TECHNOLOGY, SAFETY AND COSTS OF DECOMMISSIONING A REFERENCE PRESSURIZED WATER REACTOR POWER STATION

Appendices

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APPENDIX A
REFERENCE PWR FACILITY DESCRIPTION

This appendix characterizes the 3500 MW(t) [1175 MW(e)] PWR power station used as the basis for this decommissioning study. Descriptions of the structures and the equipment housed therein are included. The facility description is typical of the plants expected to be built in the near future and as such represents the state-of-the-art design. Although design details may differ depending upon a particular facility, most of these differences will not have a significant impact on decommissioning. The information contained herein is based on the TROJAN FSAR,⁽¹⁾ Westinghouse RESAR,⁽²⁾ SNUPPS PSAR,⁽³⁾ and personal contacts with Portland General Electric Company (PGE) personnel.

A.1 GENERAL STATION DESCRIPTION

The site consists of approximately 4.7 km² (1160 acres) located near a large body of water and is described in detail in Appendix B. The typical plant layout is shown in Figure A.1-1. Major structures on the site include the Containment Building, Turbine Building, Auxiliary Building, Fuel Building, Control Building and Cooling Tower.

A.1.1 Design Criteria

The basis for development and selection of the design criteria used in this plant are those which, first, provide protection of public safety, second, provide for reliable and economic plant performance, and third, provide an attractive appearance.

The essential systems and components of the plant are designed to enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena. The designs are based upon the most severe of the natural phenomena recorded for the vicinity of the site, with margin to account for uncertainties in the historical data.

⁽¹⁾ TROJAN Final Safety Analysis Report, Portland General Electric Co., Portland, OR, September 1973.

⁽²⁾ RESAR-3 Preliminary Safety Analysis Report, Westinghouse Electric Co., Pittsburgh, PA, November 1973.

⁽³⁾ Standardized Nuclear Unit Power Plant System - SNUPPS, Preliminary Safety Analysis Report, November 1974.

FIGURE A.1-1. Reference PWR Plant Layout

A.1.2 Operating Characteristics

The design and license power level for the reactor core is approximately 3500 MW(t), with a valves-wide-open turbine generator rating of approximately 1175 MW(e) net. The principle of operation is shown in Figure A.1-2.⁽⁴⁾

A.1.3 Nuclear Steam Supply System

The Nuclear Steam Supply System consists of a pressurized water reactor, Reactor Coolant System, and associated auxiliary fluid systems. The Reactor Coolant System is arranged as four closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and steam generator. An electrically heated pressurizer is connected to the hot leg of one reactor coolant loop.

The reactor core is composed of uranium dioxide pellets enclosed in Zircaloy tubes with welded end plugs. The tubes are supported in assemblies by a spring clip grid structure. The mechanical control rods consist of clusters of stainless steel clad silver-indium-cadmium absorber rods and Zircaloy guide tubes located within the fuel assembly. The core fuel is loaded in three regions with new fuel being introduced into the outer region, moved inward into a checkerboard pattern at successive refuelings and discharged from the inner region to spent fuel storage.

The reactor vessel and reactor internals contain and support the fuel and control rods. The reactor vessel is cylindrical with hemispherical heads and is clad internally with stainless steel.

The pressurizer is a vertical cylindrical pressure vessel with hemispherical heads and is equipped with electrical heaters and spray nozzles for system pressure control.

The steam generators are vertical U-tube type heat exchangers utilizing Inconel tubes. Integral separating equipment reduces the moisture content of the steam at the turbine throttle to 1/4 percent or less.

The reactor coolant pumps are vertical, single-stage, centrifugal pumps equipped with controlled leakage shaft seals.

⁽⁴⁾ TROJAN Nuclear Plant Information Center Brochure, Portland General Electric Company, Rainier, Oregon.

PRESSURIZED WATER REACTOR (PWR)

PRESSURIZED WATER REACTOR (PWR). AS WITH THE BOILER IN A COAL-, OIL-OR GAS-BURNING POWER PLANT, A NUCLEAR POWER REACTOR PRODUCES STEAM TO DRIVE A TURBINE WHICH TURNS AN ELECTRIC GENERATOR. INSTEAD OF BURNING FOSSIL FUEL, A REACTOR FISSIONS NUCLEAR FUEL TO PRODUCE HEAT TO MAKE THE STEAM. THE PWR SHOWN HERE IS A TYPE OF REACTOR FUELED BY SLIGHTLY ENRICHED URANIUM IN THE FORM OF URANIUM OXIDE PELLETS HELD IN ZIRCONIUM ALLOY TUBES IN THE CORE. WATER IS PUMPED THROUGH THE CORE TO TRANSFER HEAT TO THE STEAM GENERATOR. THIS COOLANT WATER IS KEPT UNDER PRESSURE IN THE CORE TO PREVENT BOILING AND TRANSFERS HEAT TO THE WATER IN THE STEAM GENERATOR TO MAKE THE STEAM. THREE U. S. COMPANIES MANUFACTURE PWRs: WESTINGHOUSE ELECTRIC CORP., COMBUSTION ENGINEERING, INC., AND THE BABCOCK & WILCOX CO.

FIGURE A.1-2. Functional Diagram of a Pressurized Water Reactor (PWR)

Auxiliary systems are provided to charge the Reactor Coolant System and to add makeup water, purify reactor coolant water, provide chemicals for corrosion inhibition and reactor control, cool system components, remove residual heat when the reactor is shutdown, cool the spent fuel storage pool, sample reactor coolant water, provide for emergency safety injection, and vent and drain the Reactor Coolant System.

A.1.4 Electrical Systems

Two diesel generators are provided as standby sources of emergency power at 4.16 kV each in the event of a loss of normal and preferred AC power. Each generator has sufficient capacity to operate all equipment necessary to prevent undue risk to public safety should a Loss of Coolant Accident (LOCA) occur. In addition, storage batteries are provided as onsite sources of power in the event of a loss of normal DC power.

