course of 30-years of caretaking activities would be approximately 2.1 person-rem. This occupational exposure is the same for the 30-year caretaking period of the no action alternative. Occupational exposure from deferred dismantlement activities would be in the range of 1.8 to 3.6 person-rem.

The combined health risks for direct radiation exposure and radioactive contamination to the air are summarized in Table B-6. It is conservatively estimated that the caretaking and dismantlement workers would receive between 3.9 to 5.7 person-rem (1.6×10^{-3} to 2.3×10^{-3} additional latent fatal cancer risk). The general population would receive 3.2×10^{-2} person-rem (1.6×10^{-5} additional latent fatal cancer risk) from exposure during caretaking and dismantlement.

5.2.10.1.2 Incident-Free Facility Activities - Nonradiological Consequences

Naval Reactors policy is to maintain a safe and healthful environment at all facilities, including the Windsor Site. During the caretaking period, no dismantlement activities would occur, no hazardous wastes or bulk supplies of materials would be stored, and facility activities would be limited to surveillance and security tours by a small number of personnel. As a result, incident-free nonradiological consequences would be insignificant. During deferred dismantlement activities, the nonradiological consequences during incident-free facility activities would be the same as the prompt dismantlement alternative, discussed in Section 5.1.10.1.2.

5.2.10.2 Incident-Free Transportation Analyses

The discussion in Section 5.1.10.2 for shipment destinations and transportation analysis assumptions applies equally to the deferred dismantlement alternative.

5.2.10.2.1 Incident-Free Transportation - Radiological Consequences

The radiological consequences associated with incident-free shipment of low-level radiological recyclable material and waste from deferred dismantlement were analyzed using the same approach described in Section 5.1.10.2.1. The potential radiological health risks associated with incident-free transportation of reactor plant components were evaluated using the RADTRAN 4 computer code. Health effects were assessed for the general population, transportation crew, hypothetical maximally exposed individuals in the general population and maximally exposed individual in the transportation crew. As discussed in Appendix C, Section C.2, a conservative simplification was made in the transportation analyses which assumed that all radioactive recyclable material or waste would be shipped to the same location, either the Savannah River disposal site in South Carolina or the Hanford disposal site in Washington State. Details for the technical approach for assessing incident-free radioactive shipments are provided in Appendix C, Section C.4. Computer model variables and assumptions are provided in Appendix C, Section C.5.

Packaging for the reactor pressure vessel shipment would be designed to meet the same transport index for both the deferred and prompt dismantlement alternatives. Analysis results for this one shipment are identical for both the deferred and prompt dismantlement alternatives. The radiological risks for shipment of all other radioactive recyclable materials and wastes under the deferred dismantlement alternative would be lower due to cobalt-60 decay. However, the volume of radioactive waste would not be reduced since the activated and contaminated reactor plant materials would still be radioactive due to long half-life radionuclides such as nickel-63.

The health risks due to low-level radioactive material shipments from the Windsor Site to the Savannah River disposal site for deferred dismantlement are summarized in Table C-14. Analyses indicate that the general population would receive 4.31×10^{-2} person-rem (2.15×10^{-5} additional latent fatal cancer risk) from shipment of low-level radioactive materials from the Windsor Site to the Savannah River disposal site. Transportation workers would receive 1.40×10^{-1} person-rem (5.61×10^{-5} additional latent fatal cancer risk) for the same shipments.

The health risks for shipments of the same packages to the Hanford disposal site are summarized in Table C-15 and are slightly higher due to the greater distance traveled. However, the radiological health risks are still very small. Analyses indicate the general population would receive 1.09×10^{-1} person-rem (5.46×10^{-5} additional latent fatal cancer risk) and the transportation crew would receive 2.18×10^{-1} person-rem (8.70×10^{-5} additional latent fatal cancer risk).

5.2.10.2.2 Incident-Free Transportation - Nonradiological Consequences

The nonradiological consequences associated with incident-free shipment of low-level radiological recyclable material and waste are identical for the prompt and deferred dismantlement alternatives. The discussion in Section 5.1.10.2.2 is equally applicable for the deferred dismantlement.

5.2.11 Utilities and Energy

Caretaking activities over a 30-year period, and the subsequent dismantlement of the defueled S1C Prototype reactor plant would not place large demands on utilities and energy resources.

Utility and energy usage during the caretaking period, such as seasonal heating and dehumidification of the reactor plant, would be minimal. Dismantlement activities would require quantities of fuel, water, and electricity typical of small to medium-sized construction or demolition projects. The amount of utilities and energy expected to be consumed would not result in any discernible environmental consequences.

5.2.12 Occupational and Transportation Accidents

Hypothetical accident scenarios were evaluated to estimate the potential for, and effects of, release of radioactive material and toxic chemicals. Appendix B, Section B.3 provides details of hypothetical facility accidents resulting in the release of radioactive materials to the environment. Appendix B, Section B.4 provides analysis of a diesel fuel fire. Appendix C, Section C.6 describes the technical approach for assessing radioactive shipment accidents. The result of these analyses are presented in terms of the health risks to facility workers and the public. The overall health risk is a product of the probability that the accident would occur and the consequences resulting from the accident.

5.2.12.1 Facility Accidents

5.2.12.1.1 Radiological Consequences of Facility Accidents

Several hypothetical accident scenarios that would result in uncontrolled release of radioactivity to the environment were evaluated to determine the long term health risks. The hypothetical release of airborne radioactivity and exposure to direct radiation during accident scenarios were assessed for the worker, maximally exposed off-site individual and the general population.

As described in Appendix B, Section B.1.2, accidents were considered if they were expected to contribute substantially to risk. Risk is defined as the product of the probability of occurrence times the consequence of the accident. The four hypothetical accident scenarios evaluated for the dismantlement activities included 1) a large component drop, 2) mechanical damage of a component due to a wind-driven missile, 3) an airplane crash into the reactor plant with damage to several components, 4) and a high efficiency particulate air filter fire. Variables considered in the analyses include source terms, population density, meteorological conditions, affected area and pathways for exposure to radiation (such as external direct exposure and internal exposure from inhalation).

For the caretaking period, a high efficiency particulate air filter fire was evaluated. The other accident scenarios were considered but were not evaluated in detail. A component drop accident was not evaluated since lifting or handling of large components would not occur during the caretaking period. The steel hull of the reactor compartment would absorb most of the energy from any airplane crashes or wind-driven missiles and would limit any release of radioactive materials to the environment. All four accident scenarios were considered for the two-year deferred dismantlement period following the thirty-year caretaking period.

The details of the analysis are provided in Appendix B, Section B.3. As shown in Appendix B, Table B-16, during the caretaking period, the cumulative risk of a member of the general population developing a latent fatal cancer from a high efficiency particulate air filter fire accident is 2.6×10^{-8}. This risk is the product of the probability of the accident occurring

(5×10^{-4} per year) times the consequence of the accident (1.7×10^{-6} latent fatal cancers) times 30 years. During deferred dismantlement activities, the accident with the greatest annual risk is an airplane crashing into the reactor plant. The annual risk of a member of the general population developing a latent fatal cancer due to an airplane crash accident is 1.1×10^{-8}. This risk is a product of the probability of the accident occurring (6.6×10^{-7} per year) times the consequence of the accident (1.7×10^{-2} latent fatal cancers). The cumulative risk during the thirty-year caretaking period combined with the cumulative risk during the two-year deferred dismantlement period yields a total cumulative risk of 4.8×10^{-8} for the entire duration of the deferred dismantlement alternative. This is an extremely small risk compared to other incident-free risks due to the low probability of the accidents occurring.

5.2.12.1.2 Nonradiological Consequences of Facility Accidents

The nonradiological consequences associated with the caretaking and dismantlement activities are the same as the prompt dismantlement alternative. The discussion of Section 5.1.12.1.2 is applicable for the deferred dismantlement alternative. During the caretaking period, no dismantlement activities would occur, no hazardous wastes would be stored, and no bulk supplies of materials would be stored.

Nonradiological occupational accidents, such as slips and falls could occur during the caretaking period and deferred reactor plant dismantlement. However, the rate is not expected to be greater than rates for other construction activities (provided in Table 4-6). Projections of the number of fatalities, injuries or illnesses were calculated based on Table 4-6 Department of Energy rates and are summarized in Table 5-2 for the deferred dismantlement alternative. These results indicate that the overall nonradiological occupational risks are small.

DEFERRED DISMANTLEMENT ALTERNATIVE

Table 5-2: Estimated Nonradiological, Occupational Impacts for Deferred Dismantlement

	Caretaking	Dismantlement
Estimated Windsor Site Staffing Level	8	150
Estimated average injuries/illnesses per year [a]	2.9×10^{-1}	9.9
Estimated fatalities per year [a]	2.4×10^{-4}	1.5×10^{-2}
Total estimated number of injuries/illnesses [b]	8.7	19.8
Combined totals	28.5	
Total estimated number of fatalities [b]	7.2×10^{-3}	3.0×10^{-2}
Combined totals	3.7×10^{-2}	

a. Calculated by multiplying Windsor Site staffing levels times the Department of Energy rates provided in Table 4-6. Rates for construction workers were used for dismantlement activities and rates for all labor categories were used for caretaking activities.

b. Total values calculated for a 30-year caretaking period and two-year dismantlement period.

5.2.12.2 Transportation Accidents

There has never been a major accident nor measurable release of radioactivity to the environment during shipment of Naval Reactors program waste or materials, however, hypothetical accidents were evaluated to determine potential environmental effects.

5.2.12.2.1 Radiological Consequences of Transportation Accidents

Appendix C, Section C.6 provides the technical approach used for assessing hypothetical radioactive shipment accidents. Health effects were assessed for the general population and the hypothetical maximally exposed individual. Analyses assumed that the transportation workers would evacuate the scene of an accident within a relatively short time after the accident occurred. Therefore, the risks of transportation accidents on transportation workers are included in the results for the general population.

Radiological health risks from uncontrolled releases of radioactivity to the environment and direct radiation exposure from damaged packages were evaluated using the RADTRAN 4 and RISKIND computer codes. Variables considered in the analyses include affected areas and pathways for exposure to radiation (such as external direct exposure and internal exposure from inhalation), weather conditions and accident and package release fractions. The major contributor to radiation exposure would be from the ground contamination pathway (more than 90% of total exposure).

DEFERRED DISMANTLEMENT ALTERNATIVE

The health risks associated with transportation accidents for shipments from the Windsor Site to the Savannah River disposal site for deferred dismantlement are summarized in Table C-18. Analyses indicate that the general population would receive 1.75×10^{-5} person-rem (8.75×10^{-9} additional latent fatal cancer risk) in this scenario.

The health risks associated with transportation accidents for shipments of the same packages from the Windsor Site to the Hanford disposal site for prompt dismantlement are summarized in Table C-19. Analyses indicate that the general population would receive 2.08×10^{-5} person-rem (1.04×10^{-8} additional latent fatal cancer risk) in this scenario. These results are slightly higher than the Savannah River destination due to the greater distance traveled. However, the radiological health risks are still very small.

When compared to the radiological health risks associated with incident-free radioactive recyclable material and waste shipments (see Section 5.2.10.2) the consequences of hypothetical accidents are less. This is due to the very low probability of a severe accident occurring.

5.2.12.2.2 Nonradiological Consequences of Transportation Accidents

The nonradiological consequences associated with the transportation accidents for the prompt dismantlement alternative, Section 5.1.12.2.2, are the same for the deferred dismantlement alternative.

5.2.13 Waste Management

As discussed in Section 3.2.2, deferred dismantlement activities would be similar to prompt dismantlement activities. Deferred dismantlement would not result in any reduction in the estimated radioactive material volume. Although cobalt-60 will decay to less than 2% of the levels at the start of a 30-year caretaking period, other long half-life radionuclides such as nickel-63 will remain. Nickel-63 has a half-life of approximately 100 years and will decay to only 81% of its initial levels after 30 years.

Deferred dismantlement would result in the same number of shipments of recyclable materials and wastes as the prompt dismantlement alternative. Low-level radioactive waste from deferred dismantlement would meet the same disposal site requirements as discussed in Section 5.1.13. Decay of radioactivity in the S1C Prototype reactor plant could allow for a greater percentage of radioactive metals to be candidates for recycling or volume reduction than the percentages discussed in Section 5.1.13. However, considering that the estimated volume and curie content of low-level radioactive wastes associated with prompt dismantlement falls within ranges currently experienced within the Department of Energy, deferred dismantlement would have an even lower environmental effect. The volume of mixed waste resulting from deferred dismantlement is estimated to be the same as discussed in Section 5.1.13.

DEFERRED DISMANTLEMENT ALTERNATIVE

During caretaking, waste generated would consist mainly of municipal trash, and disposal would be consistent with state and local regulations.

5.2.14 Irreversible and Irretrievable Commitments of Resources

The deferred dismantlement alternative would not involve any irretrievable or irreversible commitment of environmentally sensitive resources. As discussed previously in this section, this alternative would not contribute to any loss of endangered species, critical habitat, or areas of archeological, historical or cultural value. Deferred dismantlement activities would not require any significant demand on consumable resources such as utilities and energy. No additional land at disposal sites would be required to dispose of dismantlement wastes. This alternative delays release of the Windsor Site land resource for other unrestricted uses for at least thirty-two years.

5.3 No Action Alternative

This alternative would maintain the defueled and drained reactor plant in a protected condition for an indefinite period of time. Caretaking period operations for this alternative would be identical to caretaking period operations described in the deferred dismantlement alternative (section 3.2.1), except that the voluntary facility assessment process (described in Chapter 4 and Appendix F) and the radiological survey process (discussed in Appendix G) and any associated remediation activities would not be completed. Periodic inspections, radiological surveys, and reactor compartment ventilation systems would be the same. For the purposes of comparison to the other alternatives, a 30-year time frame was assumed in analyses that evaluate the environmental effects of this alternative. Section 3.3 provides a detailed description of the no action alternative. Environmental impacts are discussed below.

5.3.1 Land Use

While the no action alternative would not adversely impact area land use compared to present conditions, the alternative prevents the possibility of unrestricted release of the Windsor Site property. Caretaking activities would be confined within the Windsor Site property boundary which is an already developed area. There would be no interference with present or planned use of the surrounding areas. The U.S. government would maintain ownership of the Windsor Site property. Access to areas inside the Windsor Site's security fence would continue to be controlled for as long as the defueled S1C Prototype reactor plant remains on-site.

5.3.2 Ecological Resources

Caretaking activities over an indefinite period would not impact ecological resources at the Windsor Site or surrounding areas. Windsor Site caretaking activities would include periodic radiological monitoring, visual inspections, and maintenance of the reactor compartment (see Section 3.2.1). Environmental monitoring would be continued and results would be reported annually.

5.3.3 Water Resources

The Windsor Site is not located in a 100 or 500-year floodplain area. Caretaking activities would not involve any earth moving work. Flood plains in the vicinity of the Windsor Site would not be affected by caretaking activities.

5.3.3.1 Radiological Consequences for Water Resources

Caretaking activities would not result in any discharges of radioactive liquid effluents. Ground water under the Windsor Site and surface water in the surrounding environment would not be affected.

NO ACTION ALTERNATIVE

5.3.3.2 Nonradiological Consequences for Water Resources

During caretaking activities, liquid discharges from the Windsor Site would be limited to Storm water runoff. Approximate flow of this drainage was incorporated in the State of Connecticut Department of Environmental Protection General Storm water Permit application. Liquid effluents would continue to be monitored and results would continue to be reported annually.

5.3.4 Air Resources

5.3.4.1 Radiological Consequences for Air Resources

Airborne particulate radioactivity emissions associated with the no action alternative were evaluated. The details of the analyses are provided in Appendix B, Section B.2. Analyses for this alternative assumed an airborne particulate radioactivity source term which was derived using the minimum detectable airborne radioactivity concentration of 2×10^{-14} microcuries per milliliter and the expected volume of ventilation air which would flow through the reactor compartment. It is conservatively estimated that 1.6×10^{-6} curies per year would be discharged during the caretaking period (4.8×10^{-5} total curies over thirty years). As discussed in the deferred dismantlement alternative, Section 5.2.4.1, caretaking activities would have no significant radiological consequences on air resources.

5.3.4.2 Nonradiological Consequences for Air Resources

During the no action alternative, there would be no regulated point sources of nonradiological industrial gaseous emissions at the Windsor Site. The principal source of nonradiological airborne emissions would be from liquid propane fueled heating units for preservation of remaining Windsor Site buildings. Nonradiological emissions would be approximately the same as current baseline conditions and would have no significant environmental impact.

5.3.5 Terrestrial Resources

Caretaking activities over an indefinite period would not impact terrestrial resources on or surrounding the Windsor Site.

NO ACTION ALTERNATIVE

5.3.5.1 Expected Radiological Conditions of the Windsor Site Property During the No Action Alternative

The radiological conditions would remain essentially as they are today, as described in Section 4.5.4.1. Since there is no significant adverse radiological effect caused by the current condition, no significant adverse environmental impact would be expected in the future other than the need to restrict access to the actual prototype plant.

5.3.5.2 Expected Nonradiological Conditions of the Windsor Site Property During the No Action Alternative

The nonradiological conditions would remain as they are today, as described in Section 4.5.5.1. Since the voluntary facility assessment process (described in Chapter 4) would not be completed, a small amount of chemically contaminated soil might remain on the Windsor Site. Since no adverse environmental impacts associated with these conditions have been noted, no adverse impacts would be expected in the future, other than the remaining presence of this material on the Windsor Site.

5.3.6 Socioeconomics

Current Windsor Site staffing is about 150 personnel. The no action alternative would result in a staff reduction for the caretaking period. The labor force needed to support caretaking activities would be 7.7 equivalent full-time workers. Staff reductions associated with the no action alternative would not significantly affect regional unemployment levels.

Since the Windsor Site is currently owned by the U.S. Government, the Site is not taxable. Under the no action alternative, the possible transfer of the Windsor Site to a taxpaying entity would be delayed indefinitely. However, considering the small size of the Windsor Site, the impact on the tax base of the town is not expected to be significant. Consequently, the no action alternative would not have any discernible socioeconomic impact on the region.

5.3.7 Cultural Resources

Caretaking activities over an indefinite period would not impact any cultural resources predating Windsor Site construction.

5.3.8 Noise, Aesthetic and Scenic Resources

Caretaking activities over an indefinite period would not have any noise, aesthetic or scenic impacts. During caretaking, the aesthetic and scenic character of the Windsor Site would be maintained consistent with present conditions. Noise generation above ambient levels would not be expected.

NO ACTION ALTERNATIVE

5.3.9 Traffic and Transportation

Caretaking activities over an indefinite period would not impact regional and local traffic since the Windsor Site staffing level and associated traffic would be minimal.

5.3.10 Occupational and Public Health and Safety

Detailed analyses of potential impacts on worker and public health are described in Appendix B for facility activities. This section summarizes analysis results for expected incident-free conditions during a 30-year caretaking period for the no action alternative. There would be no off-site transport of materials during this alternative, hence, Appendix C analyses do not apply to this alternative. Section 5.3.12 summarizes the analyses results associated with potential accident scenarios during the caretaking period.

5.3.10.1 Incident-Free Facility Activities - Radiological Consequences

The radiological health risks associated with incident-free facility activities during a 30-year caretaking period were evaluated in Appendix B, Section B.2, for the no action alternative. Effects from assumed airborne particulate radioactivity releases and exposure to direct radiation were assessed for the worker, maximally exposed off-site individual and the general population.

Caretaking activities during the no action alternative and the deferred alternative are the same. The discussions in Section 5.2.10.1.1, relative to caretaking, are applicable to the no action alternative.

The details of the analyses are provided in Appendix B, Section B.2. The combined health risks for direct radiation exposure and radioactive exposure to the air have been summarized in Table B-6. It is conservatively estimated that the radiation workers would receive 2.1 person-rem (8.4×10^{-4} additional latent fatal cancer risk) and the general population would receive 3.2×10^{-2} person-rem (1.6×10^{-5} additional latent fatal cancer risk) from exposure during caretaking.

5.3.10.2 Incident-Free Facility Activities - Nonradiological Consequences

Naval Reactors policy is to maintain a safe and healthful environment at all facilities, including the Windsor Site. During this alternative, no dismantlement activities would occur, no hazardous wastes or bulk supplies of materials would be stored, and facility activities would be limited to surveillance and security tours by a small number of personnel. As a result, incident-free nonradiological consequences would be insignificant.

NO ACTION ALTERNATIVE

5.3.11 Utilities and Energy

Caretaking activities over an indefinite period would not place a large demand on utilities and energy resources. Usage during the caretaking period, such as for seasonal heating and dehumidification of the reactor plant, would be very small.

5.3.12 Facility Accidents

Hypothetical accident scenarios were evaluated to estimate the potential for, and effects of, release of radioactive material and toxic chemicals. Appendix B, Section B.3 provides details of hypothetical facility accidents resulting in the release of radioactive materials to the environment. Appendix B, Section B.4 provides analysis of a nonradiological fuel fire. The result of these analyses are presented in terms of the health risks to facility workers and the public. The overall health risk is a product of the probability that the accident would occur and the consequences resulting from the accident.

5.3.12.1 Radiological Consequences of Facility Accidents

For the no action alternative, only a high efficiency particulate air filter fire was evaluated in detail. The other accident scenarios were considered but were not evaluated in detail. A component drop accident was not evaluated since lifting or handling of large components would not occur during the caretaking period. The steel hull of the reactor compartment would absorb most of the energy from any airplane crashes or wind-driven missiles and would limit any release of radioactive materials to the environment.