A.1.5 Power Conversion

The turbine is a tandem-compound, four-element, 1800-rpm unit, having one high pressure and three identical low pressure elements. Combination moisture separator-reheaters are employed to dry and superheat the steam between the high and low pressure turbines. The auxiliaries include deaerating surface condensers, steam jet air ejectors, turbine driven main feed pumps, motor driven condensate pumps, full flow condensate demineralizer, and seven stages of feedwater heating. The steam and turbine systems are designed to receive the heat energy produced in the reactor during normal operation, as well as during a 50 percent load reduction on the turbine-generator. Heat dissipation under the latter condition is accomplished by steam dump to the condenser (40 percent full load). The steam dump permits the plant to accept loss of 50 percent of external load without causing a reactor or turbine shutdown.

A.1.6 Fuel Handling System

The reactor is refueled with equipment designed to handle spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Underwater transfer of spent fuel provides an optically transparent radiation shield, as well as a reliable

source of coolant for removal of decay heat. This system also provides capability for receiving, handling and storage of new fuel.

A.1.7 Radioactive Waste Treatment Systems

The Radioactive Waste Treatment Systems provide all equipment necessary to collect, process, monitor, and discharge radioactive liquid, gaseous and solid wastes that are produced during reactor operation.

Liquid wastes potentially containing radioactive material are collected and monitored. Prior to discharge, equipment is provided for filtering, evaporating and demineralizing the liquid as required. The treated water from the filters or demineralizers or the evaporator distillate may be recycled for use in the plant or may be discharged to the nearby river. The evaporator concentrates, spent demineralizer resins and spent filters are solidified, drummed, and shipped from the site for ultimate disposal in an authorized location.

Gaseous wastes are collected and held up for radioactive decay. Discharge to the environment after filtration is controlled to keep the offsite radiation dose within limits.

A.2 MAJOR STRUCTURES

Major structures on the plant site include the Reactor Building, Turbine Building, Auxiliary Building, Fuel Building, Control Building and Cooling Tower. Detailed descriptions of the reference PWR structures used in this study for the determination of waste volume estimates and demolition cost estimates are contained in subsequent sections.

A.2.1 Reactor Building

The nuclear containment structure is designed to house the primary nuclear system and is referred to as the Reactor Building (see Figure A.2-1).⁽³⁾ The Reactor Building is in the shape of a right circular cylinder. It has a hemispherical dome and a flat base slab with a central cavity and instrumentation tunnel.

⁽³⁾ Standardized Nuclear Unit Power Plant System - SNUPPS, Preliminary Safety Analysis Report, Volume 1, November 1974.

FIGURE A.2-1. Typical Reactor Building Outline

General Reactor Building structure dimensions are as follows:

Height	64.008 m	(~210 ft)
Diameter	42.672 m	(140 ft)
Wall Thickness	1.219 m	(~4 ft)
Dome Thickness	0.914 m	(3 ft)
Slab Thickness	3.048 m	(~10 ft)
Design Pressure	42.184 Mg/m ²	(60 psig)
Inner Steel Liner	6.35 mm	(1/4 in.)

The Reactor Building is constructed of reinforced concrete prestressed by post-tensioned tendons in the cylinder walls and dome. The interior is lined with steel plates welded to form a leak-tight barrier.

Numerous penetrations are provided for personnel, equipment, fuel transfer, piping and electrical access. These penetrations are attached to the Reactor Building liner plate and anchored into the concrete structure.

A.2.1.1 Reactor Building Internals

The Containment or Reactor Building consists of essentially two structures on a common foundation. One is the Containment^(a) whose primary function is to provide a leaktight vessel and biological shielding for normal and accident situations. The other structure is the interior structure, referred to as Containment Internals, whose primary function is to provide biological shielding around the nuclear steam supply system. The Containment Internals consist of the reactor cavity, biological shield, steam-generator and pressurizer compartments, and the refueling pool as shown in Figures A.2-2 through A.2-6. Supports for equipment, operating decks, access stairways, and platforms are included in the Containment Internals. The Containment Internals is a conventionally reinforced concrete structure. Floor slabs and gratings are supported by structural steel beams. The two structures,

(a) The Containment is defined as the Containment structure, liner and penetrations, the steam generator shell, the steam generator steam side instrumentation connections, and the steam, feedwater, blowdown and steam generator drain pipes within the Containment.

FIGURE A.2-2. Vertical Sections of the Containment Building

FIGURE A.2-3. Reactor Cavity and Biological Shield

FIGURE A.2-4. Reactor and Refueling Cavities

FIGURE A.2-5. Steam Generator Cavities

FIGURE A.2-6. Pressurizer Enclosure

on a common foundation, are structurally separated above the foundation by a gap based on considerations of maximum relative displacements during an earthquake. A reactor cavity and instrumentation tunnel are located below the foundation slab as shown in Figure A.2-7. A continuous access gallery is also provided beneath the foundation slab for installing and inspecting vertical tendons as shown in Figure A.2-8. The foundation slab is conventionally reinforced with high strength reinforcing steel as shown in Figures A.2-9 and A.2-10. The inside face of the concrete shell is steel-lined. The base liner is installed on top of the structural slab. The base slab provides support for the internal structures which in turn support the reactor vessel and other internal equipment and systems. The Containment Internals are designed in such a manner that the loads are carried by a combination of slab and shear wall action to the containment base slab. Some slabs are supported by structural steel members.

A.2.1.2 Supporting Structures, Walls and Shields (Reference Table A.2-1)

- The four secondary loops are physically separated inside the containment by concrete slabs and walls. The secondary compartment walls are anchored to the base mat by an anchorage system.
- Missile Shield Structure--This shield is a 0.737 m thick concrete slab with 25 mm thick steel facing. It is located approximately 1 m above the control rod drive mechanism.
- Generally, the minimum thickness is 0.6 m of concrete for the secondary shield, operating floor and refueling cavity walls.
- A fuel transfer penetration is between the refueling transfer canal in the Containment Internals and the spent fuel pool. The penetration consists of a 0.508 m stainless steel pipe installed inside a casing pipe. The casing pipe is provided with expansion joints and is connected to the containment liner. Reference Figure A.2-7.
- Reinforcing steel is deformed billet steel, conforming to ASTM Designation A-615. Minimum yield strength for Grade 60: $42,184 \text{ Mg/m}^2$ (60,000 psi). Grade 60 is used in the base slab, walls, dome, and internals of the Containment.

FIGURE A.2-7. Vertical Section of the Lower Containment Building

FIGURE A.2-8. Tendon Gallery, Plan View

FIGURE A.2-9. Containment Structure Base Slab Bottom Reinforcing

FIGURE A.2-10. Containment Structure Base Slab Top Reinforcing

TABLE A.2-1. Summary of Physical Data for Containment Internals

<u>Description</u>	<u>Material Quantity</u>
Biological Shield Concrete	734 m ³ (960 yd ³) 1764 Mg (1944 ton)
Base Slab Concrete	3078 m ³ (4026 yd ³) 7396 Mg (8153 ton)
Base Liner (steel)	193 m ³ (252 yd ³) 54 Mg (60 ton)
Reactor Cavity Liner (steel)	14.5 Mg (16 ton)
Floors and Cavity Liner Plates (steel)	139 Mg (153 ton)
CRD Missile Shield Concrete	41 m ³ (54 yd ³) 99 Mg (109 ton)
CRD Missile Shield Facing (steel)	11 Mg (12 ton)
Refueling Cavity Concrete	460 m ³ (602 yd ³) 1107 Mg (1220 ton)
Refueling Cavity Liner (steel)	17 Mg (19 ton)
Pressurizer Enclosure Concrete	410 m ³ (536 yd ³) 980 Mg (1085 ton)
Steam Generator Enclosures (4) Concrete	807 m ³ (1056 yd ³) 1940 Mg (2139 ton)

Materials

Basically, five types of materials are used in the construction of the Containment. These are:

Concrete: (a) $f'_c = 2109 \text{ Mg/m}^2$ (tendon gallery and personnel air lock)
 $f'_c = 3515 \text{ Mg/m}^2$ (base slab and reactor cavity)
 $f'_c = 4218 \text{ Mg/m}^2$ (walls and dome)

Reinforcing bars: ASTM-A-615-68, Grade 80

Steel liner plate: ASTM-A-442

Steel prestressing tendons: ASTM-A-421, Type B4

Structural steels

(a) The compressive strength, f'_c , refers to strength at 28 days if only cement was used and at 90 days when pozzolan was added.

- The stairs, gratings, and handrails inside Containment are galvanized.
- Floor slabs and certain other concrete and concrete block masonry surfaces within the Containment and other structures are coated to prevent possible absorption of contaminated fluids and to facilitate decontamination.

A.2.1.3 Sequence of Construction Events

The general construction procedure that was used in the erection of the Containment was as follows:

- 1) Excavation was performed.
- 2) The reactor vessel's pit and the incore instrument cable tunnel were constructed.
- 3) The base slab was constructed using placing techniques typical of large mat foundations. The tendon gallery was placed first in five circumferential segments. Following this, the base slab was poured to 140 mm below the finished floor. Bars were grouped where required to provide 203 mm by 203 mm openings for drop chutes extending from concrete conveyors. After the base slab was poured, carefully leveled steel beams were set to provide screeds for the final 140 mm of concrete and for the accurate attachment of the base liner plate.
- 4) The floor liner plate was installed and tested before the protective floor slab was placed.
- 5) Erection of the liner plate was followed by the welding and testing of the horizontal and vertical seams. After the installation of the wall tendon tubes and reinforcing steel, the wall was poured in continuous 360° - 3 m high lifts to 1.5 m below springline (see Figure A.2-1).
- 6) The concrete to about 45° on the dome was placed in circumferential segments, each about 5° high. The rest of the dome concrete was placed in two stages. First, a 152 mm thick layer was supported by the stiffened liner plates. After the installation of the tendon sheathing and reinforcing, the remaining concrete was placed.

A.2.2 Auxiliary Building

The Auxiliary Building is a rectangular, Category I^(a) structure consisting of two floors below grade and four floors above. The framing members above grade are structural steel. The exterior walls are constructed of concrete block masonry at the first floor and metal siding at the top floors. The below grade portion is constructed of reinforced concrete. The Auxiliary Building is separated from the Reactor Containment Building by a 76 mm expansion joint.

The Auxiliary Building houses the following major plant equipment:

Fourth Floor (Top)

- This floor houses ventilation equipment for the Fuel Building, Auxiliary Building and the Containment
- Access is provided for ion exchanger resin fill and radioactive filter removal

Third Floor

- Ion exchangers and radioactive fluid filters which are housed in concrete shielded vaults
- Boric acid evaporators for the boron recycle system
- Component cooling water surge tanks

Second Floor

- Two waste gas compressors
- Four waste gas decay tanks

(a) The design bases for individual structures and portions of structures are dependent on the systems which the structures support and house. Design bases for Category I structures are defined as being broadly within the following classifications: 1) support, 2) biological shield, 3) missile shield, 4) administrative access control, 5) atmosphere control, 6) liquid control, 7) seismic protection, and 8) wind protection. Each portion of a structure may have one or more of the above named design bases. The number and quantitative degree of design bases is dependent on the specific needs of components (locally) and systems (generally).

- A waste gas surge tank
- Radioactive waste evaporator
- Letdown heat exchanger
- The volume control tank
- The spent resin storage tank

Ground Floor

- Radiation sample room and chemical laboratory
- Treated waste monitor tanks
- Dirty waste monitor tank and pumps
- Service water booster pumps
- Spent fuel pool cooling pumps

Below Grade Floors

- Positive displacement and centrifugal charging pumps
- Chemical waste drain tank and pumps
- Reactor coolant drain tank pumps
- Clean waste receiver tank
- Dirty waste drain tank and pumps
- Auxiliary building drain tank and pumps
- Containment spray pumps
- Safety injection pumps
- Residual heat removal pumps
- Residual heat exchanger.

Figures A.2-11 through A.2-14 show various views of the Auxiliary Building.