The details of the analysis are provided in Appendix B, Section B.3. As shown in Table B-16, the annual risk of a member of the general population developing a latent fatal cancer due to a high efficiency air filter fire during a caretaking period is 8.5×10^{-10}. This risk is a product of the probability of the accident occurring (5×10^{-4} per year) times the consequence of the accident (1.7×10^{-6} latent fatal cancers). Over the thirty-year duration considered for the no action alternative, the cumulative risk to the general population from a high efficiency particulate air filter fire is 2.6×10^{-8}. This is extremely small compared to other incident-free radiological impacts.

5.3.12.2 Nonradiological Consequences of Facility Accidents

During the caretaking period, there would be no hazardous waste or bulk storage quantities of other products at the Windsor Site. Therefore, the only type of nonradiological accident that was considered for the no action alternative was worker accidents. Projections of the number of fatalities, injuries or illnesses were calculated based on Table 4-6 Department of Energy rates and results are summarized in Table 5-3 for the no action alternative. These results indicate that the overall nonradiological occupational risks are small.

NO ACTION ALTERNATIVE

Table 5-3: Estimated Nonradiological, Occupational Impacts for No Action

Estimated Windsor Site Staffing Level (equivalent full-time workers)	7.7
Estimated average injuries/illnesses per year [a]	2.8×10^{-1}
Estimated fatalities per year [a]	2.3×10^{-4}
Total estimated injuries/illnesses [b]	8.4
Total estimated number of fatalities [b]	6.9×10^{-3}

a. Calculated by multiplying Windsor Site staffing level times the Department of Energy rates for all labor categories provided in Table 4-6.

b. Total values calculated for a 30-year caretaking period.

5.3.13 Waste Management

Caretaking activities over an indefinite period would generate only very small volumes of waste. Waste generated would consist mainly of municipal trash, and disposal would be consistent with state and local regulations.

5.3.14 Irreversible and Irretrievable Commitments of Resources

The no action alternative would not involve any irretrievable or irreversible commitment of environmentally sensitive resources. As discussed previously in this section, this alternative does not impact any endangered species, critical habitat, or areas of archeological, historical or cultural value. Demand on consumable resources such as utilities and energy for caretaking of the S1C Prototype reactor plant and remaining Windsor Site buildings would be negligible. Under this alternative, the Windsor Site land resource would continue to be unavailable for other uses indefinitely.

5.4 Cumulative Impacts and Comparison of Alternatives

A summary of cumulative impacts associated with the different alternatives is provided in the following sections. There are no significant cumulative impacts specifically associated with any of the three reasonable alternatives for disposal of the S1C Prototype reactor plant. Because the health risks to the public from transportation of recyclable materials and wastes would be extremely small and indistinguishable from other unrelated health risks, there would be no cumulative transportation related impacts.

5.4.1 Land Use

There are no cumulative land use impacts specifically associated with any of the alternatives considered. The existing land of the entire Windsor Site, 10.8 acres, has already been disturbed from its natural state. The alternatives would not disturb any additional undeveloped land or add land to the Windsor Site. The Windsor Site and the surrounding land are both zoned for industrial use. The alternatives would not affect the current and future use of land surrounding the Windsor Site. Prompt dismantlement could allow for the unrestricted release of the Windsor Site property for other uses, consistent with the existing zoning, as early as 2001. Deferred dismantlement would postpone the unrestricted release of the Windsor Site property until 2031 at the earliest. Similarly, the no action alternative would postpone unrestricted release indefinitely while the S1C Prototype reactor compartment remains in place.

Low-level radioactive waste would meet the disposal site requirements discussed in Section 5.1.13. For the deferred dismantlement alternative, decay of radioactivity in the S1C Prototype reactor plant could allow for a greater percentage of the radioactive metals to be candidates for recycling or volume reduction than the percentages discussed in Section 5.1.13 for prompt dismantlement. However, the estimated volume and curie content of the low-level radioactive wastes associated with prompt dismantlement falls within ranges currently experienced within the Department of Energy. Deferred dismantlement would have an even lower environmental effect. The volume of mixed waste resulting from both the prompt and deferred dismantlement is estimated to be the same as discussed in Section 5.1.13.

5.4.2 Water Resources

There are no cumulative water resource impacts specifically associated with any of the alternatives. The Windsor Site property does not include any bodies of open surface water. An overview of historical impacts from liquid effluents discharged at the Windsor Site is discussed in detail in Section 4.3.

Since 1979, only nonradioactive water discharges have been released from the Windsor Site. As discussed in the annual Knolls Atomic Power Laboratory Environmental Monitoring Report (Reference 4-10), there was no significant impact from radiological discharges during former Windsor Site operations on the environment or adverse effect on the community or the public. None of the alternatives would result in the discharge of radiological effluents.

As discussed in Section 4.3.4, nonradiological discharges have included process water and storm water runoff. None of the alternatives would result in discharges other than storm water runoff. The storm water runoff that would occur during the alternatives would not add any cumulative effects from the existing conditions.

5.4.3 Air Resources

There are no cumulative air resource impacts specifically associated with any of the alternatives considered. Existing operations having a potential for the release of airborne particulate radioactivity are serviced by monitored exhaust systems. Prior to release, the exhaust air is passed through high efficiency particulate air filters to minimize radioactivity content. As reported in Section 4.4.4, the radioactivity contained in exhaust air for 1994 consisted of less than 1×10^{-3} curies of particulate fission and activation products and approximately 9×10^{-3} curies of tritium. The average radioactivity concentration was well below applicable standards (Reference 3-3). The annual radioactivity concentration at the nearest Windsor Site boundary, allowing for typical diffusion conditions, was less than 0.01 percent of the Department of Energy derived concentration guide for air released to unrestricted areas (Reference 3-3). Public radiation exposures from airborne radioactivity are calculated using computer models qualified for this specific task. These models conservatively estimate the radiation exposure to the public through many pathways, including radioactivity in surface soil, vegetation and animal pathways from airborne radioactivity sources. The exposures are calculated using computer models because direct measurement results are indistinguishable from naturally occurring background radioactivity levels.

As discussed in Sections 5.1.4.1, 5.2.4.1, 5.3.4.1, and Appendix B, Section B.2, radiological airborne emissions associated with incident-free activities under the three alternatives have been estimated from 1.5×10^{-5} to 6.2×10^{-5} curies. These emissions would not have a discernible effect on the existing Windsor Site discharge of airborne radioactivity. Therefore, none of the alternatives would have a cumulative impact on the existing radiological air emissions.

The existing nonradiological air emissions from the Windsor Site are from three liquid propane heating units. There are no longer any regulated sources of nonradiological pollutant air emissions at the Windsor Site. During the dismantlement activities of the prompt and deferred alternatives, dust from demolition work and vehicle exhaust emissions would result in a small incremental addition to the Windsor Site nonradiological air emissions. Cumulative air emissions would not threaten to exceed any applicable Federal, State, or local air quality requirement or regulation.

5.4.4 Transportation

The cumulative transportation impacts associated with the dismantlement activities of the prompt and deferred alternatives would be small. The estimated total of twenty-three radioactive material shipments from the Windsor Site would be a small part of the more than two million shipments of radioactive materials made annually in the United States (Reference 5-4). Since deferred dismantlement would not reduce the volume of radioactive waste generated compared to prompt dismantlement, due to long-lived radionuclides, the cumulative transportation impacts would be the same for both alternatives.

5.5 Unavoidable Adverse Effects

There are no discernible unavoidable adverse effects associated with the implementation of any of the alternatives and none which would help to choose among the alternatives. The prompt dismantlement alternative would result in a greater occupational dose during dismantlement, and would cause the public to be exposed to small amounts of radiation during transportation of radioactive recyclable materials and waste. However, associated health effects would be very low, with much less than one latent fatal cancer expected. There would be no changes to the ecological, cultural, geological, and aesthetic resources due to the implementation of any of the alternatives.

5.6 Preventive and Mitigative Measures

The ALARA concept (As Low As Reasonably Achievable) would be applied to work at the Windsor Site to minimize radiological exposure to the work force and to the general public. Workers would be trained to perform their assigned tasks using approved procedures in a safe, efficient manner to reduce the likelihood of personal injury, equipment or facility damage and environmental consequences.

The question of what remediation of the brook, if any, may be required is a subject that is under the purview of the property owner and appropriate regulatory agencies. Since this Environmental Impact Statement is intended to arrive at a decision on alternatives for dismantling the S1C Prototype reactor plant and releasing the Government-owned Windsor Site for unrestricted use, and since the brook is not on the Windsor Site property, specific alternatives for potential remedial actions for the brook are beyond the scope of this Environmental Impact Statement.

CHAPTER 6
ENVIRONMENTAL JUSTICE

6.0 ENVIRONMENTAL JUSTICE

6.1 Introduction

Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations* (*Federal Register* 59 FR 32, page 7629, dated February 16, 1994) directs Federal agencies to identify and address disproportionately high and adverse human health or environmental effects of their programs, policies and activities on minority or low-income populations. A disproportionately high and adverse human health or environmental effect occurs when there is a high adverse effect that occurs for minority or low-income populations at an appreciably higher rate than occurs for the general population.

6.2 Community Characteristics

The Capital (north central) Region of Connecticut was selected as a reasonable area for consideration of environmental justice impacts analysis. The region is made up of 29 municipalities, including the Town of Windsor. According to the 1990 Census, the region population is about 18% minority and about 8% at or below poverty level.

Definitions for minority and low-income populations are based on definitions used in the Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement, (Reference 1-1). For this assessment, minority populations are identified as those municipalities within the region for which the percent minority population exceeds the average for the region. There are three minority populations in the region: the Towns of Bloomfield and Windsor, and the City of Hartford. Of the three minority populations in the region, the largest percentage of minorities is in the City of Hartford. Low-income populations are identified as those municipalities within the region for which the percent of the population living in poverty exceeds 25%. The U. S. Census Bureau characterizes persons in poverty as those whose income is less than a "statistical poverty threshold." For the 1990 census, this threshold was based on a 1989 income of $12,500 per household. The only low-income population in the region is the City of Hartford.

6.3 Environmental Justice Assessment

In Chapter 5 of this Environmental Impact Statement, a review was made of human health effects and environmental impacts associated with the three alternatives under consideration. Caretaking and dismantlement activities present no significant health effects and do not constitute reasonably foreseeable adverse impacts to the regional population. The largest potential health effect, while still small, results from occupational radiation exposure to the dismantlement workers, who do not, for purposes of Executive Order 12898, comprise a low-income or minority community.

The number of potential injuries and fatalities as a result of transportation and/or occupational accidents is very small for any of the alternatives. The latent fatal cancer risk for workers and the public resulting from caretaking/dismantlement activities and from the transportation of radioactive materials off-site is very small. The prompt dismantlement and the deferred dismantlement alternatives could allow for the eventual unrestricted release of the Windsor Site. The latent fatal cancer risk from exposure to residual radioactivity levels in the soil below the established release limit is very small.

Socioeconomic impact, in terms of jobs lost to the region, would not be distinguishable for any of the alternatives. The prompt dismantlement alternative would maintain Windsor Site staffing at current level (about 150) for a period of approximately two years, after which it would be further reduced. The no action alternative would result in a more immediate staffing reduction. The deferred dismantlement alternative would also result in an immediate staffing reduction, but with a temporary rehiring after the caretaking period. For any of the alternatives, the job reductions should be absorbed readily into the regional economy.

None of the alternatives would result in the disturbance of undeveloped land or the addition of land to the Windsor Site. Caretaking and S1C Prototype reactor plant dismantlement activities would be confined within the Windsor Site property boundary and would not adversely impact any subsistence consumption of fish, game, or native plants in the region. Liquid and gaseous discharges resulting from caretaking and dismantlement activities would be controlled to maintain water quality and air quality. The aesthetic character of the area surrounding the Windsor Site would remain unchanged.

6.4 Conclusion

None of the alternatives analyzed would result in disproportionately high and adverse environmental or health effects on any particular segment of the population, including minority and low-income populations. Accordingly, none of the alternatives for disposal of the S1C Prototype reactor plant present an environmental justice concern.

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REFERENCES

1-1 Department of Energy, DOE/EIS-0203-F, *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement*, April 1995

2-1 Knolls Atomic Power Laboratory, KAPL-4753, *Windsor Site Environmental Summary Report*, June 1993.

3-1 Code of Federal Regulations, Title 40, Part 61, Subpart H. *National Emission Standards for Emission of Radionuclides Other than Radon from Department of Energy Facilities*

3-2 Welt Katzen, A. U.S. Environmental Protection Agency, Consent Agreement and Order Regarding Site Treatment Plan for Mixed Waste Generated at Knolls Atomic Power Laboratory in Windsor, Connecticut, Letter dated October 12, 1995 to A.R.Seepo, Schenectady Naval Reactors Office.

3-3 U.S. Department of Energy Order 5400.5, *Radiation Protection of the Public and the Environment.*

3-4 Nuclear Regulatory Commission, *Radiological Criteria for Decommissioning; Proposed Rule*, Federal Register Volume 59, Number 161, Pages 43200 - 43232.

3-5 Environmental Protection Agency, *Radiation Site Cleanup Regulation*, preliminary working draft for staff and public comment, dated March 16, 1995.

3-6 Nuclear Regulatory Commission, NUREG/CR-5884, *Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station*, October 1993.

4-1 Town of Windsor Planning Department, Town of Windsor Zoning Map, 1993.

4-2 U.S. Department of the Interior, National Wetlands Inventory, Windsor Locks, Connecticut. 1980

4-3 Dowhan, J.J. and Craig, R.J., *Rare and Endangered Species of Connecticut and Their Habitats. Connecticut Department of Environmental Protection*, Natural Resources Center, Report of Investigations No. 6., 1976.

4-4 McKay, Dawn M., Connecticut Department of Environmental Protection, Natural Resources Center, Letter dated April 11, 1994 to John J. Minno (Knolls Atomic Power Laboratory).

4-5 Fuss & O'Neill, Inc., *Hydrogeologic Assessment Combustion Engineering, Inc. Process Development Unit*, September 1982.

4-6 Connecticut Department of Environmental Protection, *Water Quality Classifications Map of Connecticut*, compiled by James E. Murphy, Water Compliance Unit, CTDEP, Connecticut Natural Resources Atlas Series, 1987.

4-7 Connecticut Department of Environmental Protection, Water Management Bureau, Water Quality Standards. 1992.

4-8 Ryder, R.B., Thomas, M.P. and Weiss L.A., USGS, *Water Resource Inventory of Connecticut, Part 7, Upper Connecticut River Basin*, Connecticut Water Resources Bulletin No. 24. 1981.

4-9 Federal Emergency Management Agency, National Flood Insurance Program, Flood Insurance Rate Map, Town of Windsor Connecticut - Hartford County, Panel 5 of 10; Rev. September 29, 1986.

4-10 Knolls Atomic Power Laboratory, *Environmental Monitoring Report*, calendar year 1994, KAPL-4812.

4-11 *Climates of the States*, Gale Research Co., Detroit, Michigan, 1985.

4-12 *1992 Weather Almanac*, Gale Research Co., Detroit, Michigan.

4-13 Schnabel, R.W. and Eric, J.H., USGS, *Bedrock Geologic Map of the Windsor Locks Quadrangle, Hartford County, Connecticut*, USGS Map GQ-388, 1964.

4-14 U.S. Department of the Interior, *Engineering Geology of the Northeast Corridor, Bedrock Geology*, U.S. Geological Survey, Washington, D.C., 1967.

4-15 U.S. Department of Commerce, National Oceanic and Atmospheric Administration and U.S. Department of the Interior, Geological Service *Earthquake History of the United States*, Boulder, Colorado, 1982.

4-16 Town of Windsor Plan of Development, 1991.

4-17 Maddox, D., Connecticut Historical Commission, Letter dated July 19, 1995 to S. C. Gonzalez, Knolls Atomic Power Laboratory

4-18 Bellantoni, N.F., Office of State Archaeology, Connecticut State Museum of Natural History, University of Connecticut, Letter dated May 24, 1995 to S. C. Gonzalez, Knolls Atomic Power Laboratory.

4-19 1995 World Almanac. Occupational Illnesses, by Industry and Type of Illness; Fatal Occupational Injuries.

4-20 Department of Energy, DOE/EH-0507, Occupational Injury and Property Damage Summary, January - December 1994.

4-21 National Safety Council, Accident Facts, 1995 Edition.

4-22 Argonne National Laboratory, *Derivation of Guidelines for Cobalt-60, Nickel-63, and Uranium Residual Radioactive Material in Soil at the Combustion Engineering Site, Windsor, Connecticut*, March 1996 (Draft).

4-23 Connecticut Department of Environmental Protection, Letter dated October 3, 1995, from the Bureau of Water Management to Storm Water Permittees.

4-24 Town of Windsor Zoning Regulations. Effective November 8, 1985, as amended through October 19, 1995.

4-25 Shacklette, H.T. and J.G. Boergen, *Element Concentrations in Soils and Other Surficial Materials of the Conterminous United States,* United States Geological Survey Professional Paper 1270, 1984.

4-26 United States Department of Agriculture, Soil Conservation Service, Soil Survey Series 1958, No. 14, *Soil Survey, Hartford County, Connecticut*, Sheet Number 20, February 1962.

4-27 Metzler, K.J. and K. Rozsa, Connecticut Department of Environmental Protection, *Soil Catenas of Connecticut*, Hartford, Connecticut, April 1986.

4-28 Abelquist, E.W., Oak Ridge Institute for Science and Education, *Designation Survey Addendum Report, Combustion Engineering Site, Windsor Connecticut*, Oak Ridge, Tennessee, July 1996

4-29 Department of Energy, DOE/NR-8303, *An Aerial Radiological Survey of the Natural Background Radiation Over Windsor Locks, Connecticut and Surrounding Area*, and Survey Report Addendum, October 1983.

5-1 International Commission on Radiological Protection, *1990 Recommendations of the International Commission on Radiological Protection*, ICRP Publication 60, Annals of the ICRP, Vol. 21, No. 1-3, New York: Pergamon Press. 1991.

5-2 Department of Energy, DOE/EIS-0217, *Savannah River Site Waste Management Final Environmental Impact Statement*, July 1995.

5-3 National Academy of Sciences - National Research Council, *Health Effects of Exposure to Low Levels of Ionizing Radiation, BIER V*, Report of the Committee on the Biological Effects of Ionizing Radiations, 1990.

5-4 U.S. Nuclear Regulatory Commission, *Final Environmental Statement on the Transportation of Material by Air and Other Modes*, NUREG-0170, December 1977.

GLOSSARY

Defueling	The complete removal of all nuclear fuel from the reactor plant.
Dose	The amount of radiation received (in Rem or millirem).
Dose Rate	The radiation dose per unit time (in Rem per hour or millirem per hour).
Half-life	The time required for a radioactive substance to lose 50 percent of its activity by decay.
Hazardous Waste	The Resource Conservation and Recovery Act (40 CFR Part 261) defines Hazardous Waste as a waste that is listed on one of Environmental Protection Agency's hazardous waste lists or meets one of four hazardous characteristics of ignitability, corrosivity, reactivity, or toxicity.
Long-Lived Isotope	A radionuclide with a long half-life.
Particulate Matter	Any material, except water in uncombined form, that is or has been airborne and exists as a liquid or a solid at standard conditions.
Person-Rem	The total radiation dose received by all of the individuals in a specific group over a specific period of time or during a specified work effort.
PM-10	Particulate matter with an aerodynamic diameter less than or equal to a nominal 10 micrometers as measured by a reference method based on 40 CFR Part 50 Appendix J (Reference Method for Determination of PM 10 in the Atmosphere) and designated in accordance with 40 CFR Part 53 (Ambient Air Monitoring Reference and Equivalent Methods).
Radiation	Radiation is energy in the form of waves (rays) or particles that is emitted by unstable atoms during disintegration.
Radioactivity	The process of spontaneous decay or disintegration of an unstable nucleus of an atom; usually accompanied by the emission of ionizing radiation.

Radiological Exposure	Refer to "Dose."
Radionuclide	Atoms that exhibit radioactive properties. Standard practice for naming radionuclides is to use the name or atomic symbol of an element followed by its atomic weight (for example, cobalt-60, a radionuclide of cobalt).
Record of Decision	A public document that records the final decision(s) concerning a proposed action. The Record of Decision is based on information and technical analysis generated during the decision making process, which takes into consideration public comments and community concerns.
Rem	Rem (Roentgen Equivalent Man) is a unit of radiation that relates energy deposited to biological damage. (1 Rem = 1000 millirem).
Shielding	Materials, usually concrete, water, and lead, placed around radioactive material to protect personnel from radiation exposure.
Type B Shipping Container	A container designed to retain its containment and shielding integrity under both normal transportation conditions and the hypothetical accident test conditions of 10 CFR Part 71 (Packaging and Transportation of Radioactive Material).

APPENDIX A

RADIOACTIVE SOURCES AND HEALTH EFFECTS

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TABLE OF CONTENTS

LIST OF FIGURES

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APPENDIX A

RADIOACTIVE SOURCES AND HEALTH EFFECTS

This appendix describes the sources and types of radiation encountered in the Naval Reactors program. Health effects resulting from radiation exposure are also presented.

A.1 Background Radiation

People have always lived in a sea of natural background radiation. Background radiation is as much a part of the earth's environment as the light and heat from the sun's rays. There are four principal sources of natural background radiation:

- cosmic radiation from the sun and outer space,
- terrestrial radiation from the natural radioactivity in soil and rocks,
- radiation from radon and its decay products, and
- internal radiation from the naturally radioactive elements that are part of our bodies.