The interior walls are constructed of concrete block masonry. In order to facilitate equipment removal, certain interior walls are provided with removable panels. Details of typical removable panels are shown in Figure A.2-15. Interior framing members below grade and all framing members above grade are structural steel. Auxiliary internals include interior walls, floors, gratings,

FIGURE A.2-11. Plan Views Auxiliary Building

FIGURE A.2-12. Auxiliary Building, Section A-A.

FIGURE A.2-13. Auxiliary Building, Section B-B

FIGURE A.2-14. Auxiliary Building, Section C-C.

FIGURE A.2-15. Removable Concrete Panels

reinforcing steel, duct work, and concrete. See Table A.2-2 for summary of physical data. Live loads for the design of the structural framing members are consistent with the intended use of the structure and the recommendations of the Uniform Building Code, 1967 Edition. The live loads that were used in the design of the structures are given in Table A.2-3.

Identification of Construction Materials

The following major structural materials are used in the construction of Category I structures:

Concrete^(b)

Auxiliary Building	$f'c = 3000 \text{ psi}$ (2109 Mg/m^2)
Fuel Building	$f'c = 3000 \text{ psi}$ (2109 Mg/m^2)
Reinforcing Steel	ASTM A 615, Grade 40 and 60
Structural Steel	ASTM A 36
High Strength Bolts	ASTM A 325
Stainless Steel Liner Plate	ASTM A 167, Type 304
Concrete Block Masonry	ASTM C 90, Grade PI

A.2.3 Fuel Building

The Fuel Building is a Category I structure and consists of four floors above grade. Reference Figures A.2-16 through A.2-21. The Fuel Building houses the following major plant equipment:

Fourth Floor (Top)

- A controlled access machine shop
- Storage room for solid radioactive waste
- Waste boiler
- Boric acid batching tank

^(b) The compressive strength, $f'c$, refers to strength at 28 days if only cement was used and at 90 days when pozzolan was added.

TABLE A.2-2. Summary of Physical Data for Auxiliary Building

<u>Description</u>	<u>Material</u>	<u>Quantity</u>
Primary Auxiliary Building		
Concrete	7,623 m ³	(9,970 yd ³)
	17,582 Mg	(19,381 ton)
Reinforcing Steel	734 Mg	(809 ton)
Walls (Steel)	24.2 Mg	(53,400 lb)
Roof (Steel)	16.3 Mg	(36,000 lb)
Stairs, Platforms, Tracks, Handrails, etc. (Iron)	33.2 Mg	(73,200 lb)
Structural Steel I-Beams	207 Mg	(228 ton)
Air Conditioning, Heating, Cooling and Ventilation:		
Components	64 Mg	(70 ton)
Ducts and Pipe (8.5 m)	20.4 Mg	(44,960 lb)

TABLE A.2-3. Live Loads - Auxiliary and Fuel Buildings

Auxiliary Building

Elevation 1.5 m (5 ft)		-(grade)
Elevation 7.6 m (25 ft)	221 kg/m ²	(250 psf)
Elevation 13.7 m (45 ft)	221 kg/m ²	(250 psf)
Elevation 18.6 m (61 ft)	221 kg/m ²	(250 psf)
Elevation 23.5 m (77 ft)	221 kg/m ²	(250 psf)
Elevation 28.3 m (93 ft)	221 kg/m ²	(250 psf)

Fuel Building

Elevation 13.7 m (45 ft)		-(grade)
Elevation 18.6 m (61 ft)	1,221 kg/m ²	(250 psf)
Solid Waste Drumming Area	6,835 kg/m ²	(1,400 psf)
CVCS Monitor Tank Area	2,929 kg/m ²	(600 psf)
Elevation 23.5 m (77 ft)	1,221 kg/m ²	(250 psf)
Cask Wash Pit	19,530 kg/m ²	(4,000 psf)
Elevation 28.3 m (93 ft)	1,221 kg/m ²	(250 psf)
Fuel Handling Area	4,882 kg/m ²	(1,000 psf)

FIGURE A.2-16. Fuel Building, Looking East

FIGURE A.2-17. Fuel Building, Looking North

FIGURE A.2-18. Plan Views--Fuel and Auxiliary Buildings, Grade

FIGURE A.2-19. Plan Views--Fuel and Auxiliary Buildings El. 18.6 m.

FIGURE A.2-20. Plan Views--Fuel and Auxiliary Buildings, El. 23.5 m

FIGURE A.2-21. Plan Views--Fuel and Auxiliary Buildings Operating Floor

A.2.3 Fuel Building (continued)

Third Floor

- Two waste concentrator holding tanks and pumps
- Concentrates holding tank for the boric acid evaporator

Second Floor

- The solid radioactive waste processing equipment
- CVCS monitor tanks and pumps
- Seal water heat exchangers
- Spent fuel pool heat exchangers
- The two boric acid tanks (the two tanks project up through the operating floor)
- The HVAC heating load center equipment for the Fuel Building

Ground Floor

- Component cooling water heat exchanger
- Component cooling water pumps
- Holdup tank recirculation pumps
- Gas stripper feed pumps
- Process steam boiler (extends up through the second floor)

The ground floor encloses a rail access at one end and a hatchway up through the operating floor. The CVCS holdup tanks at this elevation extend up to just below the operating floor.

The Fuel Building houses a 113 Mg (125 ton) overhead crane capable of handling the fuel cask. A fuel-handling crane runs on rails mounted on the operating floor. The travel of the overhead crane is restricted over the spent fuel pool by means of mechanical stops and administrative controls.

At the west side, the Fuel Building is structurally connected to the Auxiliary Building. Expansion joint bellows at the fuel transfer tube provide for the relative movement between containment, containment internals, and the spent fuel pool. Access for inspection and maintenance of the bellows is provided.

Exterior walls to elevation 19.2 m (63-ft) of the Fuel Building are pre-cast concrete panels and corrugated metal siding above that. The steel superstructure of the Fuel Building protects the spent fuel pool from the environment. Reference Table A.2-4 for Summary of Physical Data.