The unit of effective dose equivalent is the rem. The rem is relatively large compared with the level of doses received from natural background radiation or projected as a result of releases of radioactivity to the environment. The millirem, which is one thousandth of a rem, is frequently used instead of the rem. The National Council on Radiation Protection and Measurements estimates that the average member of the population of the United States receives an annual effective dose equivalent of approximately 300 millirem from natural background radiation (Reference A-1). This is composed of approximately 28 millirem from cosmic radiation, 28 millirem from terrestrial radiation, 39 millirem from radioactivity within the body and 200 millirem from inhaled radon and its decay products. The cosmic radiation component varies from 26 millirem at sea level to 50 millirem in Denver (at 1600 meters). The terrestrial component varies from 16 millirem on the Atlantic and Gulf coastal plain to 63 millirem in the Rocky Mountains. The dose from inhaled radon and its decay products is the most variable. The average cosmic and terrestrial natural background radiation level measured in the vicinity of the Windsor Site is approximately 70 millirem per year.

In addition to natural background radiation, people are also exposed to manmade sources of radiation, such as medical and dental x-rays. The average radiation dose from these sources is about 53 millirem per year. Other manmade sources include consumer products, such as color television sets. An individual's radiation exposure from color television averages 0.3 millirem per year. An airplane trip results in increased radiation exposure. A round-trip flight between Los Angeles and New York results in a dose of about 5 millirem.

Background fission-product radioactivity also exists in the environment, primarily due to atmospheric nuclear weapons testing. Although the level is very low, these fission products are routinely detected in air, food and water when analyzed with extremely sensitive instruments and techniques.

A.2 Uranium Fission

A brief description of how the reactor plant produces energy will help explain the origins of its radioactivity. The fuel in a reactor contains enriched uranium sealed within a metal cladding. Uranium is one of the few materials capable of producing heat in a self-sustaining chain reaction. When a neutron causes a uranium atom to fission, the uranium nucleus is split apart producing atoms of lower atomic number called fission products. See Figure A-1. Some of the fission products produced by the nuclear reaction in the fuel are highly radioactive. When formed, the fission products initially move apart at very high speeds. However, fission products only travel a few thousandths of an inch before they are stopped within the fuel cladding. As the fission product movement is stopped, the kinetic energy of the fission products is converted to heat. The heat from the fuel is transferred via the reactor coolant into a steam generator which generates non-radioactive steam. The steam is used to drive propulsion plant equipment. Figure A-2 shows a simplified schematic of the reactor plant.

Naval fuel is designed, constructed and tested to ensure it will contain the radioactive fission products within the fuel itself. The materials used in Naval nuclear fuel assemblies are highly corrosion-resistant and highly radiation-resistant. As a result, the fuel assemblies are very strong and have a very high integrity. During normal reactor operation, there is no fission product release from the fuel.

Besides fission products, the nuclear reaction in the fuel also produces neutrons. During reactor operation, most of the neutrons produced are absorbed within the fuel and continue the chain reaction. However, some of the neutrons escape from the fuel. Most of the neutrons which escape from the fuel are absorbed in the walls of the reactor pressure vessel or the shielding immediately surrounding it. The remaining neutrons which escape from the fuel interact with other materials within the reactor compartment, which become activated, or radioactive.

Reactor plant components are constructed from many different materials. During normal reactor operations, trace amounts of corrosion and wear products are generated from piping system components and carried in the reactor coolant. As the reactor coolant circulates past the fuel, some of the corrosion and wear products also can absorb neutrons and become radioactive materials. A portion of the corrosion and wear products is removed from the coolant by a purification system. The portion that is not filtered out redeposits throughout the reactor piping systems or stays in the coolant.

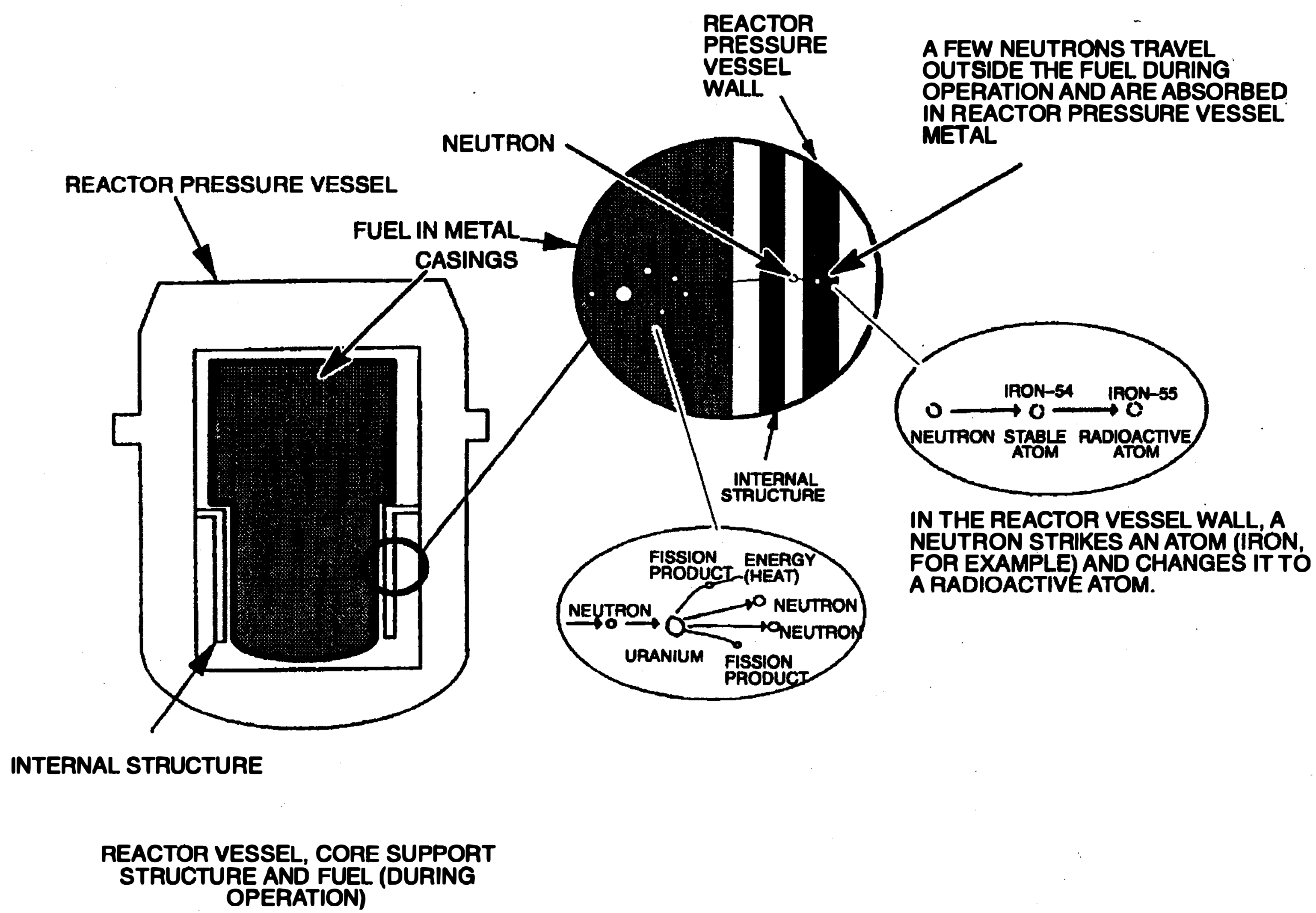

Figure A-1: Neutron and Fission Products From Uranium Fission

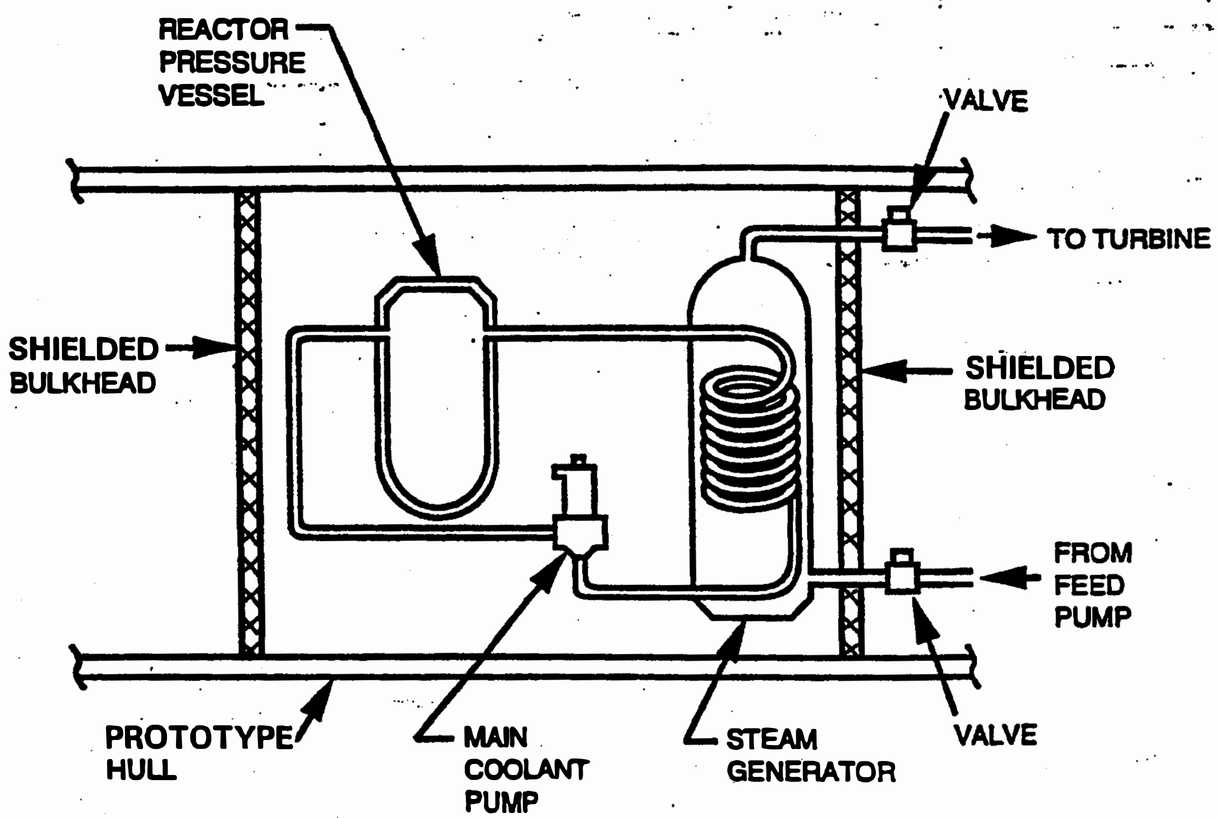

Figure A-2: Schematic of Nuclear Propulsion Plant

A.3 Radioactivation and Decay

Absorption of a neutron in the nucleus of a non-radioactive atom can produce a chemically identical radioactive atom (radionuclide). The process by which a material becomes radioactive from exposure to nuclear particles, such as neutrons, is known as activation or radioactivation. A large percentage of the radioactivity present in a defueled nuclear reactor is from activated metal. More than 99% of the remaining radioactivity in the defueled S1C Prototype reactor plant is an inseparable part of the metal components. Radioactive atoms in activated metal can only be released from the base material by the slow process of corrosion. The remaining radioactivity is comprised of the activated corrosion and wear products left from reactor operations. Release of the activated corrosion and wear products to the environment is prevented by maintaining the reactor compartment and reactor systems sealed.

The process by which radioactive atoms transform into non-radioactive atoms is known as radioactive decay. Typical particles and rays emitted during decay include alpha and beta particles, and gamma rays. Alpha radiation consists of small, positively charged particles of low penetrating power that can be stopped by a sheet of paper. Beta radiation consists of negatively charged particles that are smaller than alpha particles but are generally more penetrating and may require up to an inch of wood or other light material to be stopped. The gamma ray is an energy emission like an x-ray. Gamma rays have great penetrating power but are stopped by up to several feet of concrete or several inches of lead. In the defueled reactor plant, the most prevalent types of radiation are beta particles and gamma radiation.

Alpha particles, beta particles, and gamma rays are emitted in various combinations and energies. Each radionuclide emits a unique combination of radiations. Radionuclides may be identified by measuring the type, relative amounts, and energy of the radiations emitted. Measurement of half-life and chemical properties may also be used to help identify radionuclides. Half-life is a measure of the rate of radioactive decay. It is the time required for one-half of the atoms of a radioactive material to decay to another nuclear form.

Figure A-3 illustrates an example of the activation and radioactive decay processes. The nucleus of a non-radioactive (stable) iron atom contains a total of 54 particles, iron-54. When a non-radioactive iron atom absorbs a neutron, the nucleus contains 55 particles and is transformed to the iron-55 isotope. Iron-55 is radioactive. By releasing energy in the form of radiation, iron-55 eventually decays into manganese-55, which is non-radioactive.

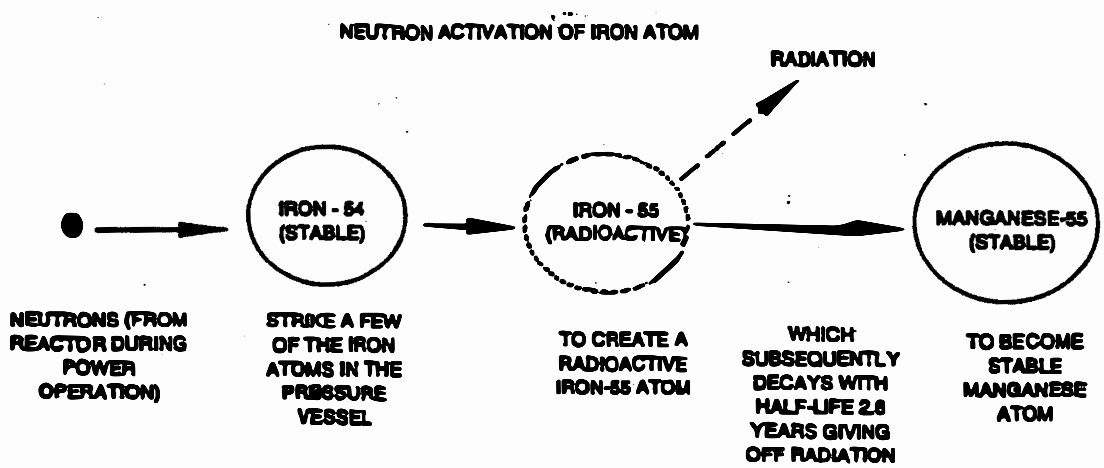

Figure A-3: Capture Neutrons in Iron of Pressure Vessel Walls

The curie (Ci) is the common unit used for expressing the magnitude of radioactive decay in a sample containing radioactive material. Specifically, the curie is that amount of radioactivity equal to 3.7×10^{10} (37 billion) disintegrations per second. For environmental monitoring purposes, the curie is usually too large a unit to work with conveniently and is broken down into smaller values such as the microcurie (μCi), which is one millionth of a curie (10^{-6} curie) and the picocurie (pCi), which is one trillionth of a curie (10^{-12} curie). The typical radium dial wrist watch has about one microcurie of radium on the dial. The average person has about 100,000 picocuries of naturally occurring potassium-40 in his body. Typical soil and sediment samples contain about one picocurie of natural uranium per gram.

A.4 Health Effects

Body tissue can be damaged if enough energy from radiation is absorbed. The amount of energy absorbed by body tissue during radiation exposure is called absorbed dose. Studies of populations exposed to radiation have been performed to develop numerical estimates of the risk of radiation exposure. These risk estimates are useful in addressing the question of how hazardous radiation exposure is, and evaluating and setting radiation protection standards.

The most recent risk estimates were prepared in 1988 and 1990 by the United Nations Scientific Committee on the Effects of Atomic Radiation (Reference A-2), and the National Academy of Sciences - National Research Council Advisory Committee on the Biological Effects of Ionizing-Radiation (Reference A-3), respectively. These estimates were based on the use of new models for predicting risk, revised dose estimates for survivors of the Hiroshima and Nagasaki atomic bombs, and additional data on the cancer experience by both atomic bomb survivors and persons exposed to radiation for medical purposes. The risk estimate for radiation-induced cancer derived from these most recent analyses can be briefly summarized as follows:

In a group of 10,000 workers in the U.S., a total of about 2,000 (20 percent) will normally die of cancer. If each of the 10,000 received over his or her career an additional one rem of radiation exposure, an estimated 4 additional cancer deaths (0.04 percent) might occur. Therefore, the average worker's lifetime risk of cancer has been increased nominally from 20 percent to 20.04 percent. This risk estimate was extrapolated from estimates applicable to high doses and dose rates, and probably overstates the true lifetime risk at low doses and dose rates. In an assessment of this uncertainty, the National Academy of Sciences pointed out that "the possibility that there may be no risks from exposures comparable to external natural background radiation cannot be ruled out" (Reference A-3).

The health risk conversion factors used in this evaluation are taken from the International Commission on Radiation Protection, Reference A-4, which specifies 0.0005 latent fatal cancers per person-rem of exposure to the general public and 0.0004 latent fatal cancers per person-rem to workers. Risk factors are lower for workers than for the general public because occupational exposures do not have to account for children. These risk factors are consistent with the most recent risk estimates for radiation exposure (References A-2 and A-3).

A.5 References

A-1 National Council on Radiation Protection and Measurements Report 93, *Ionizing Radiation Exposure of the Population of the United States*, September 1987

A-2 United Nations Scientific Committee on the Effects of Atomic Radiation, *Sources, Effects and Risks of Ionizing Radiation*, 1988

A-3 National Academy of Sciences-National Research Council, *Health Effects of Exposure to Low Levels of Ionizing Radiation, BEIR V*, Report of the Committee on the Biological Effects of Ionizing Radiations, 1990

A-4 International Commission on Radiological Protection (ICRP), *1990 Recommendations of the International Commission on Radiological Protection*, ICRP Publication 60, Annals of the ICRP, Vol. 21, No. 1-3, New York: Pergamon Press, 1991

APPENDIX B

ANALYSIS OF NON-TRANSPORTATION RELATED IMPACTS

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TABLE OF CONTENTS

LIST OF TABLES

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APPENDIX B

ANALYSIS OF NON-TRANSPORTATION RELATED IMPACTS

This appendix presents estimated environmental consequences, event probabilities, and risk (a product of probability and consequence) for both facility activities and postulated accident scenarios related to the disposal of the S1C Prototype reactor plant. Facility activities and accident scenarios are evaluated to estimate the effects of potential releases of radioactive material and toxic chemicals to the environment. The results of these analyses are presented in terms of predicted health effects to facility workers and to the general population. Effects on the environment are also presented, based on the amount of land which could be impacted due to postulated accidents. Analysis results are presented for each of the three alternatives being considered for the disposal of the S1C Prototype reactor plant - Prompt Dismantlement, Deferred Dismantlement, and No Action.

B.1 Basis of Radiological Impact Analyses for Facility Activities

B.1.1 Reactor Plant Conditions

The S1C Prototype reactor plant is defueled, however, the remaining reactor plant piping systems and components are still radioactive. The S1C Prototype hull provides a shielded, containment structure for the reactor plant. The reactor plant systems and components are in a safe, stable condition (de-energized and drained) within the hull structure.

B.1.1.1 Caretaking Activities

The No Action and the Deferred Dismantlement alternatives include a 30-year caretaking period. During the caretaking period, the S1C Prototype reactor plant would be periodically monitored. Periodic radiological surveys of the reactor plant would be performed as part of a comprehensive environmental monitoring program to be maintained during the caretaking period. This monitoring program would be a continuation of the current monitoring program at the Windsor Site and would involve air sampling, the continuous monitoring of radiation levels at Site perimeter locations and at off-site locations, and the routine collection and analysis of water samples, sediment samples, and fish. Periodic monitoring would verify reactor plant integrity and expected radiological conditions. To further ensure that reactor plant system and component integrities are maintained, a heating and dehumidifying system would be installed. Airflow from the reactor compartment would be exhausted through a controlled exhaust system containing high efficiency particulate air filters to the environment. This analysis evaluates the radiological impacts of direct radiation exposure to workers and the general population during the caretaking period. In addition, radiological impacts from potential airborne releases during the caretaking period, including potential accidents, are estimated.

B.1.1.2 Dismantlement Activities

Dismantlement activities for the Prompt and Deferred Dismantlement alternatives are similar. The dismantlement work includes removal of reactor plant piping systems and components, disassembly of the prototype hull and preparations for shipment. Dismantlement activities would be performed using proven radiological control methods to prevent the spread of any contamination. The radiological exposures associated with dismantlement work will be significantly lower for the Deferred Dismantlement alternative due primarily to cobalt-60 radioactivity decay. Evaluations of the impacts associated with transportation of dismantled materials from the reactor plant are discussed in Appendix C. This analysis evaluates the radiological impacts of direct radiation exposure to workers and the general population during dismantlement activities. Radiological impacts from potential releases to the atmosphere during dismantlement activities, including potential accidents, are also estimated.

B.1.2 Selection of Facility Accidents for Detailed Evaluation

In selecting accidents to include in detailed analyses, several variables were considered. Variables included risk of an accident, probability of occurrence and consequences. Risk is defined as the product of the probability of occurrence of the accident multiplied by the consequence of the accident. This analysis only evaluates accidents that contribute substantially to risk.

B.1.2.1 Accident Probability Considerations

Accidents were categorized into three types as either Abnormal Events, Design Basis Accidents, or Beyond Design Basis Accidents. These categories are characterized by their probability of occurrence as described below.

Abnormal Events

Abnormal Events are unplanned or improper events which result in little or no consequence. Abnormal events include industrial accidents and accidents during facility activities such as spills of radioactive liquids, or exposure to direct radiation due to improper placement of shielding. The occurrence of these unplanned events has been anticipated and mitigative procedures are in place which immediately detect and eliminate the events and limit the effects of these events on individuals. As a result, there is little hazard to the general population from these events. Such events are considered to occur in the probability range of 1 to 10^{-3} per year. The probability referred to here includes the probability the event occurs multiplied by other probabilities required for the consequences. For accidents included in this range, results are presented for the 95% meteorological condition.

Design Basis Accident Range

Accidents which have a probability of occurrence in the range of 10^{-3} to 10^{-6} per year are included in the range called the Design Basis Accident Range. The terminology "design basis accident," which normally refers to facilities to be constructed, also includes the "evaluation" basis accident which applies to existing facilities. For accidents included in this range, results are presented for the 95% meteorological condition.

Beyond Design Basis Accidents

This range includes accidents which are less likely to occur than the design basis accidents but which may have very large or catastrophic consequences. Accidents included in this range typically have a total probability of occurrence in the range of 10^{-6} to 10^{-7} per year. For accidents included in this range, results are presented for the 95% meteorological condition. Accidents which are less likely than 10^{-7} per year typically are not discussed since it is expected they would not contribute in any substantial way to the risk.

B.1.2.2 Accident Consequence Considerations

Only accidents which could reasonably be assumed to result in severe consequences were evaluated. Severe consequences include a significant release of radioactive material to the environment or a significant increase in radiation levels. Variables affecting accident severity include: dispersibility of the radioactive materials involved, the mechanism that causes the release of radioactive materials from the facility, and the conditions affecting off-site dispersion of the released materials. Initiating events for severe consequence accidents can include natural phenomena (earthquakes, volcanic activity, tornadoes, hurricanes, and other natural events) and human induced events (human error, equipment failures, fires, explosions, plane crashes, transportation accidents, and terrorism). The resulting exposure pathways from accidental releases of radioactive materials include direct exposure to radiation, inhalation of radioactive materials, or ingestion of radioactive materials.

Most accident events, such as procedure violations, equipment failures, and spills affect very limited areas and the environmental consequences are insignificant. For example, such events may not involve enough radioactive material or radiation to result in a significant release to the environment or result in a meaningful increase in radiation levels. Despite the higher frequency of occurrence, the very low severity of these events results in very low risk. Accidents involving small releases and affecting small areas were eliminated from further evaluation.

B.1.2.3 Accidents Selected for S1C Prototype Dismantlement Evaluation

Based on the selection process described above, several accident scenarios were developed for further detailed analysis. The following four hypothetical accident scenarios are considered to be more severe than all other reasonably foreseeable accidents.

- A large component drop, resulting in a breach of the component,
- mechanical damage of a component due to a wind-driven missile,
- an airplane crash into the reactor plant, resulting in the breach of several components,
- and a high efficiency particulate air filter fire.

B.1.3 Analysis Methods for Evaluation of Radiation Exposure

B.1.3.1 Computer Programs and Meteorological Modeling

The radiation exposures to the general population, dismantlement workers and specific individuals were calculated using the following computer programs and meteorological modeling. Radiation exposures were calculated for incident-free facility activities and for hypothetical accidents conditions. The calculation methods are consistent with similar evaluations by the International Commission on Radiological Protection (References B-1 and B-2).

GENII

GENII (Reference B-4) was used in the facility activity evaluations of long-term exposure to released radioactive contaminants. This program was developed at Pacific Northwest Laboratory by Battelle Memorial Institute. The program incorporates internal dosimetry models recommended by the International Commission on Radiological Protection in Publication 26 (Reference B-1) and Publication 30 (Reference B-2). The code uses averaged meteorological conditions to evaluate long-term effects of airborne releases.

RSAC-5

The Radiological Safety Analysis Computer Program, RSAC-5 (Reference B-5), was used to calculate the consequences of the release of radionuclides to the atmosphere. Calculations include potential radiation exposures to maximally exposed individuals or population groups via inhalation, ingestion, exposure to radionuclides deposited on the ground surface, immersion in airborne radioactive material, and radiation from a cloud of radioactive material. This program was developed by Westinghouse Idaho Nuclear Co., Inc., for the Department of Energy - Idaho Operations Office. RSAC-5 meteorological modeling capabilities include Gaussian plume dispersion for Pascal-Gifford conditions. RSAC-5 release scenario modeling allows reduction of radionuclides by chemical group or element and calculates decay and buildup during transport through operations, facilities, and the environment. It allows the amount of each nuclide from a nuclear event to be designated individually or to be calculated internally by the code. It also models the effect of filters or other cleanup systems.

SPAN4

SPAN4 (Reference B-6) was used to calculate the direct radiation levels. The computer code was developed by the Bettis Atomic Power Laboratory for use in Naval Nuclear Propulsion Program work. The SPAN 4 program models the effects of distance from a radiation source on resulting radiation exposure. Estimated exposures are derived by mathematical integration over specified areas.

WATER RELEASE

WATER RELEASE, an unpublished computer code developed by the Bettis Atomic Power Laboratory, was used to calculate exposures to humans arising from radionuclides which have been introduced into water in the vicinity of the radiological facilities. There are two processes by which radionuclides might enter water - via liquid discharge or via airborne discharges. The WATER RELEASE computer code models the resulting effects on humans from exposure to the assumed released radioactivity. Exposure to such releases can be received in several different pathways. Examples of pathways that the program can analyze include consumption of affected water, consumption of affected foods, and immersion (for example, swimming). The total exposure to the general population or individual is the resultant sum of the exposures from each pathway analyzed.

Meteorological Modeling

Meteorological data used in the analyses were obtained from the Support Center for Regulatory Air Models bulletin board system. The Support Center for Regulatory Air Models is an organization within the Environmental Protection Agency, Office of Air Quality Planning and Standards. Bulletin board data files for surface meteorological conditions consist of data acquired from the National Climatic Data Center. Meteorological data from the Bradley International Airport, for the years 1988 - 1992, were used in this evaluation.

Data and computer programs from the Support Center for Regulatory Air Models were used to develop meteorological data in the Stability Array format. The Stability Array format is a joint frequency distribution of six wind speed intervals, 16 wind directions, and six stability categories. The Stability Array meteorology data were used to calculate the 95% meteorological conditions for the accident analyses. The 95% condition represents the meteorological conditions which could produce the highest calculated exposures. This is defined as that condition which is not exceeded more than

5% of the time or is the worst combination of weather stability class and wind speed. Each of these conditions is evaluated for 16 wind directions. The Stability Array data were also reformatted for use in the GENII program calculations.

B.1.3.2 Radiation Exposure Categories

Radiation exposures were calculated for the following categories of individuals for the three disposal alternatives and hypothetical accidents:

Workers

Workers are individuals who would be directly involved in performing the actual dismantlement or caretaking activities. The occupational exposures were calculated based on actual radiation survey data obtained after the reactor plant was defueled and drained. Occupational exposures in person-rem were estimated for specific dismantlement and packaging tasks. Similar estimates were calculated for workers who would perform surveillance tours or security duties during a caretaking period.

Maximally Exposed Off-Site Individual

The maximally exposed off-site individual is a hypothetical individual living at the Windsor Site boundary receiving the maximum exposure. No evacuation of this individual is assumed to occur.

Population

The population consists of the actual number of people, based on 1990 United States Census data, living within a 50-mile radius of the Windsor Site. The total number of people living within a 50-mile radius of the Windsor Site is 3,425,290. The population distribution in 16 compass directions, and various radial intervals from the prototype location is included in Chapter 4, Table 4-3, of the Environmental Impact Statement.

B.1.3.3 Health Effect Evaluations

Table B-1 lists the health risk conversion factors used in this appendix. Health effects are calculated based on the radiation exposure results from incident-free facility activities and hypothetical accidents. The risk factors used for calculations of health effects are taken from Publication 60 of the International Commission on Radiological Protection (Reference B-3). Health risk conversion factors are weighted higher for the general population to account for longer life expectancies of children compared to adult workers.

Table B-1: Health Risk Conversion Factors for Ionizing Radiation Exposure

Effect	Nuclide	Risk Factor (probability per Rem)	
		Worker	General Population
Fatal cancer (all organs)	All	4.0×10^{-4}	5.0×10^{-4}
Weighted non-fatal cancer [1]	All	8.0×10^{-5}	1.0×10^{-4}
Weighted genetic effects [1]	All	8.0×10^{-5}	1.3×10^{-4}
Weighted total effects [1]	All	5.6×10^{-4}	7.3×10^{-4}

1. In determining a means of assessing health effects from radiation exposure, the International Commission on Radiological Protection has developed a weighting method for non-fatal cancers and genetic effects to obtain a total weighted effect, or "health detriment."

B.1.3.4 Evaluation of Impacted Areas for Hypothetical Accident Analyses

The impacted area following a facility accident was determined for each accident scenario. The impacted area was defined as that area in which the plume deposited radioactive material to such a degree that an individual standing on the boundary of the fallout area would receive approximately 0.01 millirem per hour of exposure above background. If this individual spends 24 hours a day at this location, that person would receive an additional 88 millirem per year from direct radiation from radioactivity deposited on the ground. This is within the Nuclear Regulatory Commission dose limit of 100 millirem per year for individual members of the general population (10 CFR Part 20, Standards for Protection Against Radiation).

To best characterize the affected areas for each casualty, a typical 50% meteorology (Pasquill-Gifford Class D, wind speed 10 mph) was chosen. The 95% worst case meteorology was used when calculating exposure and risk to workers and the general population. Computer modeling results (RSAC-5) for ground surface dose were interpolated to determine the distance downwind where the centerline dose had dropped to approximately 88 millirem per year based on 24 hours per day exposure. For the wind class chosen, the plume remains within a single 22.5-degree sector. The area affected by the plume is conservatively assumed to be the entire sector contaminated to the calculated downwind distance rather than the narrower plume profile. Use of a typical 50% meteorology is also a conservative assumption for the footprint evaluation of a tornado generated wind-driven missile accident. Stormy, windy conditions would disperse any release sufficiently such that no location would have a dose greater than 88 millirem per year.

Table B-2 shows impacted area dimensions (footprints) for each hypothetical accident scenario. Although the radioactive plume resulting from an accident would be contained within a single wind chart sector, the direction of the wind is unknown. Since the accidents occur over a short duration of time, calculations assumed no changes in the general wind direction. Impacts were evaluated in each of the sixteen directions around the facility out to a distance equaling the footprint length. Table B-3 describes secondary effects of hypothetical facility accidents.

Table B-2: Footprint Estimates for Hypothetical Facility Accidents

Accident Scenario	Footprint Length (meters)	Footprint Area (acres)
Airplane Crash (Prompt Dismantlement)	170	1.4
All Other Accidents for All Alternatives	<100	<1.0

Table B-3: Secondary Impacts of Hypothetical Facility Accidents

Topic	Impact
Surrounding Environment	Due to the small size of the Windsor Site, contamination would extend beyond the Windsor Site boundary.
Biotic Resources Including Endangered Species	Plants and animals on the Windsor Site and around the Windsor Site will experience no long-term impacts. An accident would not result in the extinction or adversely affect potential for survival of any endangered species.
Water Resources	The water used for drinking and industrial purposes is monitored and use may be temporarily suspended during cleanup operations. Some recreational activities may also be temporarily suspended. No enduring impacts are expected.
Economic Impacts	A small number of individuals may experience temporary job loss due to temporary restrictions on farming, fishing and other support activities near the facility during cleanup operations. Some costs would also be incurred for the actual cleanup operation.
Land Use	Access to some areas may be temporarily restricted until cleanup is completed. The total area restricted would be no greater than the areas identified in Table B-2.

B.1.3.5 Estimated Exposure Times and Mitigative Measures Following Hypothetical Facility Accidents

Accident analysis calculations take no credit for any preventive or mitigative actions that would limit exposure to members of the general population who are assumed to reside in close proximity to the Windsor Site. Radiation dose calculations for the maximally exposed off-site individual (individual who lives nearest the Windsor Site boundary) assume exposure to the entire contaminated plume as it travels downwind from the accident site. Calculations assume no action is taken to prevent these people from continuing their normal day-to-day routines or changing their food sources. The general population is assumed to spend approximately 30% of the day within their homes or other buildings. Since buildings and homes provide shielding, general population annual exposure to ground surface radiation was reduced by 30%.

Facility workers all undergo training to take quick, decisive action during a casualty. In the event of an accident, workers would quickly evacuate the affected area and assemble in an area upwind of the affected area. Analyses assumed that workers would move to an area 100 meters from the affected area. Analyses conservatively assumed that workers would receive exposure to the released radioactivity for a total of five minutes. Worker exposures were calculated for the direct radiation and inhalation pathways. Exposures due to ingestion of contaminated food were not specifically calculated for workers since workers would not eat following the accident.

Table B-4 provides the individual exposure times utilized in the hypothetical facility accident analyses.

Table B-4: Estimated Exposure Times Following a Hypothetical Facility Accident

Exposure Pathway	Worker	Maximally Exposed Off-Site Individual
Plume	5 minutes	100% of release time
Fallout on Ground Surface	5 minutes	0.7 year
Food Ingestion	None	1 year

B.1.3.6 Modeling Assumptions for Hypothetical Facility Accident Evaluations

Unless stated otherwise, the following post-accident modeling assumptions were used when performing airborne radioactivity release calculations with the RSAC-5 computer program. In most cases, these conditions are the default conditions in the computer program.

Meteorological Data

- Wind speed, direction, and Pasquill stability are taken from 95% meteorology.
- The release is calculated as occurring at ground level (0 meters).
- Mixing layer height is 400 meters (1320 feet). Airborne materials freely diffuse in the atmosphere near ground level in what is known as the mixing depth. A stable layer exists above the mixing depth which restricts vertical diffusion.
- Wet deposition is zero (no rain occurs to accelerate deposition and reduce the area affected).
- Dry deposition of the cloud is modeled. During movement of the radioactive plume, a fraction of the plume is deposited on the ground due to gravitational forces and becomes available for exposure by ground surface radiation and ingestion.
- The quantity of deposited radioactive material is proportional to the material size and speed. The following dry deposition velocities (meters per second) were used:
  solids = 0.001 halogens = 0.01 noble gases = 0.0 cesium = 0.001
- If radioactive releases occur through a stack, then additional plume dispersion can be accounted for by calculating a jet plume rise. In this analysis, jet plume rise is ignored.
- When released gases have a heat content, the plume can disperse more quickly. In this calculation, buoyant plume effects are ignored.

Inhalation Data

- Breathing rates are 3.33×10^{-4} cubic meters per second for workers and 2.66×10^{-4} cubic meters per second for people at the site boundary and beyond.
- Particle size is 1.0 micron.
- The internal exposure period is 50 years for individual organs and tissues which have radionuclides committed.
- Exposure to the entire plume for the general population. Exposure to the plume for workers is discussed in Section B.1.3.5.
- Inhalation exposure factors based on Reference B-2.

Ground Surface Exposure

- Exposed to contaminated soil for one year for the general population.
- Building shielding factor is 0.7 which exposes the individual to contaminated soil for 16 hours a day.

Ingestion Data

- The following dietary consumption rates were used:
  177 kilograms of stored vegetables (produce) per year
  18.3 kilograms of fresh (leafy) vegetables per year
  94 kilograms of meat per year
  112 liters of milk per year
- 10% of the food consumed is assumed to be locally grown (such as in a person's garden) and contaminated by the accident.

B.2 Radiological Analysis Results - Incident-Free Facility Activities

B.2.1 Facility Activities

The purpose of this analysis is to determine the hypothetical health effects on workers and the general population due to incident-free facility activities associated with disposal of the S1C Prototype reactor plant. Unique source terms were used for each alternative for the evaluation of facility activities. Windsor Site-specific meteorological and population data were used. For facility activities, the radiation dose evaluation addresses workers, the maximally exposed off-site individual, and the general population.

B.2.1.1 Source Terms

The radioactive material release source terms for the analysis are based on conservative calculations of expected releases. For the no action alternative and the first 30 years of the deferred dismantlement alternative, the S1C Prototype reactor compartment will be maintained in a heated and dry condition. The systems and components will be closed and sealed such that none of the contamination will be available for release to the environment. None of the contaminated plant systems will be vented. Therefore, the routine airborne release was calculated based on a minimum detectable airborne activity level of 2×10^{-14} microcuries per milliliter and the expected volume of air which would flow through the reactor compartment. For both dismantlement alternatives, the airborne release source term was selected based on data from typical reactor servicing ventilation systems. The ventilation systems have high efficiency particulate air filters installed and have a 99.95% efficiency for removal of potential airborne particulate radioactivity. The source term was derived from the radiation levels measured on typical air filters installed in ventilation systems used during maintenance work on radioactive systems.

Table B-5 lists the radioactive nuclides and the estimated amounts of radioactivity that result in at least 99% of the possible exposure due to airborne releases to the environment.

Table B-5: Source Terms for Facility Activities Releases

NUCLIDE	RADIOACTIVITY DISCHARGED (Curies per year) [1]		
	No Action	Prompt Dismantlement	Deferred Dismantlement [2]
Co-60	4.5×10^{-8}	2.1×10^{-7}	4.1×10^{-9}
C-14	1.5×10^{-6}	7.1×10^{-6}	7.1×10^{-6}
Fe-55	5.5×10^{-8}	2.6×10^{-7}	1.3×10^{-10}
Ni-63	2.2×10^{-8}	1.0×10^{-7}	8.4×10^{-8}
Sr-90	2.8×10^{-11}	1.3×10^{-10}	6.3×10^{-11}
Nb-93M	9.6×10^{-10}	4.5×10^{-9}	1.2×10^{-9}
Nb-94	1.5×10^{-11}	7.1×10^{-11}	7.1×10^{-11}
Cs-137	2.8×10^{-11}	1.3×10^{-10}	6.5×10^{-11}
Pu-238	1.8×10^{-13}	8.6×10^{-13}	6.8×10^{-13}
Pu-239	3.0×10^{-14}	1.4×10^{-13}	1.4×10^{-13}
Pu-240	1.9×10^{-14}	8.9×10^{-14}	8.9×10^{-14}
Pu-241	6.3×10^{-12}	2.9×10^{-11}	6.9×10^{-12}
Am-241	2.6×10^{-13}	1.2×10^{-12}	1.2×10^{-12}

1. Ventilation system discharges are estimated for the first year of the prompt and no action alternatives and the thirty-first year of the deferred dismantlement alternative (first year of deferred dismantlement operations). The no action source term is used for the 30-year caretaking period prior to deferred dismantlement. Listed radionuclides are from activated corrosion products which could be released.

2. The radionuclides listed for deferred dismantlement were derived based on prompt dismantlement data and individual nuclide decay rates.

B.2.1.2 Facility Activities Analysis Results

Table B-6 contains the detailed analysis results for radiation exposure from facility activities, through various pathways, assuming no accidents occur. Since each of the alternatives represent different lengths of time, the results presented are cumulative exposures and effects. For the no action alternative, the exposures represent the total received over a 30-year caretaking period. For the deferred dismantlement alternative, the exposures represent the total received over the 30-year caretaking period plus the expected exposures during the 2-year dismantlement period. For the prompt dismantlement alternative, the exposures represent the total received during the 2-year dismantlement period. The health effects represent those expected based on the total radiation exposure received over the period of interest.

For occupational exposure to workers, the largest doses result from the prompt dismantlement alternative. The deferred dismantlement dose reflects the radioactive decay of cobalt-60 over a 30-year period and also includes exposure received by workers during the 30-year caretaking period.

Exposure to the general population is essentially the same for the no action and deferred dismantlement alternatives because the time durations are approximately the same. The radiation dose from facility activities to the general population during the prompt dismantlement alternative is significantly lower because of the short two-year duration.

Table B-6: Exposure Results for Incident-Free Facility Activities

		No Action	Deferred Dismantlement	Prompt Dismantlement
Workers (Occupational Exposure) [1]	Collective Dose (person-rem)	2.1	3.9 to 5.7	94 to 188
	Risk of Latent Fatal Cancers	8.4×10^{-4}	1.6×10^{-3} to 2.3×10^{-3}	3.8×10^{-2} to 7.5×10^{-2}
Maximally Exposed Off-Site Individual [2]	Dose (rem)	1.0×10^{-2}	1.0×10^{-2}	2.6×10^{-3}
	Risk of Latent Fatal Cancer	5.1×10^{-6}	5.2×10^{-6}	1.3×10^{-6}
Population [3]	Collective Dose (person-rem)	3.2×10^{-2}	3.2×10^{-2}	8.1×10^{-3}
	Risk of Latent Fatal Cancers	1.6×10^{-5}	1.6×10^{-5}	4.0×10^{-6}

1. The collective dose values for workers represent the occupational exposure for each alternative based on estimates of worker staffing levels, time in or near the reactor compartment, and general area dose rates. General area dose rates were based on actual radiation survey data measured after the reactor plant was defueled and drained. The larger values for the prompt and deferred dismantlement ranges represent estimates based on preliminary plans. The lower values for the prompt and deferred dismantlement ranges reflect experience that detailed work planning typically results in additional exposure reductions. Individual worker exposure would be limited to 2 rem per year.

2. The dose values for the Maximally Exposed Off-Site Individual represent conservative estimates for a hypothetical individual who resides at the boundary of the Windsor Site for the duration of the respective alternative.

3. The collective dose values for the Population represent conservative estimates of cumulative exposure to all members of the general population living within a 50-mile radius of the Windsor Site.

B.3 Radiological Analysis Results - Hypothetical Facility Accidents

B.3.1 Component Drop Accident

B.3.1.1 Description of Conditions

During dismantlement of the S1C Prototype reactor plant, many large components and portions of piping systems will be disassembled and removed from the facility. Because of strict verbatim procedure compliance rules, proven safe rigging practices, and required crane maintenance, coupled with independent oversight, a drop of one of these large components at a Naval nuclear facility is not considered a credible accident. However, a drop accident of one of these components will be considered based on using commercial industry failure probabilities (References B-9, B-10, and B-11). Since these components contain some radioactive materials in the form of corrosion products, it is postulated that some portion of these corrosion products could become released into the environment.