The Fuel Building also includes the following structural features:

- Spent fuel pool
- Cask loading pit
- New fuel storage pit
- Cask wash pit
- Three reinforced concrete vaults with 0.76 m (2-ft 6-in.) thick walls enclose the three Chemical Volume Control System (CVCS) holdup tanks.

Reference Figures A.2-18 through A.2-21 for locations of the areas listed above.

The spent fuel pool walls and base slab are constructed of thick (1.57 m to 2.0 m) reinforced concrete. The inside face of the walls and base slab are lined with 6.4 mm (1/4-in.) thick stainless steel liner plate to provide leak-tightness. Framing consists of structural steel with reinforced concrete floor slabs. Interior walls are concrete block masonry. Fuel Building internals include interior walls, base slabs, liner plates, floors, gratings, reinforcing steel, duct work and concrete. The live loads that were used in the design of the structures are given in Table A.2-3.

TABLE A.2-4. Summary of Physical Data for Fuel Building

Description	Material	Quantity
New Fuel Storage Racks	998 kg	(2,200 lb)
Spent Fuel Storage Racks	48,081 kg	(106,000 lb)
Fuel Handling System	18,470 kg	(40,720 lb)
Air Conditioning, Heating, Cooling and Ventilation		
Components, Ducts and Pipe	18,665 kg	(41,150 lb)
Cask Loading Pit Including Base Slab		
Concrete	269 m ³	(352 yd ³)
	646 Mg	(712 ton)
Spent Fuel Pool Including Base Slab		
Concrete	1,586 m ³	(2,075 yd ³)
	3,660 Mg	(4,034 ton)
Liner (steel)	37 Mg	(41 ton)
Three Reinforced Vaults for Three CVCS Holdup Tanks		
Concrete	1,015 m ³	(1,328 yd ³)
	2,341 Mg	(2,581 ton)
Concrete:		
13.7 m (45 ft) Elevation	105 m ³	(137 yd ³)
18.6 m (61 ft) Elevation	223 m ³	(292 yd ³)
23.5 m (77 ft) Elevation	187 m ³	(245 yd ³)
28.3 m (93 ft) Elevation	397 m ³	(519 yd ³)
Inner Hatchway Walls	54 m ³	(70 yd ³)
Outer Hatchway Walls	213 m ³	(278 yd ³)
28.3 m (93 ft) Elevation Walls	118 m ³	(154 yd ³)

A.2.4 Turbine Building

The Turbine Building is a structural steel framed building with reinforced concrete slab floors. The turbine pedestal is constructed of reinforced concrete. The building consists generally of two floors above grade. It houses the following major equipment:

- The turbine generator and other equipment associated with the power conversion system
- Emergency diesel generator units
- Engineered Safety Features (ESF)
- Steam generator auxiliary pumps
- Cable trays for ESF system.

The Turbine Building is provided with a crane having a 181 Mg (200-ton) main hook and 23 Mg (25-ton) auxiliary hook.

The emergency diesel generators and ESF switch gear are housed in a two-floor structure located at the north end of the turbine pedestal. This structure is framed separately from the Turbine Building by a 76 mm (3 in.) expansion joint.

A.2.5 Control Building

The Control Building is a Category I structure and consists of four floors above grade (elevation 13.7 m) as shown in Figure A.2-22. The Auxiliary Building is located to the east of the Control Building and the two buildings are structurally connected. The Turbine Building, located to the west, is separated from the Control Building by a 76 mm (3-in.) gap which is adequate to prevent interaction between the two structures during an earthquake.

FIGURE 2.2-22. Isometric View of the Control Building

The Control Building houses the following major items of equipment:

Elevation 13.7 m

- Radiological chemical laboratory
- Radiation sample room
- Counting room
- Decontamination room
- Showers
- Locker rooms

Elevation 18.6 m and 19.8 m

- Battery rooms
- Electrical auxiliaries
- Telephone equipment
- Mechanical room

Elevation 23.5 m

- Cable spreading room
- Computer room

Elevation 28.3 m

- Control room
- Control room annex
- Supervisor's room
- Chart room
- Water sampling laboratory.

The framing members for the Control Building are structural steel. Floor slabs are reinforced concrete and precast prestressed concrete panels. Concrete block masonry is used for walls.

A.2.6 Shop and Warehouse

This building is basically one-story high and is a steel frame building. The building contains the following shops, offices and equipment:

- Warehouse
- A 9 Mg capacity overhead crane
- Paint shop

- General shop
- Foreman's office
- Lockers and lunchroom.

A.2.7 Chlorine Building

This building is a steel frame building with precast concrete walls. It houses the following equipment:

- Chlorine tanks
- Chlorine dispensers
- Chlorine evaporators.

A.2.8 Administration Building

This building is a two-story steel frame structure. It houses the gatehouse and offices for the plant superintendent and staff.

A.2.9 Condensate Demineralizer Building

The Condensate Demineralizer Building is a rectangular structure approximately 43.3 m long x 10.4 m wide consisting of two floors below grade and one floor above grade. Reference Figures A.2-23 and A.2-24. The framing members above grade are structural steel with metal siding exterior walls. The below grade walls are constructed of reinforced concrete.

This building houses the equipment required for condensate demineralization and expended resin disposal. Reference Table A.2-5 for an equipment listing. A resin backwash handling system has provision for transfer of potentially radioactive resin directly from the backwash receiving tank to a waste disposal truck. The building ventilation system is designed such that all spaces that may contain expended resin are exhausted through a HEPA filter. The combined ventilation exhaust from the building is monitored continuously.