B.3.1.2 Source Term

The source term for the component drop accident is based on the following considerations. The corrosion product activity on the component is the best estimate deposition on non-reactor core wetted surfaces. The steam generator is the component with the most corrosion deposits since it has the largest internal surface area. The impact associated with the component drop accident is assumed to loosen 33% of the corrosion products adhering to the steam generator surfaces. Of this loose activity, 10% is assumed to be released to the environment as an airborne contaminant. Thus, a total release of 3.3% of the corrosion products from the steam generator is assumed in the airborne dose analysis.

The following amounts of radionuclides from activated corrosion products could be released into the environment. This listing includes radionuclides that result in at least 99% of the possible exposure.

Nuclide	Deferred Dismantlement (Curies)	Prompt Dismantlement (Curies)
Co-60	1.1×10^{-3}	5.7×10^{-2}
C-14	9.7×10^{-4}	9.7×10^{-4}
Fe-55	3.5×10^{-5}	7.0×10^{-2}
Ni-63	2.3×10^{-2}	2.8×10^{-2}
Sr-90	1.7×10^{-5}	3.5×10^{-5}
Nb-93M	3.4×10^{-4}	1.2×10^{-3}
Nb-94	1.9×10^{-5}	1.9×10^{-5}
Cs-137	1.8×10^{-5}	3.5×10^{-5}
Pu-238	1.9×10^{-7}	2.4×10^{-7}
Pu-239	3.9×10^{-8}	3.9×10^{-8}
Pu-240	2.4×10^{-8}	2.4×10^{-8}
Pu-241	1.9×10^{-6}	8.0×10^{-6}
Am-241	3.2×10^{-7}	3.4×10^{-7}

B.3.1.3 Radiological Analysis Results - Component Drop Accident

Tables B-7 and B-8 summarize the health risks to individuals and the general population that might result from the hypothetical drop of a component during dismantlement activities. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defmed as the number of fatal cancers times the probability of occurrence. The results are presented for the design basis accident with 95% meteorology. Section B.1.3.4 discussed the affected area size. The probability of crane failure is in the range of 1×10^{-5} to 1.5×10^{-4} per operation (References B-10 and B-11). For this evaluation, a probability of 1×10^{-4} per operation was used. The probability of failure leading to an uncontrolled lowering of a component due to failure of either electrical or mechanical backups, given the initial crane failure, is in the range of 10^{-2} to 10^{-1} per failure (Reference B-12). For this evaluation, the smaller probability of 1×10^{-2} per failure was used since it is already conservatively assumed that any uncontrolled lowering would result in component damage and a release to the environment. This results in a probability of dropping a single component of 1×10^{-6} per operation. Since there will be about four lifts per year during the dismantlement period, a probability of 4×10^{-6} per year was used in the risk assessment.

Table B-7: Individual Exposure Results - Hypothetical Component Drop Accident

	Deferred Dismantlement		Prompt Dismantlement	
	Dose (rem)	Risk of Latent Fatal Cancer	Dose (rem)	Risk of Latent Fatal Cancer
Individual Worker	1.8×10^{-3}	7.3×10^{-7}	3.4×10^{-2}	1.4×10^{-5}
Maximally Exposed Off-Site Individual	1.2×10^{-2}	6.2×10^{-6}	4.4×10^{-1}	2.2×10^{-4}

Table B-8: General Population Exposure Results for Hypothetical Component Drop Accident

	Deferred Dismantlement	Prompt Dismantlement
Collective Dose Within 50-Mile Radius (person-rem)	3.0	110
Number of Fatal Cancers	1.5×10^{-3}	5.4×10^{-2}
Probability per Year of Accident Occurring	4.0×10^{-6}	4.0×10^{-6}
Risk per Year of Single Latent Fatal Cancer	6.0×10^{-9}	2.2×10^{-7}

B.3.2 Wind-Driven Missile Accident

B.3.2.1 Description of Conditions

During dismantlement activities, portions of the plant and large components will be vulnerable to wind-driven missiles while being prepared for shipment off-site. There is a small probability that one of these large components could be damaged during this period. Since these components contain some radioactive materials in the form of corrosion products, it is postulated that some portion of these particles could become released into the environment. During the caretaking period, the thick steel hull of the reactor compartment will provide protection from any naturally caused wind-driven missiles.

B.3.2.2 Source Term

The source term for the wind-driven missile accident is based on the following considerations. The best estimate corrosion product activity is used as the basis of the source term. The steam generator is assumed to be the component which is hit by the wind-driven missile because it has the highest inventory of activity. The impact associated with the missile strike is assumed to loosen 33% of the corrosion products adhering to the steam generator surfaces. Of this loose activity, 1% is assumed to be released to the environment as an airborne contaminant. Thus, a total release of 0.33% of the corrosion products from the steam generator is assumed in the airborne dose analysis.

The following amounts of radionuclides from activated corrosion products could be released into the environment. This listing includes radionuclides that result in at least 99% of the possible exposure.

Nuclide	Deferred Dismantlement (Curies)	Prompt Dismantlement (Curies)
Co-60	1.1×10^{-4}	5.7×10^{-3}
C-14	9.7×10^{-5}	9.7×10^{-5}
Fe-55	3.5×10^{-6}	7.0×10^{-3}
Ni-63	2.3×10^{-3}	2.8×10^{-3}
Sr-90	1.7×10^{-6}	3.5×10^{-6}
Nb-93M	3.4×10^{-5}	1.2×10^{-4}
Nb-94	1.9×10^{-6}	1.9×10^{-6}
Cs-137	1.8×10^{-6}	3.5×10^{-6}
Pu-238	1.9×10^{-8}	2.4×10^{-8}
Pu-239	3.9×10^{-9}	3.9×10^{-9}
Pu-240	2.4×10^{-9}	2.4×10^{-9}
Pu-241	1.9×10^{-7}	8.0×10^{-7}
Am-241	3.2×10^{-8}	3.4×10^{-8}

B.3.2.3 Radiological Analysis Results - Wind-Driven Missile Accident

Tables B-9 and B-10 summarize the health risks to individuals and the general population that might result from the hypothetical wind-driven missile accident. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defmed as the number of fatal cancers times the probability of occurrence. The results are presented for the design basis accident with 95% meteorology. Section B.1.3.4 discussed the affected area size. The probability of occurrence of a tornado was obtained using the data in the Atomic Energy Commission document WASH-1300 (Reference B-13). These analyses assumed the probability of a tornado occurrence in the continental United States is 10^{-3} per year per square mile. The probability of generation of a missile sufficient to cause a release of radioactive material is assumed to be 1.0. The probability of the missile hitting the target component was assumed to be 10^{-2}. This is considered to be very conservative due to the small size of the component and the limited amount of time it is in a vulnerable position. Thus, the probability of a wind-driven missile accident occurrence of 1×10^{-5} per year was used in the risk assessment.

Table B-9: Individual Exposure Results - Hypothetical Wind-Driven Missile Accident

	Deferred Dismantlement		Prompt Dismantlement	
	Dose (rem)	Risk of Latent Fatal Cancer	Dose (rem)	Risk of Latent Fatal Cancer
Individual Worker	1.8×10^{-4}	7.3×10^{-8}	3.4×10^{-3}	1.4×10^{-6}
Maximally Exposed Off-Site Individual	1.2×10^{-3}	6.2×10^{-7}	4.4×10^{-2}	2.2×10^{-5}

Table B-10: General Population Exposure Results for Hypothetical Wind-Driven Missile Accident

	Deferred Dismantlement	Prompt Dismantlement
Collective Dose Within 50-mile Radius (person-rem)	3.0×10^{-1}	11
Number of Fatal Cancers	1.5×10^{-4}	5.4×10^{-3}
Probability per Year of Accident Occurring	1.0×10^{-5}	1.0×10^{-5}
Risk per Year of Single Latent Fatal Cancer	1.5×10^{-9}	5.4×10^{-8}

B.3.3 Airplane Crash Accident

B.3.3.1 Description of Conditions

During the dismantlement of the S1C Prototype reactor plant, the components and piping systems could be damaged if a large airplane crashed into the facility while the reactor compartment is partly removed for the dismantlement work. Due to component damage and a subsequent fire, the corrosion products in several components could be released into the environment. It is expected that during any caretaking period, the thick steel hull of the S1C Prototype reactor plant will absorb most of the energy from a plane crash; therefore, the consequences and risks will be smaller than the values shown below for the prompt dismantlement alternative.

B.3.3.2 Source Term

The source term for the airplane crash accident is based on the following considerations. The best estimate corrosion product activity in the reactor plant is used as the basis of the source term. A combination of the three components having the highest corrosion product inventory is assumed to be damaged by the plane crash. The impact of the crash is assumed to loosen 33% of the corrosion products adhering to major component surfaces. Of this loose activity, 50% is assumed to be released to the environment as an airborne contaminant. Thus, a total release of 16.5% of the corrosion product activity in the damaged components is assumed in the airborne dose analysis.

The following amounts of radionuclides from activated corrosion products could be released into the environment. This listing includes radionuclides that result in at least 99% of the possible exposure.

Nuclide	Deferred Dismantlement (Curies)	Prompt Dismantlement (Curies)
Co-60	1.2×10^{-2}	6.2×10^{-1}
C-14	1.1×10^{-2}	1.1×10^{-2}
Fe-55	3.8×10^{-4}	7.6×10^{-1}
Ni-63	2.5×10^{-1}	3.1×10^{-1}
Sr-90	1.9×10^{-4}	3.8×10^{-4}
Nb-93M	3.7×10^{-3}	1.3×10^{-2}
Nb-94	2.1×10^{-4}	2.1×10^{-4}
Cs-137	1.9×10^{-4}	3.9×10^{-4}
Pu-238	2.0×10^{-6}	2.6×10^{-6}
Pu-239	4.2×10^{-7}	4.2×10^{-7}
Pu-240	2.6×10^{-7}	2.6×10^{-7}
Pu-241	2.1×10^{-5}	8.7×10^{-5}
Am-241	3.5×10^{-6}	3.7×10^{-6}

B.3.3.3 Airplane Crash Probability

The probability of an airplane crashing into the S1C Prototype reactor plant was evaluated. The method outlined in the U.S. Nuclear Regulatory Commission Standard Review Plan for Aircraft Hazards (Reference B-8) was used to predict the crash probability.

The aircraft crash probability analysis was based on three major aircraft categories - commercial, military, and general aviation. Two general types of flight sources were addressed in this crash probability - operations at nearby airports (such as landings and takeoffs), and operations in nearby airways (in-flight travel). Airports which met one of the following criteria were considered:

- has a runway used by commercial or military flights that is at least partially located within ten miles of the S1C Prototype,
- has a runway used for general aviation that is at least partially located within five miles of the S1C Prototype, or
- the centerline of the defined airway is located within ten miles of the S1C Prototype.

The probability per year for an airplane crash into the S1C Prototype was calculated by adding the individual crash scenario probabilities for the flight sources listed above (combined takeoff, landing, and in-flight crash probabilities). The airport operation crash probability was determined by multiplying the probability per square mile of a crash for each aircraft by the number of landings and takeoffs for each aircraft along each runway, and by the effective target area for each type of aircraft. The airway crash probability was calculated by multiplying the in-flight crash rate per mile by the number of flights per year along the airway by the effective target area (S1C Prototype reactor compartment) divided by the width of the airway.

The Windsor Site is located approximately 3.5 miles from Bradley International Airport and approximately four miles from a grass runway at the Simsbury Airport. Table B-11 summarizes the airport traffic information. The Windsor Site is located approximately two miles from a high altitude route between Boston and New York which has 21,535 large commercial and large military aircraft flights per year.

Table B-11: Airport Landings and Takeoffs per Year

Airport	Large Civilian Aircraft Landings and Takeoffs	Military A-10 Aircraft Landings and Takeoffs	General Aviation Landings and Takeoffs	Air Taxi Landings and Takeoffs
Bradley International	60,042	5,894	45,804	50,116
Simsbury	0	0	1,094	0

Results of this calculation indicate the probability of an airplane crashing into the S1C Prototype reactor compartment is 6.6×10^{-7} per year.

B.3.3.4 Radiological Analysis Results - Airplane Crash Accident

Tables B-12 and B-13 summarize the health risk to individuals and the general population that might result from the hypothetical airplane crash accident. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence. The results are presented for the design basis accident with 95% meteorology. Section B.1.3.4 discussed the affected area size.

Table B-12: Individual Exposure Results for Hypothetical Airplane Crash

	Deferred Dismantlement		Prompt Dismantlement	
	Dose (rem)	Risk of Latent Fatal Cancer	Dose (rem)	Risk of Latent Fatal Cancer
Individual Worker	2.0×10^{-2}	7.9×10^{-6}	3.7×10^{-1}	1.5×10^{-4}
Maximally Exposed Off-Site Individual	1.3×10^{-1}	6.7×10^{-5}	4.8	2.4×10^{-3}

Table B-13: General Population Exposure Results for Hypothetical Airplane Crash

	Deferred Dismantlement	Prompt Dismantlement
Collective Dose Within 50-mile Radius (person-rem)	33	1200
Number of Fatal Cancers	1.7×10^{-2}	5.8×10^{-1}
Probability per Year of Accident Occurring	6.6×10^{-7}	6.6×10^{-7}
Risk per Year of Single Latent Fatal Cancer	1.1×10^{-8}	3.8×10^{-7}

B.3.4 High Efficiency Particulate Air Filter Fire Accident

B.3.4.1 Description of Conditions

In this hypothetical accident scenario, a fire in a bank of high efficiency particulate air filters is postulated. This accident could be initiated by the ignition of a flammable mixture released upstream of the system by an external, unrelated fire that spreads to the system. Although the risks associated with this accident are relatively minor, it was analyzed to bound the higher-probability, lower-consequence type accident category. The airborne release fractions associated with this accident were conservatively chosen so that a high efficiency particulate air filter failure by crushing or impact was also bounded.

B.3.4.2 Source Term

A maximum inventory of activity in a high efficiency particulate air filter bank is assumed to be present in the filters at the time of the fire. This activity will only occur after an extended period of operation and is based on previous experience during reactor plant maintenance which included work on the open reactor plant and the release of corrosion products during operations. For the caretaking period, the activity in the filters is based on the minimum detectable activity being released. The hypothetical fire is assumed to spread to the filters from another source and is assumed to release 1% of the radioactive materials from the filter to the environment. The release is relatively small because the filters are constructed of material containing glass fibers which would melt during a fire and trap the radioactive particles in the medium. Measurements from experiments show that 0.01% of the material in the filter could be released during a fire (Reference B-15). The use of 1% is conservatively selected for this analysis.

The following amounts of radionuclides from activated corrosion products could be released into the environment. This listing includes radionuclides that result in at least 99% of the possible exposure. For the no action and prompt dismantlement alternatives, the fire is assumed to occur at the end of the first year. For the deferred dismantlement alternative, the fire is assumed to occur at the end of the thirty-first year (the end of the first year of the dismantlement period after a thirty-year caretaking period). The cumulative impact from the deferred dismantlement alternative incorporates the 30-year caretaking period which is represented by the no action values listed below.

Nuclide	Deferred Dismantlement (Curies)	Prompt Dismantlement (Curies)	No Action (Curies)
Co-60	1.6×10^{-7}	8.4×10^{-6}	1.8×10^{-6}
C-14	1.4×10^{-7}	1.4×10^{-7}	3.0×10^{-8}
Fe-55	5.1×10^{-9}	1.0×10^{-5}	2.2×10^{-6}
Ni-63	3.4×10^{-6}	4.1×10^{-6}	8.9×10^{-7}
Sr-90	2.5×10^{-9}	5.2×10^{-9}	1.1×10^{-9}
Nb-93M	4.9×10^{-8}	1.8×10^{-7}	3.8×10^{-8}
Nb-94	2.8×10^{-9}	2.8×10^{-9}	6.1×10^{-10}
Cs-137	2.6×10^{-9}	5.2×10^{-9}	1.1×10^{-9}
Pu-238	2.7×10^{-11}	3.4×10^{-11}	7.4×10^{-12}
Pu-239	5.7×10^{-12}	5.7×10^{-12}	1.2×10^{-12}
Pu-240	3.5×10^{-12}	3.6×10^{-12}	7.6×10^{-13}
Pu-241	2.8×10^{-10}	1.2×10^{-9}	2.5×10^{-10}
Am-241	4.7×10^{-11}	4.9×10^{-11}	1.1×10^{-11}

B.3.4.3 Radiological Analysis Results - High Efficiency Particulate Air Filter Fire Accident

Tables B-14 and B-15 summarize the health risks to individuals and the general population that might result from the hypothetical high efficiency particulate air filter fire accident. The number of fatal cancers would be expected to occur over a 50-year period. "Risk" is defined as the number of fatal cancers times the probability of occurrence. The results are presented for the design basis accident with 95% meteorology. Section B.1.3.4 discussed the affected area size. The probability of a chemical fire is 5×10^{-3} per year (Reference B-14). The probability of high efficiency particulate air filter fires is considered to be less than a chemical fire since chemicals would not be stored in the immediate vicinity of the high efficiency particulate air filter system and high efficiency particulate air filters are not volatile or explosive. It is estimated that the probability for an existing fire to spread to the high efficiency particulate air filters is less than 10^{-1}. Thus, the probability of occurrence of an event leading to a high efficiency particulate air filter fire is estimated at 5×10^{-4} per year. This probability is applied to all alternatives but is very conservative for the no action alternative because no flammable materials will be stored in the reactor plant.

Table B-14: Individual Exposure Results - Hypothetical High Efficiency Particulate Air Filter Fire Accident

	Prompt Dismantlement	Deferred Dismantlement	No Action
Individual Worker Dose (Rem)	5.0×10^{-6}	2.7×10^{-7}	1.1×10^{-6}
Risk of Latent Fatal Cancer	2.0×10^{-9}	1.1×10^{-10}	4.3×10^{-10}
Maximally Exposed Off-Site Individual Dose (Rem)	6.5×10^{-5}	1.8×10^{-6}	1.4×10^{-5}
Risk of Latent Fatal Cancer	3.2×10^{-8}	9.0×10^{-10}	6.9×10^{-9}

Table B-15: General Population Exposure Results - Hypothetical High Efficiency Particulate Air Filter Fire Accident

	Prompt Dismantlement	Deferred Dismantlement	No Action
Collective Dose Within 50-Mile Radius (person-rem)	1.6×10^{-2}	4.4×10^{-4}	3.4×10^{-3}
Number of Fatal Cancers	7.9×10^{-6}	2.2×10^{-7}	1.7×10^{-6}
Probability per Year of Accident Occurring	5.0×10^{-4}	5.0×10^{-4}	5.0×10^{-4}
Risk per Year of Single Latent Fatal Cancer	4.0×10^{-9}	1.1×10^{-10}	8.5×10^{-10}

B.3.5 Cumulative Radiological Impacts to the General Population from Hypothetical Facility Accidents

Table B-16 presents cumulative risk results to the general population living within a 50-mile radius of the Windsor Site for the specific hypothetical accidents that were evaluated in this analysis. For each accident type, the cumulative results are based on the annual risk multiplied by the duration of the alternative.

Table B-16: Cumulative Radiological Impacts Risk to the General Population from Hypothetical Accidents

	Prompt Dismantlement (risk of latent fatal cancer)	Deferred Dismantlement (risk of latent fatal cancer)	No Action (risk of latent fatal cancer)
Component Drop [3]			
Annual Risk (Table B-8)	2.2×10^{-7}	6.0×10^{-9}	not applicable [1]
Cumulative Risk	4.4×10^{-7}	1.2×10^{-8}	
Wind-Driven Missile [3]			
Annual Risk (Table B-10)	5.4×10^{-8}	1.5×10^{-9}	not analyzed
Cumulative Risk	1.1×10^{-7}	3.0×10^{-9}	in detail [2]
Airplane Crash [3]			
Annual Risk (Table B-13)	3.8×10^{-7}	1.1×10^{-8}	not analyzed
Cumulative Risk	7.6×10^{-7}	2.2×10^{-8}	in detail [2]
High Efficiency Particulate Air Filter Fire [4]			
Annual Risk (Table B-15)	4.0×10^{-9}	1.1×10^{-10}	8.5×10^{-10}
Cumulative Risk [5]	8.0×10^{-9}	2.6×10^{-8}	2.6×10^{-8}

1. Lifting of components would not occur during the no action alternative. The thick steel hull of the reactor compartment would remain in place during the caretaking period.

2. The steel hull would absorb most of the energy from any airplane crashes or wind-driven missiles. No radiological releases to the environment would be expected for the hypothetical wind-driven missile accident. Potential consequences and risks would be smaller than the values shown for the hypothetical airplane crash accident - prompt dismantlement.

3. The cumulative risks associated with the component drop, wind-driven missile and airplane crash accidents are based on a two-year period for the dismantlement activities during the prompt and deferred alternatives.

4. The cumulative risks associated with the high efficiency particulate air filter fire accident is based on a two-year period for prompt dismantlement, thirty-two-year period (caretaking and dismantlement) for deferred dismantlement, and for a thirty-year period for the no action alternative.

5. The cumulative risks from a high efficiency particulate air filter fire are essentially the same for the deferred dismantlement and the no action alternatives. The annual risk for deferred dismantlement activities commence after a thirty-year caretaking period. The cumulative risk during the two-year deferred dismantlement period are small and do not add significantly to the cumulative risk during the caretaking period.

B.3.6 Summary of Analysis Uncertainties

The calculations in this Environmental Impact Statement have generally been performed in such a way that the estimates of risk provided are unlikely to be exceeded during dismantlement activities, caretaking activities, or in the event of an accident. For dismantlement activities, the results of radiation surveys and monitoring of similar operations provide realistic source terms, which, when combined with conservative estimates of the effects of radiation, produce estimates of risk which are very unlikely to be exceeded.

The analyses of hypothetical accidents provide more opportunities for uncertainty, primarily because the calculations must be based on sequences of events and models of effects which have not occurred. The models have attempted to provide estimates of the probabilities, source terms, pathways for dispersion and exposure, and the effects on human health and the environment which are as realistic as possible. However, in many cases, the very low probability of the accidents postulated has required the use of models or values for input which produce estimates of consequences and risks which are higher than would actually occur because of the desire to provide results which will not be exceeded. The risks presented in this appendix are believed to be at least 10 to 100 times larger than what would actually occur. Despite the use of conservative analytical methods, the risks for all of the alternatives are very small. Since the resulting risks are so small, the significance of any uncertainty in analysis parameters is greatly reduced.

The use of conservative analyses does not create a bias in this Environmental Impact Statement since all of the alternatives have been evaluated using the same methods and data. The potential impacts of each alternative can be fairly compared on the same basis.

B.4 Nonradiological Analysis Results - Hypothetical Facility Accidents

For the purposes of comparison with other risks associated with dismantlement and caretaking activities, an evaluation of a diesel fuel oil fire was analyzed in detail. This accident was selected based on the potential duration of the accident, the potential size of the affected area, and the combustion products that would result. A hypothetical accident scenario involving a fire in the Windsor Site's hazardous waste container storage area was considered but eliminated from detailed analysis. No hazardous waste would be left at Windsor Site during a caretaking period. The quantity of hazardous waste which would be stored at any time during dismantlement is expected to be maintained small by routine disposal shipments. The Windsor Site's hazardous waste container storage area is constructed and operated such that the overall environmental risks, including risks from accidents, are insignificant.

B.4.1 Fire Involving Diesel Fuel

B.4.1.1 Accident Description

This analysis assumed that during dismantlement operations, a diesel fuel oil storage tank could be temporarily located near a work area for refueling power equipment and on-site vehicles. A catastrophic failure of a temporarily located diesel fuel storage tank was postulated to occur. This could result in the spilling of the entire quantity of diesel fuel (500 gallons) and a subsequent fire. The airborne release of toxic chemicals resulting from the fire was evaluated with respect to the maximally exposed off-site individual. The maximally exposed off-site individual is assumed to be an individual located 100 meters from the fire.

B.4.1.2 Computer Model Used to Estimate Toxic Chemical Exposures

The Emergency Prediction Information Computer Code (EPIcode™) was used for estimating airborne concentrations resulting from most releases of toxic chemicals (Reference B-16). The computer code uses the well-established Gaussian Plume Model to calculate the airborne toxic chemical concentrations. The computer code database contains information on over 600 toxic substances listed by the American Conference of Governmental Industrial Hygienists. Factors such as locations of affected persons, terrain, meteorological conditions, release conditions, and characteristics of the chemical inventory are required as input parameters for calculations to determine human exposure from airborne releases of toxic chemicals.

B.4.1.3 Source Term

The material involved in this accident was diesel fuel with the fire generating the following toxic chemicals due to combustion:

- Carbon monoxide
- Oxides of nitrogen (90% nitric oxide and 10% nitrogen dioxide)
- Lead
- Sulfur dioxide.

B.4.1.4 Conditions and Key Parameters

- A 500-gallon capacity fuel oil storage tank would spill all its contents into a welded revetment made from 1/4 inch steel plate with dimensions of approximately 8 feet by 8 feet by 1 foot deep. The entire amount of diesel fuel is consumed by the fire in about 2 hours.

- The releases per gallon of fuel burned are as follows:

  Carbon monoxide = 0.34 pound
  Oxides of nitrogen = 1.58 pound
  Lead = 4.2×10^{-6} pound
  Sulfur dioxide = 0.105 pound

- The airborne release of toxic chemicals occurs at ground level.

- Standard rural terrain is used and building wake effects are not considered.

- Wind speeds and atmospheric stability classifications are based on 95% meteorology.

- The estimated concentrations are compared against the Emergency Response Planning Guideline levels 1, 2, and 3 concentration limits or alternates to determine the health impacts. Emergency Response Planning Guideline values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (Reference B-17).

B.4.1.5 Results

The airborne concentrations, averaged over the duration of each exposure, were calculated using the Emergency Prediction Information computer program for the combustion products resulting from the fire for the maximally exposed off-site individual under 95% meteorology. Table B-17 lists the downwind concentrations and corresponding Emergency Response Planning Guideline (or equivalent) values. The significance of the Emergency Response Planning Guideline (or equivalent) is explained in the footnotes to the table. Results for the diesel fuel fire indicate that the toxic chemical concentration for lead concentrations were well below Emergency Response Planning Guideline level 1 values. Carbon monoxide may exceed Emergency Response Planning Guideline level 2 value. Sulfur dioxide and oxides of nitrogen may exceed Emergency Response Planning Guideline level 3 values for the maximally exposed off-site individual.

For the on-site workers and any member of the general population that could be exposed to toxic chemicals at above Emergency Response Planning Guideline level 3 values, it is expected that actual toxic chemical exposures would be much less due to the mitigative measures that would be implemented. During dismantlement operations employees would quickly move out of the smoke from the fire. Since the area immediately surrounding the Windsor Site is uninhabited, it is unlikely that a member of the general population would be standing at the Windsor Site boundary and very unlikely that a person would remain in the smoke plume from a fire (as assumed in the analysis).

Additional information on the toxic properties for the chemicals that dominate the toxic effects is provided below.

Sulfur dioxide is a colorless and toxic gas with a pungent odor. Sulfur dioxide is an eye, skin, and mucous membrane irritant. It chiefly affects the upper respiratory tract and bronchi and at higher concentrations, sulfur dioxide causes respiratory paralysis (Reference B-18).

Nitric oxide and **nitrogen dioxide** occur together in dynamic equilibrium. Nitric oxide is a colorless gas, and nitrogen dioxide is a reddish brown gas. Both chemicals are eye, skin, and mucous membrane irritants and primarily affect the respiratory system. Exposure to 47 milligrams per cubic meter of nitrogen dioxide can cause respiratory irritation and chest pain, 93 milligrams per cubic meter can cause lung injuries, and 187 milligrams per cubic meter can be fatal (Reference B-18).

Carbon Monoxide is a colorless, odorless and toxic gas which is a product of incomplete combustion. It is a potent chemical asphyxiant capable of causing headache, nausea, fatigue, confusion, and coma when present in high concentrations.

Table B-17: Typical Chemical Concentrations of a Diesel Fuel Fire

	CHEMICAL CONCENTRATIONS (milligrams per cubic meter) - 95% METEOROLOGY				
	Sulfur Dioxide ERPG-1 0.79 ERPG-2 7.9 ERPG-3 39	Carbon Monoxide TWA 29 .1(IDLH) 172 IDLH 1720	Nitric Oxide TWA 31 .1(IDLH) * IDLH 123	Nitrogen Dioxide TWA 5.6 .1(IDLH) 9.4 IDLH 94	Lead TWA 0.15 .1(IDLH) 70 IDLH 700
Maximally exposed off-site individual (located 100 meters from the fire)	130	430	1800	200	1.6×10^{-3}

ERPG = Emergency Response Planning Guidelines

ERPG-1 = The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing other than mild, transient adverse health effects or without perceiving a clearly defined objectionable odor.

ERPG-2 = The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms which could impair an individual's ability to take protective action.

ERPG-3 = The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing life - threatening health effects.

Where ERPG values have not been derived for a toxic substance, other chemical toxicity values are substituted, as follows:

For ERPG-1, Threshold Limit Value, Time-Weighted Average (TLV-TWA) values (Reference B-19) are substituted: The TWA is the time-weighted average concentration for a normal 8-hour workday and a 40-hour workweek, to which nearly all workers may be repeatedly exposed, day after day, without adverse effect.

For ERPG-2, Level of Concern values (equal to 0.1 of Immediately Dangerous to Life or Health) are substituted: Level of Concern is defined as the concentration of a hazardous substance in air, above which there may be serious irreversible health effects or death as a result of a single exposure for a relatively short period of time (Reference B-20).

For ERPG-3, Immediately Dangerous to Life or Health (IDLH) values are substituted: IDLH is defined as the maximum concentration from which a person could escape within 30 minutes without a respirator and without experiencing any effects which would impair the ability to escape or irreversible side effects (Reference B-7).

* The .1(IDLH) level not assigned since the value (12.3) would be less than the TWA level.

B.5 Analysis of Dust Emissions

This section provides the results of an evaluation of dust emissions that could be generated during dismantlement of the defueled S1C Prototype reactor plant. None of the S1C Prototype reactor plant dismantlement activities would be expected to generate nuisance dust emissions. Examples of other Windsor Site dismantlement activities that could generate some dust emissions include demolition of buildings (generally concrete and cinder block construction), and earth moving activities (such as backfilling and surface grading for establishing a smooth final contour for the Windsor Site property).

B.5.1 Computer Modeling to Estimate Dust Emissions

Factors such as locations of affected persons, terrain, meteorological conditions, release conditions, and grain size distributions are required as input parameters for calculations to determine particulate concentrations from dust emissions during dismantlement activities. This section describes the computer model used to perform dust concentration estimates. Specific input parameters used in this analysis are summarized in Section B.5.2.

The Fugitive Dust Model computer code was used to evaluate dust emissions from dismantlement activities of the S1C Prototype reactor plant. The computer model was specifically designed for estimating dust emissions from point, line, or area sources in support of air quality evaluations (Reference B-9).

The computer model for dust is designed to work with properly prepared meteorological data in either hourly or Stability Array format. The computer model is based on the well-known Gaussian plume formulation for computing concentrations, but the model has been specifically adapted to incorporate an improved gradient transfer deposition algorithm. Emissions for each source are apportioned by the user into a series of particle size classes. A gravitational settling velocity and a deposition velocity are subsequently calculated in the computer model for each class, and dust concentrations and depositions are then calculated for locations selected by the user.

B.5.2 Conditions and Key Parameters

- Restoration area is 8.8 acres.
- An emission factor of 2.0 tons per acre-month is used.
- Grain sizes used are as follows:

Average Diameter (microns)	% of Total
1.25	3
3.75	5
7.5	15
12.5	10
20.0	67

- Meteorological conditions used are the 5-year average Stability Array data sets.
- Roughness height is 30 centimeters.
- The maximally exposed off-site individual is located 100 meters from the center of the restoration area due to the small area of the Windsor Site.

B.5.3 Results

Dust concentration was calculated for the maximally exposed off-site individual located 100 meters from the center of the restoration area. The calculated dust concentration for the maximally exposed off-site individual during dismantlement of the S1C Prototype reactor plant was 1.7 milligrams per cubic meter. When this airborne concentration is compared against the Threshold Limit Value - Time Weighted Average concentration for inhalable particulates (10 milligrams per cubic meter), it is concluded that dust emissions associated with dismantlement activities would not result in any adverse effects. The estimated levels of dust generation would not require regulation under the Clean Air Act.

REFERENCES

B-1 ICRP (International Commission on Radiological Protection), 1977, *Report of the Task Group on Reference Man,* ICRP Publication 26, International Commission on Radiological Protection, Oxford, Great Britain: Pergamon Press.

B-2 ICRP (International Commission on Radiological Protection), 1979, *Limits for Intakes of Radionuclides by Workers,* Part 1, ICRP Publication 30, International Commission on Radiological Protection, Oxford, Great Britain: Pergamon Press.

B-3 ICRP (International Commission on Radiological Protection), 1991, *The 1990 Recommendations of the ICRP,* ICRP Publication 60, Annals of the ICRP, Volume 21 (1-3), International Commission on Radiological Protection, Elmsford, New York: Pergamon Press.

B-4 Napier, B. A., R. A. Peloquin, D. L. Strenge, J. V. Ramsdell, 1988, *GENII The Hanford Environmental Radiation Dosimetry Software System,* PNL-6584, UC-600, November.

B-5 Wenzel, D. R., 1993, *The Radiological Safety Analysis Computer Program (RSAC-5),* U.S. Department of Energy Report WINCO-1123, Westinghouse Idaho Nuclear Company, Inc., Idaho Falls, Idaho, October.

B-6 Wallace, O. J., 1972, *SPAN4 A Point Kernel Computer Program for Shielding,* WAPD-TM-809(L),Volumes I and II, October.

B-7 NIOSH (National Institute for Occupational Safety and Health), 1990, *Pocket Guide to Chemical Hazards,* June.

B-8 NRC (U.S. Nuclear Regulatory Commission), 1981, Standard Review Plan 3.5.1.6, "Aircraft Hazards," NUREG-0800, July.

B-9 EPA (U.S. Environmental Protection Agency), 1992, *Users Guide for the Fugitive Dust Model (FDM),* EPA-910/9-88-202R.

B-10 EPRI (Electric Power Research Institute), 1984a, *Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment,* EPRI NP-3365, Palo Alto, California, February.

B-11 EPRI (Electric Power Research Institute), 1984b, *Surveillance of LWR Spent Fuel in Wet Storage,* EPRI NP-3765, Palo Alto, California, October.

B-12 NRC (U.S. Nuclear Regulatory Commission), 1975, *Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants,* WASH - 1400 (NUREG-75/014), Washington, D.C.

B-13 AEC (Atomic Energy Commission), 1974, *Technical Basis for Interim Regional Tornado Criteria,* WASH 1300, U.S. Atomic Energy Commission Office of Regulation, May.

B-14 Ganti, C.S. and L.M. Krasner, 1984, *Navy Fire Risk Management: A Methodology for Prioritizing Fire Protection Recommendations,* Factory Mutual Research, Norwood, Massachusetts, September.

B-15 DOE (U.S. Department of Energy), 1993, *DOE HANDBOOK: Recommended Values and Technical Bases for Airborne Release Fractions, Airborne Release Rates, and Respirable Fractions at DOE Non-Reactor Nuclear Facilities,* DOE-STD-0013-93.

B-16 Homan (Homan Associates, Inc.), 1988, *Emergency Prediction Information Computer Code (EPIcode™) Manual*, Fremont, California.

B-17 Rusch, G. M., 1993, "The History and Development of Emergency Response Planning Guidelines," *Journal of Hazardous Materials,* Volume 33, Amsterdam: Elsevier Publishers, pp. 193-202.

B-18 TOXnet, 1993, Toxicology Data Network, National Library of Medicine, U.S. Department of Health and Human Services, Bethesda, Maryland.

B-19 ACGIH (American Conference of Governmental Industrial Hygienists), 1993, *Threshold Limit Values and Biological Exposure Indices for 1993-1994.*

B-20 EPA (U.S. Environmental Protection Agency), 1987, *Technical Guidance for Hazards Analysis,* Federal Emergency Management Agency, U.S. Department of Transportation, December.

APPENDIX C

ANALYSIS OF TRANSPORTATION RELATED IMPACTS

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TABLE OF CONTENTS

LIST OF TABLES

APPENDIX C

ANALYSIS OF TRANSPORTATION RELATED IMPACTS

C.1 Background

This appendix presents an evaluation of the health risks to the public and workers from the shipment of all materials and components that would result from dismantlement of the defueled S1C Prototype reactor plant. These analyses cover the prompt and deferred dismantlement alternatives. Transportation analyses for the no action alternative are not required because there would be no dismantlement wastes generated or shipments made. Analyses were performed consistent with the methods and computer models used in the development of the Department of Energy's Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement (Reference C-1) and the Draft Environmental Impact Statement on the Disposal of Decommissioned, Defueled Cruiser, Ohio Class, and Los Angeles Class Naval Reactor Plants (Reference C-2).

C.2 Shipments Evaluated

This evaluation assumes all shipments originate at the Windsor Site located in Windsor, Connecticut. The analyses assume that nonradioactive materials would be recycled or disposed of at facilities located within an average distance of 200 kilometers from the Windsor Site. The analyses evaluated two Department of Energy destinations for low level radioactive materials - the Savannah River disposal site in the State of South Carolina and the Hanford disposal site in the State of Washington. These analyses include additional general assumptions to keep the meaning of the results simple and conservative. For example, the Savannah River disposal site and the Hanford disposal site are examined individually as the destination for all radioactive shipments. The Savannah River disposal site represents a reasonable close location and distance for transportation analyses, and the Hanford disposal site represents a reasonable but significantly more distant location. Combinations of shipping destinations, including available recycling facility locations for radioactive materials, are not examined. This is a conservative simplification because the cumulative mileage of any combination of available destinations would be less than the cumulative mileage of all shipments going cross-country to the Hanford disposal site. Actual disposal of dismantlement materials would utilize multiple shipping destinations with emphasis on recycling as much material as practical. The topic of waste management and recycling is discussed in detail in the environmental impact statement text. Table C-1 summarizes the types of packages, the transportation modes, the origin and the destinations that are analyzed for shipments of low level radioactive materials from S1C Prototype dismantlement.

Table C-1: Summary of Package Type, Transportation Mode, Origin, and Destination

PACKAGE TYPE	TRANSPORTATION MODE	ORIGIN	DESTINATION
Miscellaneous Components	Truck	Windsor Site	Savannah River Site
			Hanford Site
Pressure Vessel	Heavy Hauler	Windsor Site	Griffin Line Railhead [1]
	Rail	Griffin Line Railhead [1]	Savannah River Site
			Hanford Site
Steam Generator	Truck	Windsor Site	Savannah River Site
			Hanford Site
Pressurizer	Truck	Windsor Site	Savannah River Site
			Hanford Site

1. Alternate transportation modes that would eliminate the use of the Griffin Line railhead for shipment of the reactor pressure vessel package were also considered but eliminated from detailed evaluation. As a single package, the pressure vessel shipping package would measure approximately 18.5 feet in length and 10.5 feet in diameter and would weigh approximately 160 tons. Due to load limiting bridges and speed limitations that would result in traffic disruptions, transport of the pressure vessel package for long distances over highways was considered impractical.

Analyses assumed there would be a total of 19 shipments of miscellaneous component packages. The major components would be shipped as whole units in four individual shipments. All shipments were assumed to occur over a two-year period. In addition to the reactor pressure vessel, one additional shipment by rail may be necessary in order to ship the primary shield tank in a single large package.

C.3 General Technical Approach for Calculating Health Risks

This section describes the general approach taken to evaluate the health risks associated with the shipment of dismantled S1C Prototype reactor plant materials. First, the radiological health risks to the general population, to the transport crew and to hypothetical maximally exposed individuals are evaluated for gamma radiation emanating directly from the packages during normal (incident-free) transport conditions. Radiological health risks are reported in terms of latent fatal cancers. Next, the radiological health risks to the general population for accident scenarios are evaluated. Accidents are evaluated based on corrosion product (crud) release to the atmosphere, probability for occurrence, and accident severity. To upper bound the significance of an accident, the radiological consequences are also evaluated for hypothetical maximally exposed individuals and the general population. In conjunction with these radiological evaluations, nonradiological risks to the population are also evaluated for vehicular exhaust emissions and transportation accidents.

C.3.1 Computer Codes

Several computer codes were used in the analysis of transportation related impacts. General analyses used the RADTRAN 4 and RISKIND computer codes. Several other computer programs, such as INTERLINE, HIGHWAY, and SPAN4, were used to provide input for the RADTRAN 4 and RISKIND computer codes. Due to the simplicity of variables for calculating the risks to the maximally exposed individual in the general population during incident-free conditions, simple equations without computer modeling were sufficient for the analysis.

RADTRAN 4

The RADTRAN 4 computer code was developed by Sandia National Laboratories (References C-3 and C-4). RADTRAN 4 was used to calculate radiological risks for the general population and transportation crew for incident-free and accident risk scenarios. RADTRAN 4 was also used to calculate radiological risks for the maximally exposed individual worker for incident-free scenarios.

RISKIND

The RISKIND computer code was developed by Argonne National Laboratory (Reference C-5). RISKIND was used to calculate the maximum radiological consequences to the general population and the maximally exposed individual in the general population for postulated accident conditions.

INTERLINE

The INTERLINE computer program was developed at Oak Ridge National Laboratory (Reference C-6). The latest available version of INTERLINE was used to model conditions in the vicinity of railroad routes. The INTERLINE database consists of networks representing various competing railroad companies in the United States. The routes used in this study use the standard assumptions in the INTERLINE model which simulate the selection process that railroads would use to direct shipments of Naval reactor plant components. The code is updated periodically to reflect current track conditions and has been benchmarked against reported mileage and observations. INTERLINE also provides the weighted population densities for rural, suburban, and urban populations averaged over all states along the shipment route and the percentage of mileage traveled in each population density. The version of INTERLINE used in these analyses contains 1990 census data. The distance traveled, weighted population density, and percentage of distance in each population density are input variables in the RADTRAN 4 computer code.

HIGHWAY

The HIGHWAY computer program was developed at Oak Ridge National Laboratory (Reference C-7). The latest available version of HIGHWAY was used to model conditions in the vicinity of highway routes. The code is updated periodically as new roads are added. The routes used for this study use the standard assumptions in the highway model. Similar to the INTERLINE computer code, HIGHWAY provides the distance between the origin and destination, the weighted population densities along the route and the percentage of distance traveled in each population density, which are all input variables for the RADTRAN 4 computer code.

SPAN4

The SPAN4 computer code was developed by the Bettis Atomic Power Laboratory for use in Naval Nuclear Propulsion Program work (Reference C-8). SPAN4 was used to model the effect of distance from a radiation source on the resulting radiation exposure. Estimated exposures are derived by mathematical integration over specified areas.

C.3.2 Radiological and Nonradiological Fatality Rates

The health risk conversion factors used in this evaluation are taken from the International Commission on Radiological Protection (Reference C-9) which specified 0.0005 latent fatal cancers per person-rem for members of the public and 0.0004 latent fatal cancers per person-rem for workers. Risk factors are lower for workers than for the general population because occupational exposures do not have to account for children. These risk estimates were extrapolated from estimates applicable to high doses and dose rates and probably overstate the true lifetime risk at low doses and dose rates. In an assessment of this uncertainty, the National Academy of Sciences pointed out that "the possibility that there may be no risks from exposures comparable to external natural background radiation cannot be ruled out" (Reference C-14).