A.2.10 Cooling Tower

A hyperbolic natural draft cooling tower is used to dissipate waste heat from the turbine condenser system. Its height is 152 m (499 ft) above grade and the diameter at the base is 119 m (390 ft). The tower is constructed of reinforced concrete. The nominal wall thickness of the tower is 0.254 m (10 in.),

FIGURE A.2-23. Plan Views, Condensate Demineralizer Building

FIGURE A.2-24. Section Views, Condensate Demineralizer Building

TABLE A.2-5. Equipment List: Condensate
Demineralizer Building

<u>Name</u>	<u>Rating or Size (Each)</u>	
Precoat Tank	4.54 m ³	(1,200 gal)
Filter Demineralizer Vessels (8)	14.01 m ³ /min	(3,700 gpm) throughput
	1.8 diam x 3 m high	(6 ft diam x 10 ft high)
Backwash Filtrate Tanks (2)	1.51 m ³	(400 gal)
Backwash Receiving Tank with Agitator	56.78 m ³	(15,000 gal)
Main Sump Pump	0.23 m ³ /min	(60 gpm)
Holding Pumps (8)	0.23 m ³ /min	(60 gpm)
Filter Feed Pumps (2)	0.57 m ³ /min	(150 gpm)
Precoat Pump	5.30 m ³ /min	(1,400 gpm)
Backwash Pump	4.54 m ³ /min	(1,200 gpm)
Backwash Filters (2)	1.5 m x 7.3 m x 2.1 m	(5 ft x 24 ft x 9 ft)

but it thickens at the top and bottom. Seven hundred fifty thousand concrete asbestos plates are hung vertically in the lower 18.3 m portion of the structure. The plates provide 1.58 km² of cooling surface. The tower is supported by 88 diagonal legs that are 1.07 m in diameter and 12.5 m high. The tower contains 30,964 m³ of concrete and 2722 Mg of reinforcing steel. Figure A.2-25 presents other design statistics.⁽⁵⁾

A.3 MAJOR EQUIPMENT

Major pieces of equipment include the Reactor Vessel and Internals, four Steam Generators, four Reactor Coolant Pumps, Pressurizer and the Reactor Coolant Piping.

A.3.1 Reactor Vessel and Internals

The reactor vessel is cylindrical, with a welded hemispherical bottom head and a removable, bolted, flanged and gasketed, hemispherical upper head. The reactor vessel flange and head are sealed by two hollow metallic O-rings. The vessel contains the core, core support structures, control rods, and other parts directly associated with the core. Inlet and outlet nozzles are spaced evenly around the vessel. Outlet nozzles are located on the vessel to

⁽⁵⁾ TROJAN, Final Safety Analysis Report, Volume 6, Amendment 3, Portland General Electric Co., Portland, OR, September, 1973.

FIGURE A.2-25. Natural Draft Cooling Tower

facilitate optimum layout of the Reactor Coolant System equipment. The inlet nozzles are tapered from the coolant loop vessel interfaces to the vessel inside wall to reduce loop pressure drop.

The bottom head of the vessel contains penetration nozzles for connection and entry of the nuclear incore instrumentation. Each nozzle consists of a tubular member made of an Inconel-stainless steel composite tube. Each tube is attached to the inside of the bottom head with a partial penetration weld.

Internal surfaces of the vessel which are in contact with primary coolant are weld overlaid with 3.96 mm (0.156 in.), minimum, of stainless steel or Inconel. The exterior of the reactor vessel is insulated with canned stainless steel reflective sheets. The insulation is 76 mm (3 in.) thick and contoured to enclose the top, sides and bottom of the vessel. The reactor pressure vessel is shown in Figure A.3-1. All the insulation modules are removable but access to vessel side insulation is limited by the surrounding concrete. Reactor vessel design parameters are shown in Table A.3-1.⁽⁶⁾

FIGURE A.3-1. Reactor Pressure Vessel

⁽⁶⁾ TROJAN, Final Safety Analysis Report, Volume 3, Amendment 3, Portland General Electric Co., Portland, OR, September 1973.

TABLE A.3-1. Reactor Vessel Design Parameters

Overall Height of Vessel and Closure Head (Bottom Head OD to Top of Control Rod Mechanism Adapter)	13.36 m
Thickness of Insulation, minimum	76 mm
Number of Reactor Closure Head Studs	54
Diameter of Reactor Closure Head Studs	0.178 m
ID of Flange	4.242 m
OD of Flange	5.207 m
ID at Shell	4.394 m
Inlet Nozzle ID	0.699 m
Outlet Nozzle ID	0.737 m
Clad Thickness, minimum	3.97 mm
Lower Head Thickness, minimum	0.136 m
Vessel Belt-Line Thickness, minimum	0.216 m
Closure Head Thickness	0.165 m
Weight, without Head and Insulation	308.4 Mg
Weight of Head without Insulation including Stud Nuts and Washers	88.5 Mg

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and neutron shield pad assembly), the upper core support structure and the in-core instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, direct flow past the fuel elements and to the pressure vessel head, provide gamma and neutron shielding and guide the in-core instrumentation (see Figure A.3-2).

All major material for the reactor internals is type 304 stainless steel. Parts not fabricated from 304 stainless steel are bolts and dowel pins fabricated from type 316 stainless steel, radial support key bolts fabricated from Inconel and hold down springs fabricated from series 403 stainless steel.

FIGURE A.3-2. Reactor Vessel Internals

A.3.1.1 Lower Case Support Structure

The major containment and support member of the reactor internals is the lower core support structure, shown in Figure A.3-3. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the neutron shield pads, and the core support that is welded to the core barrel. The lower core support structure is supported at its upper flange from a ledge in the reactor vessel head flange and its lower end is restrained from transverse motion by a radial support system attached to the vessel wall.

Within the core barrel are an axial baffle and a lower core plate, both of which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core support structure and principally the core barrel serve to provide passageways and control for the coolant flow as shown in Figure A.3-4. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a 51 mm thick member through which the necessary flow distribution holes for each fuel assembly are machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the core support of the core barrel in order to provide stiffness and to transmit the core load to the core support. Adequate coolant distribution is obtained through the use of the lower core plate and core support.

The neutron shield pad assembly consists of four pads that are bolted and pinned to the outside of the core barrel. These pads are constructed of type 304 stainless steel and are approximately 0.91 to 1.22 m wide x 3.76 m long x 69 to 71 mm thick. The pads are located azimuthally to provide the required degree of vessel protection. Rectangular specimen guides in which material surveillance samples are inserted and irradiated during reactor operation are bolted and pinned to the pads.

FIGURE A.3-3. Core Barrel and Lower Core Support Assembly

FIGURE A.3-4. Lower Core Support Assembly

The radial support system of the core barrel is accomplished by "key" and "keyway" joints to the reactor vessel wall. At six equally spaced points around the circumference, an Inconel clevis block is welded to the vessel inner diameter. An Inconel insert block is bolted to each of these clevis blocks, and has a "keyway" geometry. Opposite each of these is a "key" which is welded to the lower core support. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction.

The approximate weight of the entire lower core support structure is 127 Mg.

A.3.1.2 Upper Core Support Assembly

The upper core support assembly, shown in Figures A.3-5 and A.3-6, consists of the upper support assembly and the upper core plate between which are contained support columns and guide tube assemblies. The support columns establish the spacing between the upper support assembly and the upper core plate and are fastened at top and bottom to these plates. The support columns transmit the mechanical loadings between the upper support and upper core plate. The approximate weight of the entire upper core support assembly is 54.4 Mg.

The guide tube assemblies shield and guide the control rod drive shafts and control rods. They are fastened to the upper support and are guided by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the upper guide tube which is attached to the upper support.

The upper core support assembly, which can be removed as a unit, is positioned in its proper orientation with respect to the lower support structure by slots in the upper core plate which engage flat-sided upper core plate alignment pins which are welded into the core barrel. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 90° from each other. As the upper support structure is lowered into the lower internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies and control rods are thereby assured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly.

A.3.1.3 In-Core Instrumentation Support Structure

The in-core instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom. The approximate weight of the entire structure is 5 Mg.

FIGURE A.3-5. Upper Core Support Structure

FIGURE A.3-6. Plan View of Upper Core Support Structure

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to in-line columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are stainless steel tubes.

In addition to the upper in-core instrumentation, there are reactor vessel bottom port columns which carry the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 3.66 m and the trailing ends of the thimbles (at the seal line) are extracted approximately 4.57 m during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere. Mechanical seals between the retractable thimbles and conduits are provided at the seal line.

A.3.2 Steam Generators (4 each)

The approximate total weight of each steam generator is 312 Mg. The steam generator shown in Figures A.3-7 and A.3-8 is a vertical shell and U-tube evaporator with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tube sheet. Manways are provided for access to both sides of the divided head. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel. The unit is primarily carbon steel. The heat transfer tubes and the divider plate are Inconel and the interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel. The primary side of the tube sheet is weld clad with Inconel.

The steam drum has two bolted and gasketed access openings for inspection and maintenance of the dryers which can be disassembled and removed through the opening.

Table A.3-2 provides steam generator design data.

FIGURE A.3-7. Steam Generator

FIGURE A.3-8. Steam Generator Supports

TABLE A.3-2. Steam Generator Design Data

Total heat transfer surface area	4786 m ²	(51,500 ft ²)
Overall height	20.63 m	(67.67 ft)
Number of U-tubes	3388	
U-tube outer diameter	22.2 mm	(0.875 in.)
Tube wall thickness, nominal	1.27 mm	(0.050 in.)
Number of manways	4	
ID of manways	0.406 m	(16 in.)
Number of handholes	2	
ID of handholes	0.152 m	(6 in.)

A.3.3 Reactor Coolant Pumps (4 each)

The reactor coolant pump, shown in Figures A.3-9 and A.3-10, is a vertical, single stage, centrifugal, shaft seal pump designed to pump large volumes of main coolant at high temperatures and pressures. Reactor coolant pump data is given in Table A.3-3.

The pump consists of three areas from bottom to top. They are the hydraulics, the shaft seals, and the motor.

- 1) The hydraulic section consists of an impeller, diffuser, casing, thermal barrier, heat exchanger, lower radial bearing, bolting ring, motor stand, and pump shaft.
- 2) The shaft seal section consists of three devices. They are the No. 1 controlled leakage, film riding face seal and the No. 2 and No. 3 rubbing face seals. These seals are contained within the main flange and seal housing.
- 3) The motor section consists of a vertical solid shaft, squirrel cage induction type motor, an oil lubricated double Kingsbury type thrust bearing, two oil lubricated radial bearings, and a flywheel.

TABLE A.3-3. Reactor Coolant Pump Data

Capacity per pump	335 m ³ /min	(88,500 gpm)
Developed head	84.4 m of water	(277 ft of water)
Discharge nozzle, ID	0.699 m	(27-1/2 in.)
Suction nozzle, ID	0.787 m	(31 in.)
Overall unit height	8.702 m	(28.55 ft)
Water volume	1.586 m ³	(56 ft ³)
Weight, dry	85.37 Mg	(188,200 lb)
Motor		
Type	AC induction, single speed, air cooled	

Attached to the bottom of the pump shaft is the impeller. The reactor coolant is drawn up through the impeller, discharged through passages in the diffuser, and out through the discharge nozzle in the side of the casing. Above the impeller is a thermal barrier heat exchanger which limits heat transfer between hot system water and seal injection water.

The motor is an air-cooled, Class B thermalastic epoxy insulated, squirrel cage induction motor. The rotor and stator are of standard construction. The top of the motor consists of a flywheel and an antireverse rotation device.

All parts of the pump in contact with the reactor coolant are austenitic stainless steel except for seals, bearings and special parts. Component cooling water is supplied to the two oil coolers on the pump motor and to the pump thermal barrier heat exchanger.

The pump shaft, seal housing, thermal barrier, bolting ring and motor stand can be removed from the casing as a unit without disturbing the reactor coolant piping. The flywheel is available for inspection by removing the cover.

A.3.4 Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads constructed of carbon steel, with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant. A cutaway of the pressurizer is shown in Figure A.3-11.

The surge line nozzle and removable electric heaters are installed in the bottom head. The heaters are removable for maintenance or replacement. A thermal sleeve is provided to minimize stresses in the surge line nozzle.

The pressurizer surge line connects the pressurizer to one reactor hot leg. The line enables continuous coolant volume pressure adjustments between the RCS and the pressurizer.