In these analyses, the radiological impacts are first expressed as the calculated total effective exposure. Exposures to the general population and transportation crew are reported as person-rem and exposures to maximally exposed individuals are reported as rem. The appropriate health risk conversion factor, above, is then applied to the calculated total exposure in order to estimate the health risks in terms of latent fatal cancers. When interpreting the results of these analyses, the health risk per person-rem of exposure to the general population is equivalent to the health risk per rem of exposure to an individual. For example, ten people in the general population receiving 0.1 rem exposure each yields the same health risk as one individual who receives one rem of exposure (10 people x 0.1 rem each = 1.0 person-rem = 1 person x 1 rem).

Nonradiological risks related to the transportation of waste and recyclable materials from dismantlement and demolition activities and from the transportation of associated materials such as fill and topsoil are also evaluated. The nonradiological risks are those resulting from vehicle exhaust emissions for incident-free transportation and fatalities resulting from transportation accidents for accident risk assessment. The nonradiological risks associated with return of transport vehicles to their points of origin are also included. Risk factors for exhaust emissions and fatality rates used in these analyses were obtained from References C-10, C-11, and C-12 and are provided in Table C-2.

Table C-2: Fatality Rates for Nonradiological Risks

	RAIL	TRUCK
Fatalities per Kilometer Due to Pollutants	1.3×10^{-7}	1.0×10^{-7}
Fatalities per Kilometer Due to Accidents	2.82×10^{-8}	5.82×10^{-8}

C.3.3 Formulas Used for Nonradiological Shipment Health Risk Calculations

The estimated fatalities during incident-free transportation of nonradiological materials are determined according to the following formula:

$$F_1 = D \times U \times R_1 \times N \times 2$$

where:

F_1 = Estimated fatalities for the total number of shipments.

D = Average distance traveled (kilometers), per package.

U = Percent of the distance traveled through urban areas, which has been conservatively estimated to be 27 percent for nonradiological shipments. (This estimate was used since the destination for all nonradiological shipments is not precisely known).

R_1 = Fatalities per kilometer due to pollutants based on Reference C-10.

N = Number of shipments.

2 = Factor which is applied for the return of the transport vehicle to its point of origin.

A summary of the variables and the estimated fatalities due to incident-free shipment of the nonradiological materials is provided in Table C-3.

Table C-3: Variables and Fatalities for Incident-Free Shipment of Nonradiological Materials

D	U	R_1	N	F_1
200 kilometers	27%	1.0×10^{-7} fatalities per kilometer	1600 shipments	1.7×10^{-2} estimated fatalities

For the shipments of nonradiological materials involving an accident, the estimated fatalities are determined according to the following formula:

$$F_2 = D \times R_2 \times N \times 2$$

where:

F_2 = Estimated fatalities for the total number of shipments.

D = Average distance traveled (kilometer), per package.

R_2 = National average truck accident fatality rate (per kilometer) based on Reference C-11.

N = Number of shipments.

2 = Factor which is applied for the return of the transport vehicle to its point of origin.

A summary of the variables and the estimated fatalities due to accidents involving shipment of the nonradiological materials is provided in Table C-4.

Table C-4: Variables and Fatalities Due to Accidents Involving Shipments of Nonradiological Materials

D	R_2	N	F_2
200 kilometers	5.82×10^{-8} fatalities per kilometer	1600 shipments	3.7×10^{-2} estimated fatalities

C.4 Technical Approach for Assessing Incident-Free Radioactive Shipments

C.4.1 General Population Exposure and Transportation Crew Exposure

The RADTRAN 4 computer code includes models for calculating incident-free risks for shipment of radioactive packages. For shipments of S1C Prototype related radioactive materials, RADTRAN 4 models were used to estimate: (1) exposure to persons within about one-half mile of each side of the transport route (off-link exposures), (2) exposures to persons sharing the transport route (such as passengers on passing trains or vehicles, known as on-link exposures), (3) exposures to persons at stops (such as residents or workers not directly involved with the shipment), and (4) exposures to transportation crew members. The exposures calculated for the first three groups were added together to obtain the general population exposure estimates. The exposure calculated for the transportation crew was designated as the occupational exposure. The impacts of exposure to the S1C Prototype package handlers are included in the facility activities analyses in Appendix B, Section B.2.

Highway shipments of packages similar in size to the pressure vessel package have occurred between the Windsor Site and the Griffin Line railhead in the past. Based on past experience for these shipments, local police escorts have allowed traffic to pass to reduce congestion. Related analyses assumed that limited traffic would pass the slow moving heavy hauler portion of the pressure vessel shipment.

The transportation crew would receive radiation exposure directly from radioactive packages during transit and/or inspection periods. For truck and heavy hauler shipments, RADTRAN 4 assumes crew exposure is only received during the transit period and no inspections occur. For rail shipments, RADTRAN 4 assumes crew exposure is only received during periods of package inspections. Crew exposure is assumed to be negligible during transit due to the relatively long separation distance between the crew and the package and massive shielding of intervening structures. Therefore, for rail shipments, RADTRAN 4 assigns crew exposure to one individual, the inspector.

C.4.2 Maximally Exposed Individuals

To estimate the maximum radiological exposure to an individual member of the transportation crew and an individual in the general public during incident-free radioactive shipments, various hypothetical scenarios were evaluated. Four scenarios were evaluated for individuals in the general population during rail shipments: 1) a rail yard worker working at a distance of ten meters from the radioactive package for two hours, 2) a resident living 30 meters from a rail line used to ship a radioactive package with the package in transit, 3) a resident living 200 meters from a rail line used to ship a radioactive package and the shipment is stopped for 20 hours, and 4) a person standing still for one hour at a distance of six meters from a radioactive package loaded on a railcar. Since the inspector is the only transportation crew member exposed during rail shipments, he is also the maximally exposed individual worker.

Three hypothetical scenarios were evaluated for individuals in the general population during highway shipments: 1) a person who is caught in traffic at a distance of one meter from the radioactive package for one half hour, 2) a resident living 30 meters from a highway used to ship a radioactive package with the package in transit, and 3) a service station worker working at a distance of 20 meters from the package for two hours. The maximally exposed individual worker for highway shipments is the truck driver.

The following formula was used to calculate the radiological exposures to individuals at a fixed distance from a radioactive package during a stop:

$$E = (T \times K \times TI) / D^2$$

where:

E = Exposure (Rem).

T = Total exposure time (hours).

K = Point source conversion factor (meters squared).

TI = Transport Index (equal to the radiation level at one meter from the package surface, Rem per hour).

D = Average distance from centerline of container to exposed person (meters).

Exposure to individuals at a fixed distance from the route along which the shipment is being transported was calculated using the following formula for a moving radiation source traveling with a fixed velocity, V, in meters per hour. All other terms are the same as described for the previous formula.

$$E = (\pi \times K \times TI) / (V \times D)$$

C.5 Computer Model Variables and Assumptions

This section highlights various assumptions and specific variables that were used in transportation related analyses for S1C Prototype related shipments. Table C-5 provides the RADTRAN 4 computer program default values and also identifies different values used in analyses when a default value did not reflect the best estimate of current conditions.

Table C-5: Values for RADTRAN 4 Key Input Parameters

RADTRAN 4 Input Parameter		Default Value		Value Used in Analyses			
				Hanford		Savannah River	
		Truck	Rail	Truck	Rail	Truck	Rail
1)	Fraction of Travel in Rural Zone	0.90	0.90	0.79 a,d	0.78 a	0.51 a,d	0.53 a
2)	Fraction of Travel in Suburban Zone	0.05	0.05	0.18 a,d	0.18 a	0.42 a,d	0.35 a
3)	Fraction of Travel in Urban Zone	0.05	0.05	0.03 a	0.04 a	0.07 a	0.12 a
4)	Velocity in Rural Zone (kilometers per hour)	88.49	64.37	= d	=	= d	=
5)	Velocity in Suburban Zone (kilometers per hour)	40.25	40.25	= d	=	= d	=
6)	Velocity in Urban Zone (kilometers per hour)	24.16	24.16	=	=	=	=
7)	Number of Crew Members Exposed on a Shipment	2	5	c	1.00 b	c	1.00 b
8)	Average Distance from Radiation Source to Crew During Shipment (meters)	3.10	152.40	e	=	e	=
9)	Number of Handlings per Shipment	0.0	2.00	=	=	=	=
10)	Stop Time for Shipment (hours per kilometer)	0.011	0.033	0.005 b	=	0.005 b	=
11)	Minimum Stop Time per Trip (hours)	0.0	10	=	=	=	=
12)	Distance-Independent Stop Time per Trip (hours)	0.0	60	=	=	=	=
13)	Minimum Number of Rail Inspections or Classifications	0.0	2	=	=	=	=
14)	Number of Persons Exposed During Stop	50	100	=	=	=	=
15)	Average Exposure Distance When Stopped (meters)	20	20	=	=	=	=
16)	Storage Time per Shipment (hours)	0.0	4.0	=	0.0 b	=	0.0 b
17)	Number of Persons Exposed During Storage	100	100	0.0 b	0.0 b	0.0 b	0.0 b
18)	Average Exposure Distance During Storage (meters)	100	100	0.0 b	0.0 b	0.0 b	0.0 b
19)	Number of Persons per Vehicle Sharing the Transport Link	2	3	=	=	=	=
20)	Fraction of Urban Travel During Rush Hour	0.08	0.0	=	=	=	=
21)	Fraction of Urban Travel on City Streets	0.05	1.0	=	=	=	=
22)	Fraction of Rural and Suburban Travel on Freeways	0.85	0.0	=	=	=	=
23)	One-way Traffic Count in Rural Zones	470	1	=	=	=	=
24)	One-way Traffic Count in Suburban Zones	780	5	=	=	=	=
25)	One-way Traffic Count in Urban Zone	2800	5	=	=	=	=

= RADTRAN 4 default value was assumed.

a. RADTRAN 4 default value not used. Data obtained from INTERLINE and HIGHWAY computer programs.

b. RADTRAN 4 default value not used. Data based on historical information.

c. RADTRAN 4 default value used for normal truck highway shipment. Crew size of 4 assumed for heavy hauler shipment.

d. Transportation analysis of the pressure vessel package by heavy hauler from the Windsor Site to the Griffin Line Railhead used the following values: parameter 1) fraction of travel in a rural zone = 0.999, parameter 2) fraction of travel in a suburban zone = 0.001, and parameters 4) and 5) velocity = 3.2 kilometers per hour.

e. The following average distances from radiation source to the transportation crew were used for highway shipments: 4.02 meters for miscellaneous packages, 5.95 meters for the pressure vessel package, 5.02 meters for the steam generator packages, and 4.66 meters for the pressurizer package.

C.5.1 Planned Number of Shipments and Package Sizes

Table C-6 defines the assumed size of each radioactive package that would be shipped from the Windsor Site. Analyses assume there would be six miscellaneous waste packages per truck shipment and a total of 19 miscellaneous shipments. The major components would be shipped as whole units in 4 individual shipments. The SPAN4 computer code used all of the package dimensions shown in Table C-6. The RADTRAN 4 computer code used an effective package size based on the length dimension only.

Table C-6: Package Data for the S1C Prototype Reactor Plant Components

Package Type	External Package Dimensions
Miscellaneous components via truck	72 inches wide x 48 inches tall x 48 inches deep
Pressure Vessel via heavy hauler and rail	224 inches long x 127 inches diameter
Steam Generator via truck	151 inches long x 49 inches diameter
Pressurizer via truck	123 inches long x 35 inches diameter

C.5.2 Transport Indexes

Transport index values represent the radiation levels at one meter from the package surface of radiological shipments. The transport index values used in the transportation analyses, listed in Table C-7, are based on records of similar low level radioactive waste shipments. The transport index values for the steam generator, pressurizer, and miscellaneous shipments conservatively assume that the components would be shipped without shielding and in standardized disposal containers. As a result of cobalt-60 radioactive decay during the 30-year caretaking period, the transport index values for deferred dismantlement reflect a 98% reduction when compared to prompt dismantlement.

For the pressure vessel shipment, a large shielded disposal container would be required. It was assumed that the large shielded disposal container would be designed to meet a desired transport index at the time of shipment. As a result, the same transport index value was used in the transportation analyses of the pressure vessel shipments for the prompt and deferred dismantlement alternatives.

Table C-7: Transport Indexes [1, 2]

Package Type	Prompt Dismantlement	Deferred Dismantlement
Pressure Vessel	1.5	1.5
Steam Generator	14.4	0.3
Pressurizer	15.4	0.3
Miscellaneous (6 boxes per shipment)	13.9	0.3

1. The Transport Index is a dimensionless number (rounded to the first decimal place) that represents the radiation level at one meter from the package surface in millirem per hour.

2. All packages would be designed and prepared for shipment to meet Department of Transportation requirements, 49 CFR Part 173.

C.5.3 Transportation Distances and Population Densities

As discussed in Section C.3.1, the HIGHWAY and INTERLINE computer codes were used for determining transportation distances and the population densities along the transportation routes. Based on historical data from similar radioactive material shipments, and for added conservatism, the total distances used for pressure vessel rail shipment analysis were increased by approximately 11% above the distances predicted by the INTERLINE computer program. Similarly, the total distances used for highway shipment analyses were increased by approximately 3% above the distances predicted by HIGHWAY computer program. The increased distance factors were applied equally for each population density area.

C.5.4 Shipment Storage Time

Shipments made under the cognizance of Naval Reactors are made in an efficient and safe manner. Shipments of radioactive material are not stored while in the process of being shipped; therefore, there was no shipment storage time associated with any of the shipments.

C.5.5 Rail Shipment Variables

Train Velocity: The RADTRAN 4 computer code provides different default values for train velocity that correspond to travel in the various population density areas. These analyses used the default values for train velocity. The RADTRAN 4 computer code also provides standard values for train stop times that were used in this study.

Crew Size: The RADTRAN 4 computer code default value for the number of personnel that accompany a special radioactive shipment is five, which includes three crew members plus two courier escorts. Although the reactor pressure vessel is radioactive, it does not contain spent fuel and would not be considered to be a special shipment; therefore, couriers would not be required. For this analysis, the train crew size was assumed to be three. RADTRAN 4 assumes crew exposure is received during routine package inspections while the train is stopped. During transit, crew exposure is assumed to be negligible due to the relatively long separation distance between the crew and the package and the shielding effects of intervening structures. Therefore, crew exposure is assigned to only one individual, the inspector.

Shielding Factor: For train stops, the standard RADTRAN 4 computer code gamma shield factor default value is 0.1. This value assumes the presence of substantial rail yard structures equivalent to approximately four inches of steel. Four inches of steel reduces gamma radiation exposure by more than a factor of 10. Therefore, a shield factor of 0.1 was considered to be reasonable.

Distance to the Package: The RADTRAN 4 default value of 152.4 meters was used for the distance between the pressure vessel package and the transportation crew during transit.

C.5.6 Truck and Heavy Hauler Shipment Variables

Truck Velocity: For truck shipment of smaller packages, the RADTRAN 4 defaults were used in all three population density zones. For the heavy hauler segment of the pressure vessel shipment, the velocity was assumed to be 3.2 kilometers per hour.

Crew Size: The RADTRAN 4 computer code default values for the truck crew were used for the truck shipments for the smaller packages. For the reactor pressure vessel, the number of persons assumed to be in the heavy hauler transportation crew was four.

Number of Truck Inspections: Radioactive package shipments would be inspected prior to leaving the Windsor Site. Analyses assumed that there would be no inspections during transport.

Truck Stop Time: A calculated stop time of 0.005 hours per kilometer was used for all highway and heavy hauler shipments. This is based on historical data from other low level radioactive waste shipments that originated at the Windsor Site.

Distance to the Package: The crew was assumed to be located 3.1 meters from the outside of the package for the truck and the heavy hauler shipments.

C.6 Technical Approach for Assessing Radioactive Shipment Accidents

Health risks from hypothetical accidents involving radioactive shipments were evaluated for the general population only. Risk is the product of an event probability and consequence. Analyses assumed that the transportation workers would evacuate the scene of an accident within a relatively short time after the accident occurred. Therefore, the risks of transportation accidents on transportation workers are included in the results for the general population.

C.6.1 General Population and Risk

The RADTRAN 4 computer code was used to calculate the radiological risk to the general population under accident conditions. The RADTRAN 4 computer code evaluates six pathways for radiation exposures resulting from an accident. The six pathways are:

- Direct radiation exposure from the damaged package.
- Inhalation exposure from the plume of radioactive material released from the damaged package.
- Direct radiation exposure from immersion in the plume of radioactive material released from the damaged package.
- Direct radiation exposure from ground deposition of the radioactive material released from the damaged package.
- Inhalation exposure from resuspension of the radioactive material deposited on the ground.
- Ingestion exposure from food products grown on the soil contaminated by ground deposition of radioactive material released from the damaged package.

A specific formula is used to estimate the radiological exposure from each pathway. The formula accounts for the probability of an accident occurring and the severity. The internal pathways (inhalation and ingestion) exposures are based on exposure to the body over a 50-year period. The total radiation exposure resulting from the hypothetical accident equals the sum of the exposures from each pathway. The general equation for the radiation exposure to the general population from all pathways is:

$$D_R = \sum_{c,r} L_C \, P_r \times \sum_{i,j,k} (P_j \times RF_j \times D_{i,j,k})$$

where: D_R = Total radiation exposure risk to the general population from the accident.

L_C = Shipment distance.

P_r = Probability of traffic accidents per unit distance (Accident Probabilities, Table C-8).

P_j = Probability that an accident of a specific severity category occurs. Section C.6.3 provides further discussion.

RF_j = Fraction of curies released from shipping container after a severe accident (Corrosion Product Release Fractions, Table C-11).

$D_{i,j,k}$ = Radiation exposure commitment resulting from a accident of a specific severity category (j), received through a specific pathway (I) in a specific population density zone (k).

Because it is impossible to predict the specific location of a transportation accident, neutral weather conditions were assumed (Pasquill Stability Class D as defined in Reference C-13). Since neutral meteorological conditions are the most frequently occurring atmospheric conditions in the United States, these conditions are most likely to be present in the event of a transportation accident.

C.6.2 Accident Probability

The average probability of an accident in the United States by transportation mode was obtained from Reference C-11. The probabilities used in transportation accident analyses are presented in Table C-8. These probabilities represent all categories of accidents. Rail accident rates are the same for rural, suburban, and urban areas.

Table C-8: Accident Probabilities

Transport Mode	Accidents per Kilometer in Rural Zones (National Average)	Accidents per Kilometer in Urban and Suburban Zones (National Average)
Truck	2.03×10^{-7}	3.58×10^{-7}
Rail	5.57×10^{-8}	5.57×10^{-8}

C.6.3 Severe Accident Probability

The severe accident probability for S1C-related shipments was based on the Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement (Reference C-1). That study conservatively estimated that 99.4% of truck and rail accidents involving Type B packages would not result in any release of package contents to the environment. The study estimated that 0.6% of truck and rail accidents involving Type B packages would be severe enough to cause a breach in the container and would result in a release of loose corrosion products to the environment. A severe accident probability of 0.6% was assumed in analyses of all S1C-related truck and rail shipments. Further discussion of package types is provided in Section C.6.5.

C.6.4 Corrosion Product Activity

Analyses assumed that the amount of corrosion product activity in each package type was equally distributed. The amount of activated corrosion products was derived based on formulas that correlate reactor plant pipewall dose rate measurements with calculated wetted surface areas and corrosion product deposition levels. The radioactivity amounts used in the transportation accident analyses were based on actual end-of-life data decay corrected to the time of dismantlement. Values for prompt dismantlement are decay corrected for four years and values for deferred dismantlement were decay corrected for 34 years. As discussed in Section C.5.2 for transport indexes, nearly all of the gamma emitting radioactive material in the reactor plant comes from cobalt-60. Cobalt-60 contributes more than 99% to the total exposure levels in the accident analyses for the prompt dismantlement alternative and approximately 85% for the deferred dismantlement alternative. Table C-9 provides the total amount of corrosion product radioactivity assumed in transportation analyses for the package types. Table C-10 lists the radionuclides that result in at least 99% of the possible exposure for the deferred dismantlement alternative.

Table C-9: Corrosion Product Radioactivity Content of Package Types, Decay Corrected

Package Type	Prompt Dismantlement [1] (Curies)	Deferred Dismantlement [2] (Curies)
Miscellaneous	0.59	0.10
Steam Generator	4.85	0.79
Reactor Vessel	2.91	0.47
Pressurizer	0.83	0.13

1. Values decay corrected for four years after final S1C Prototype reactor plant shutdown.
2. Values decay corrected for 34 years after final S1C Prototype reactor plant shutdown.

Table C-10: Radionuclides Included in S1C Prototype Corrosion Product Calculations

Co-60	Ni-63
Sr-90	Pu-239
Nb-94	Pu-241
Cs-137	Am-241
Pu-238	Cm-244

C.6.5 Package Categorization

All reactor plant components would be shipped as packages meeting Department of Transportation regulations 49 CFR Part 173 (Shippers - General Requirements for Shipments and Packagings). The regulations include requirements for several types of packaging. Transportation risk analyses assumed that the reactor pressure vessel would be shipped in a single package meeting Type B criteria. Type B packaging is designed and tested to rigorous standards to prevent any release of contents under most accident conditions. Type B packaging design and testing standards are defined in Nuclear Regulatory Commission regulations 10 CFR Part 71 (Packaging and Transportation of Radioactive Material). The steam generator, pressurizer, and miscellaneous components would be shipped as packages meeting the Department of Transportation criteria for either Low Specific Activity materials or Surface Contaminated Objects. The design and performance standards for these packages are not required to be as rigorous as Type B packaging criteria.

C.6.5.1 Package Release Fractions

The package release fraction represents the percentage of radioactive material in the shipment that would be released to the environment following a severe accident. The amount of radioactivity that could be released from each package was derived based on historical activated corrosion product models. The corrosion product model accounts for all activated corrosion products which adhered to all wetted surfaces inside the reactor vessel and coolant system over plant life. Most of the radioactive corrosion products contained in reactor plant materials are strongly adhering to the inside surfaces. Based on conservative results of laboratory testing, transportation accident analyses for each package type assumed that 33% of corrosion product radioactivity would be loosened from the impact of a hypothetical accident.

As discussed in Section C.6.3 for severe accident probability, only severe accidents would result in a release of radioactivity to the environment. Although the same severe accident probability was assumed in the accident analyses for all packages, the Type B reactor pressure vessel package would be much less susceptible to damage or breaching. Consistent with the Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement (Reference C-1), transportation risk analysis of the Type B reactor pressure vessel package assumed that 10% of its loose corrosion products would be released following a severe accident. Since 33% is a conservative prediction of the available corrosion products which could be loosened, as discussed above, the severe accident analysis of the reactor pressure vessel package applied a package release fraction of 0.033 (33% of the total available corrosion product radioactivity in the package × 10% release = 3.3% = 0.033).

Since the steam generator, pressurizer, and miscellaneous materials would be shipped in packages other than Type B, transportation risk analyses conservatively assumed that all (100%) of the loose corrosion products could be released following a severe accident. Severe accident analyses of these package types applied a package release fraction of 0.33 (33% of the total available corrosion product radioactivity × 100% release = 33% = 0.33). Table C-11 summarizes the release fractions used in transportation risk analyses.

Table C-11: Package Release Fractions for Severe Accident Conditions

Package Type	Release Fraction
Pressure Vessel	0.033
Steam Generator	0.33
Pressurizer	0.33
Miscellaneous	0.33

C.6.6 Maximum Consequence to Individual and Population

Maximum consequences were evaluated for the steam generator, pressurizer, and miscellaneous packages assuming that a hypothetical accident occurs. For the reactor pressure vessel shipment, maximum consequences were evaluated for very severe accidents having a low probability of occurrence. For all package types, radiological exposures were calculated for the maximally exposed individual and the general population. Because it is impossible to predict the specific location of a transportation accident, exposures to the general population were calculated for each of the three population density regions (rural, suburban and urban) over a 50-mile radius. The RISKIND computer code was used to calculate the maximum consequence exposures.

The exposure pathways evaluated by RISKIND are identical to those used in the RADTRAN 4 computer code for exposures to the general population as discussed in Section C.6.1. However, the analyses for the maximum consequence exposure to an individual considered acute doses only. Because the food ingestion pathway does not result in an acute dose, this pathway was not included in the maximum consequence analyses for individuals. Analyses assumed that the maximally exposed individual would be exposed unshielded during the passage of the radioactive plume released from the accident under worst (stable) atmosphere conditions.

Remedial actions following an accident would significantly reduce the consequences of the accident; however, analyses conservatively assume no cleanup actions.

C.6.6.1 Probability Cutoff Criterion

Consistent with Reference C-1, maximum consequence analyses applied a cutoff criterion of one in ten-million (1.0×10^{-7}) chance of occurrence per year for excluding improbable accidents from detailed evaluation. Severe accident probability calculations considered variables such as the probability of an accident occurring (Section C.6.2), the severe accident probability (Section C.6.3), the fraction of travel in each population area, the number of shipments, and the probability of meteorological conditions that would lead to the higher consequences.

C.6.7 Plume Release Height Following an Accident

For the accident risk assessment, a ground level release was used in the RADTRAN 4 model. For the maximum consequence assessment, a plume release height of ten meters was used in the RISKIND model.

C.6.8 Direct Exposure from a Damaged Package

The radiation level following an accident was assumed to be at the Nuclear Regulatory Commission limit in 10 CFR Part 71 of one rem per hour at one meter from the component surface. Analyses concluded that the total direct exposure to the general population or maximally exposed individual from the damaged package is negligible.

C.6.9 Food Transfer Factors

These transportation analyses used the same food transfer factors as similar analyses in the Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement (Reference C-1).

C.6.10 Distance from the Accident Scene to the Maximally Exposed Individual

Analyses assumed that the maximally exposed individual would remain in one location, unshielded, during the time that a radioactive plume passed by following a hypothetical accident. The location of maximum exposure was also assumed to be within the range of 100 meters to 400 meters from the accident site. This location was determined using RISKIND based on the assumed atmospheric stability and plume release height.

C.6.11 Population Density in the Vicinity of a Hypothetical Accident

The standard national average for each population density from the RADTRAN 4 computer code was used for the RISKIND maximum consequence assessment. The assessment considers the population within 80 kilometers (50 miles) of the hypothetical accident site under both neutral and stable weather conditions. The population ranged from 2.6 thousand in rural areas to 1.5 million in urban areas.

C.7 Summary of Analysis Results

This section provides the results of all transportation-related analyses performed for radioactive packages that would be shipped as a result of S1C Prototype reactor plant dismantlement.

C.7.1 Incident-Free Risk

Incident-free transportation analysis results are provided in the following tables:

Destination	Prompt Dismantlement	Deferred Dismantlement
Savannah River Site	Table C-12	Table C-14
Hanford Site	Table C-13	Table C-15

Radiological exposure and latent fatal cancers are provided for the general population, for the transportation crew and for the maximally exposed individual. The predicted numbers of fatalities from nonradiological sources (pollutants) are provided for comparison purposes. The results show the nonradiological risks are comparable to the radiological risks.

C.7.2 Accident Risk

Transportation accident analysis results are provided in the following tables:

Destination	Prompt Dismantlement	Deferred Dismantlement
Savannah River Site	Table C-16	Table C-18
Hanford Site	Table C-17	Table C-19

Tables C-16 through C-19 represent the risks of accidents that would involve a release of radioactivity to the environment and the probabilities of occurrence. Radiological exposure and latent fatal cancer risks are provided for the general population. The predicted numbers of fatalities from nonradiological sources (traffic accidents) are included for comparison purposes. The major contributor is the ground contamination pathway (more than 90% of the total exposure). The ingestion pathway is the next important pathway. The analyses indicate that the nonradiological risks from accidents exceed the radiological risks for both the prompt and deferred alternatives.

C.7.3 Accident Maximum Consequences

Analysis results that estimate the maximum consequences from a severe accident are provided in Table C-20 for the prompt dismantlement alternative and Table C-21 for the deferred dismantlement alternative. These results apply to shipments to either the Savannah River Site or to the Hanford Site. The accident with the highest maximum consequences involves the steam generator shipment because the steam generators have the highest corrosion product radioactivity content.

Table C-12: Incident-Free Transportation Risks, Windsor Site to Savannah River Disposal Site, Prompt Dismantlement Alternative

	General Population		Transportation Crew		Maximally Exposed Individual in the General Population [1]		Maximally Exposed Worker [1]		
	Exposure (Person-Rem)	Latent Fatal Cancer Risk	Exposure (Person-Rem)	Latent Fatal Cancer Risk	Exposure (Rem)	Latent Fatal Cancer Risk	Exposure (Rem)	Latent Fatal Cancer Risk	Non-Radiological Fatality Risk
Misc.	1.42×10^{0}	7.10×10^{-4}	5.85×10^{0}	2.34×10^{-3}	1.31×10^{-1}	6.55×10^{-5}	2.92×10^{0}	1.17×10^{-3}	3.71×10^{-4}
Pressure Vessel	4.52×10^{-3}	2.26×10^{-6}	7.03×10^{-3}	2.81×10^{-6}	6.16×10^{-4}	3.08×10^{-7}	5.43×10^{-3}	2.17×10^{-6}	5.67×10^{-5}
Steam Generator	3.60×10^{-1}	1.80×10^{-4}	5.47×10^{-1}	2.19×10^{-4}	1.44×10^{-2}	7.20×10^{-6}	2.73×10^{-1}	1.09×10^{-4}	3.90×10^{-5}
Pressurizer	1.48×10^{-1}	7.42×10^{-5}	2.61×10^{-1}	1.05×10^{-4}	7.73×10^{-3}	3.87×10^{-6}	1.31×10^{-1}	5.23×10^{-5}	1.95×10^{-5}
TOTAL	1.93×10^{0}	9.66×10^{-4}	6.66×10^{0}	2.67×10^{-3}	1.54×10^{-1}	7.69×10^{-5}	3.33×10^{0}	1.33×10^{-3}	4.86×10^{-4}

1. Data for the maximally exposed individual are conservatively assumed to apply to the same person for all shipments.

Table C-13: Incident-Free Transportation Risks, Windsor Site to Hanford Disposal Site, Prompt Dismantlement Alternative

	General Population		Transportation Crew		Maximally Exposed Individual in the General Population [1]		Maximally Exposed Worker [1]		
	Exposure (Person-Rem)	Latent Fatal Cancer Risk	Exposure (Person-Rem)	Latent Fatal Cancer Risk	Exposure (Rem)	Latent Fatal Cancer Risk	Exposure (Rem)	Latent Fatal Cancer Risk	Non-Radiological Fatality Risk
Misc.	3.75×10^{0}	1.88×10^{-3}	8.27×10^{0}	3.31×10^{-3}	1.31×10^{-1}	6.55×10^{-5}	4.13×10^{0}	1.65×10^{-3}	4.35×10^{-4}
Pressure Vessel	7.28×10^{-3}	3.64×10^{-6}	1.22×10^{-2}	4.90×10^{-6}	6.16×10^{-4}	3.08×10^{-7}	1.06×10^{-2}	4.25×10^{-6}	4.90×10^{-5}
Steam Generator	9.53×10^{-1}	4.76×10^{-4}	1.35×10^{0}	5.39×10^{-4}	1.44×10^{-2}	7.20×10^{-6}	6.74×10^{-1}	2.70×10^{-4}	4.58×10^{-5}
Pressurizer	3.94×10^{-1}	1.97×10^{-4}	6.45×10^{-1}	2.58×10^{-4}	7.73×10^{-3}	3.87×10^{-6}	3.23×10^{-1}	1.29×10^{-4}	2.29×10^{-5}
TOTAL	5.11×10^{0}	2.55×10^{-3}	1.03×10^{1}	4.11×10^{-3}	1.54×10^{-1}	7.69×10^{-5}	5.14×10^{0}	2.06×10^{-3}	5.53×10^{-4}

1. Data for the maximally exposed individual are conservatively assumed to apply to the same person for all shipments.

Table C-14: Incident-Free Transportation Risks, Windsor Site to Savannah River Disposal Site, Deferred Dismantlement Alternative

	General Population		Transportation Crew		Maximally Exposed Individual in the General Population [1]		Maximally Exposed Worker [1]		
	Exposure (Person-Rem)	Latent Fatal Cancer Risk	Exposure (Person-Rem)	Latent Fatal Cancer Risk	Exposure (Rem)	Latent Fatal Cancer Risk	Exposure (Rem)	Latent Fatal Cancer Risk	Non-Radiological Fatality Risk
Misc.	2.84×10^{-2}	1.42×10^{-5}	1.17×10^{-1}	4.68×10^{-5}	2.62×10^{-3}	1.31×10^{-6}	5.85×10^{-2}	2.34×10^{-5}	3.71×10^{-4}
Pressure Vessel	4.52×10^{-3}	2.26×10^{-6}	7.03×10^{-3}	2.81×10^{-6}	6.16×10^{-4}	3.08×10^{-7}	5.43×10^{-3}	2.17×10^{-6}	5.67×10^{-5}
Steam Generator	7.20×10^{-3}	3.60×10^{-6}	1.09×10^{-2}	4.37×10^{-6}	2.88×10^{-4}	1.44×10^{-7}	5.47×10^{-3}	2.19×10^{-6}	3.90×10^{-5}
Pressurizer	2.97×10^{-3}	1.48×10^{-6}	5.23×10^{-3}	2.09×10^{-6}	1.55×10^{-4}	7.75×10^{-6}	2.61×10^{-3}	1.05×10^{-6}	1.95×10^{-5}
TOTAL	4.31×10^{-2}	2.15×10^{-5}	1.40×10^{-1}	5.61×10^{-5}	3.08×10^{-3}	1.54×10^{-6}	7.20×10^{-2}	2.88×10^{-5}	4.86×10^{-4}

1. Data for the maximally exposed individual are conservatively assumed to apply to the same person for all shipments.

Table C-15: Incident-Free Transportation Risks, Windsor Site to Hanford Disposal Site, Deferred Dismantlement Alternative

	General Population		Transportation Crew		Maximally Exposed Individual in the General Population [1]		Maximally Exposed Worker [1]		
	Exposure (Person-Rem)	Latent Fatal Cancer Risk	Exposure (Person-Rem)	Latent Fatal Cancer Risk	Exposure (Rem)	Latent Fatal Cancer Risk	Exposure (Rem)	Latent Fatal Cancer Risk	Non-Radiological Fatality Risk
Misc.	7.51×10^{-2}	3.75×10^{-5}	1.65×10^{-1}	6.62×10^{-5}	2.62×10^{-3}	1.31×10^{-6}	8.27×10^{-2}	3.31×10^{-5}	4.35×10^{-4}
Pressure Vessel	7.28×10^{-3}	3.64×10^{-6}	1.22×10^{-2}	4.90×10^{-6}	6.16×10^{-4}	3.08×10^{-7}	1.06×10^{-2}	4.25×10^{-6}	4.90×10^{-5}
Steam Generator	1.91×10^{-2}	9.53×10^{-6}	2.70×10^{-2}	1.08×10^{-5}	2.88×10^{-4}	1.44×10^{-7}	1.35×10^{-2}	5.39×10^{-6}	4.58×10^{-5}
Pressurizer	7.87×10^{-3}	3.94×10^{-6}	1.29×10^{-2}	5.16×10^{-6}	1.55×10^{-4}	7.75×10^{-8}	6.45×10^{-3}	2.58×10^{-6}	2.29×10^{-5}
TOTAL	1.09×10^{-1}	5.46×10^{-5}	2.18×10^{-1}	8.70×10^{-5}	3.08×10^{-3}	1.54×10^{-6}	1.13×10^{-1}	4.53×10^{-5}	5.53×10^{-4}

1. Data for the maximally exposed individual are conservatively assumed to apply to the same person for all shipments.

Table C-16: Transportation Accident Risks, Windsor Site to Savannah River Disposal Site, Prompt Dismantlement Alternative

	General Population		Non-Radiological Fatality Risk
	Exposure (Person-Rem)	Latent Fatal Cancer Risk	
Miscellaneous	3.97×10^{-4}	1.99×10^{-7}	3.13×10^{-3}
Pressure Vessel	2.56×10^{-6}	1.28×10^{-9}	1.03×10^{-4}
Steam Generator	3.45×10^{-4}	1.73×10^{-7}	3.29×10^{-4}
Pressurizer	2.96×10^{-5}	1.48×10^{-8}	1.65×10^{-4}
TOTAL	7.74×10^{-4}	3.88×10^{-7}	3.73×10^{-3}

Table C-17: Transportation Accident Risks, Windsor Site to Hanford Disposal Site, Prompt Dismantlement Alternative

	General Population		Non-Radiological Fatality Risk
	Exposure (Person-Rem)	Latent Fatal Cancer Risk	
Miscellaneous	4.66×10^{-4}	2.33×10^{-7}	1.01×10^{-2}
Pressure Vessel	2.82×10^{-6}	1.41×10^{-9}	2.80×10^{-4}
Steam Generator	4.05×10^{-4}	2.03×10^{-7}	1.07×10^{-3}
Pressurizer	3.47×10^{-5}	1.74×10^{-8}	5.33×10^{-4}
TOTAL	9.09×10^{-4}	4.55×10^{-7}	1.20×10^{-2}

Table C-18: Transportation Accident Risks,
Windsor Site to Savannah River Disposal Site,
Deferred Dismantlement Alternative

	General Population		Non-Radiological Fatality Risk
	Exposure (Person-Rem)	Latent Fatal Cancer Risk	
Miscellaneous	8.97×10^{-6}	4.49×10^{-9}	3.13×10^{-3}
Pressure Vessel	5.79×10^{-8}	2.90×10^{-11}	1.03×10^{-4}
Steam Generator	7.80×10^{-6}	3.90×10^{-9}	3.29×10^{-4}
Pressurizer	6.67×10^{-7}	3.34×10^{-10}	1.65×10^{-4}
TOTAL	1.75×10^{-5}	8.75×10^{-9}	3.73×10^{-3}

Table C-19: Transportation Accident Risks,
Windsor Site to Hanford Disposal Site,
Deferred Dismantlement Alternative

	General Population		Non-Radiological Fatality Risk
	Exposure (Person-Rem)	Latent Fatal Cancer Risk	
Miscellaneous	1.07×10^{-5}	5.35×10^{-9}	1.01×10^{-2}
Pressure Vessel	6.49×10^{-8}	3.25×10^{-11}	2.80×10^{-4}
Steam Generator	9.28×10^{-6}	4.64×10^{-9}	1.07×10^{-3}
Pressurizer	7.93×10^{-7}	3.97×10^{-10}	5.33×10^{-4}
TOTAL	2.08×10^{-5}	1.04×10^{-8}	1.20×10^{-2}

Table C-20: Hypothetical Severe Accident Analysis Results (Maximum Consequences), Prompt Dismantlement Alternative

Maximally Exposed Individual		Rural		Suburban		Urban	
Exposure (rem)	Latent Fatal Cancer Risk	Collective Exposure (person-rem)	Latent Fatal Cancer Risk	Collective Exposure (person-rem)	Latent Fatal Cancer Risk	Collective Exposure (person-rem)	Latent Fatal Cancer Risk
8.63×10^{-2}	4.32×10^{-5}	1.41×10^{1}	7.07×10^{-3}	1.62×10^{2}	8.10×10^{-2}	2.63×10^{2}	1.32×10^{-1}

Table C-21: Hypothetical Severe Accident Analysis Results (Maximum Consequences), Deferred Dismantlement Alternative

Maximally Exposed Individual		Rural		Suburban		Urban	
Exposure (rem)	Latent Fatal Cancer Risk	Collective Exposure (person-rem)	Latent Fatal Cancer Risk	Collective Exposure (person-rem)	Latent Fatal Cancer Risk	Collective Exposure (person-rem)	Latent Fatal Cancer Risk
4.60×10^{-3}	2.30×10^{-6}	4.28×10^{-1}	2.14×10^{-4}	3.57×10^{0}	1.79×10^{-3}	6.07×10^{0}	3.04×10^{-3}

REFERENCES

C-1 Department of Energy, *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement*, DOE/EIS-0203-F, April 1995.

C-2 Department of Navy, *Draft Environmental Impact Statement on the Disposal of Decommissioned, Defueled Cruiser, Ohio Class, and Los Angeles Class Naval Reactor Plants*, August 1995.

C-3 Neuhauser, K. S. and F. L. Kanipe, *RADTRAN 4 Users Guide*, SAND89-2370, TTC-0943, UC-722, Sandia National Laboratories, Albuquerque, New Mexico, January 1992.

C-4 Neuhauser, K. S. and F. L. Kanipe, *RADTRAN 4 Volume II: Technical Manual*, SAND89-2370, Sandia National Laboratories, Albuquerque, New Mexico, August 1993.

C-5 Yuan, Y. C., S. Y. Chen, D. J: LePoire, R. Rothman, *RISKIND - A Computer Program for Calculating Radiological Consequences and Health Risks from Transportation of Spent Nuclear Fuel*, Environmental Assessment and Information Sciences Division, Argonne National Laboratory, Argonne, Illinois, February 1993.

C-6 Johnson, P. E., D. S. Joy, D. B. Clark, J. M. Jacobi, *INTERLINE 5.0, An Expanded Railroad Routing Model*: Program Description, Methodology, and Revised User's Manual, ORNL/TM-12090, Oak Ridge National Laboratory, Oak Ridge, Tennessee, March 1993.

C-7 Johnson, P. E., D. S. Joy, D. B. Clark, J. M. Jacobi, *HIGHWAY 3.01, An Enhanced Transportation Routing Model*: Program Description, Methodology, and Revised User's Manual, ORNL/TM-12124, Oak Ridge National Laboratory, Oak Ridge, Tennessee, March 1993.

C-8 Wallace, O. J., *SPAN4 A Point Kernel Computer Program for Shielding*, WAPD-TM-809(L), Volumes I and II, October 1972.

C-9 International Commission on Radiological Protection (ICRP), *1990 Recommendations of the International Commission on Radiological Protection*, ICRP Publication 60, Annals of the ICRP, Vol. 21, No. 1-3, New York, Pergamon Press, 1991.

C-10 Rao, R. K., E. L. Wilmot, R. E. Luna, *Nonradiological Impacts of Transporting Radioactive Material*, SAND81-1703, TTC-0236, Sandia National Laboratories, Albuquerque, New Mexico, February 1982.

C-11 Saricks, C. and T. Kvitek, *The Longitudinal Review of State-Level Accident Statistics for Carriers of Interstate Freight*, ANL/ESD/TM-68, Argonne National Laboratory Center for Transportation Research, Argonne, Illinois, 1994.

C-12 Cashwell, J. W., K. S. Neuhauser, P. C. Reardon, G. W. McNair, *Transportation Impacts of the Commercial Radioactive Waste Management Program*, SAND85-2715, TTC-0633, Sandia National Laboratories, Albuquerque, New Mexico, April 1986.

C-13 Pasquill, F., *Atmospheric Diffusion*, 2nd ed., John Wiley & Sons, New York, 1974.

C-14 National Academy of Sciences - National Research Council, *Health Effects of Exposure to Low Levels of Ionizing Radiation, BEIR V*, Report of the Committee on the Biological Effects of Ionizing Radiations, 1990.

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