A.3.5 Reactor Coolant Piping (See Figure A.3-12)

Principal design data for the reactor coolant piping are given in Table A.3-4. Major system piping data are shown in Table A.3-5.

FIGURE A.3-11. Cutaway of Pressurizer

FIGURE A.3-12. Reactor Coolant System Flow Schematic

TABLE A.3-4. Reactor Coolant Piping Design Parameters

Reactor inlet piping, ID	0.699 m	(27.5 in.)
Reactor inlet piping, nominal wall thickness	58.9 mm	(2.32 in.)
Reactor outlet piping, ID	0.737 m	(29 in.)
Reactor outlet piping, nominal wall thickness	62.0 mm	(2.44 in.)
Coolant pump suction piping, ID	0.737 m	(31 in.)
Coolant pump suction piping, nominal wall thickness	65.8 mm	(2.59 in.)
Pressurizer surge line piping, nominal pipe size	0.356 m	(14 in.)
Pressurizer surge line piping, nominal wall thickness	35.7 mm	(1.406 in.)

TABLE A.3-5. Major System Piping Data

System	Size, in.	Length		Weight	
		m	ft	kg	lb
Reactor Coolant	27 to 31 ID	81	267	100,698	222,000
Piping	2 to 14 OD	677	2,200	11,793	26,000
Residual Heat Removal	2 to 14 OD	194	637	11,929	26,300
Chem. and Volume Control	2 to 8 OD	2103	6,900	17,763	39,160
Emergency Core Cooling	2 to 24 OD	1923	6,310	100,698	222,000
Containment Spray	2 to 14 OD	1210	3,970	66,224	146,000
Auxiliary Feedwater	2 to 10 OD	604	1,980	14,361	31,660
Spent Fuel Pool Cooling	2 to 6 OD	594	1,950	3,783	8,340
Condensate Facility	2 to 8 OD	518	1,700	38,782	85,500
Station Service	2 to 20 OD	1779	5,835	59,874	132,000
Component Cooling	2 to 24 OD	4887	16,035	168,736	372,000
Service Cooling	2 to 18 OD	427	1,400	25,107	55,352
Makeup Water System	2 to 6 OD	998	3,275	5,625	12,400

The RCS piping includes those sections of piping interconnecting the reactor vessel, steam generator, and reactor coolant pump. It also includes the following:

- 1) Charging line and alternate charging line from the isolation valve up to the branch connections on the reactor coolant loop.
- 2) Letdown line and excess letdown line from the branch connections on the reactor coolant loop to the isolation valve.
- 3) Pressurizer spray lines from the reactor coolant cold legs to the spray nozzle on the pressurizer vessel.
- 4) Residual heat removal lines to or from the reactor coolant loops up to the designated isolation or check valve.
- 5) Safety injection lines from the designated isolation or check valve to the reactor coolant loops.
- 6) Accumulator lines from the designated isolation or check valve to the reactor coolant loops.
- 7) Resistance temperature detector manifold bypass loop piping.
- 8) Loop fill, loop drain, sample, and instrument lines to or from the designated isolation valve to or from the reactor coolant loops.
- 9) Pressurizer surge line from one reactor coolant loop hot leg to the pressurizer vessel inlet nozzle.
- 10) Resistance temperature detector scoop element, pressurizer spray scoop, sample connection with scoop, reactor coolant temperature element installation boss, and the temperature element well itself.
- 11) All branch connection nozzles attached to reactor coolant loops.
- 12) Pressure relief lines from nozzles on top of the pressurizer vessel up to and through the power-operated pressurizer relief valves and pressurizer safety valves.

- 13) Seal injection water and labyrinth differential pressure lines to or from the reactor coolant pump inside reactor Containment.
- 14) Auxiliary spray line from the isolation valve to the pressurizer spray line header.
- 15) Sample lines from pressurizer to the isolation valve.

A.3.6 Fuel and Auxiliary Building Major Equipment

Table A.3-6 lists the major equipment contained within the Fuel Storage and Auxiliary Buildings. It is expected that this equipment will be contaminated since it is in direct communication with the primary system.

TABLE A.3-6. Equipment List: Spent Fuel Storage and Auxiliary Building

Name	Weight, (each)		Overall Dimensions, (each)
	lb	kg	
Waste Gas Compressor (2)	8,000	3,629	3 x 1.2 x 1.5 m (10 x 4 x 5 ft)
Pool Purification Filter (1)	360	163	279 mm (11 in.) dia x 1.17 m (46 in.)
Pool Skimmer Filter (1)	150	68	229 mm (9 in.) dia x 1.14 m (45 in.)
Pool Demineralized Filter (1)	360	163	279 mm (11 in.) dia x 1.17 m (46 in.)
Dirty Waste Filter (2)	76	34	178 mm (7 in.) dia x 0.91 m (36 in.)
Clean Waste Filter (1)	67	30	178 mm (7 in.) dia x 0.66 m (26 in.)
Reactor Coolant Drain Filter (1)	350	159	406 mm (16 in.) dia x 1.42 m (56 in.)
Reactor Coolant Filter (1)	200	91	381 mm (15 in.) dia x 1.30 m (51 in.)
Boric Acid Filter (1)	200	91	381 mm (15 in.) dia x 1.30 m (51 in.)
Ion Exchange Filter (1)	150	68	305 mm (12 in.) dia x 1.02 m (40 in.)
Condensate Filter (2)	150	68	305 mm (12 in.) dia x 1.02 m (40 in.)
Seal Water Filter (1)	400	181	406 mm (16 in.) dia x 1.68 m (66 in.)
Seal Water Injection Filter (2)	1,650	748	254 mm (10 in.) dia x 1.91 m (75 in.)
Component Cooling Water Hx. (2)	70,000	31,751	1499 mm (59 in.) dia x 9.75 m (32 ft)
Spent Fuel Pond Hx. (2)	6,100	2,767	508 mm (20 in.) dia x 5.79 m (19 ft)
Seal Water Hx. (1)	1,700	771	356 mm (14 in.) dia x 4.27 m (14 ft)
Letdown Hx. (1)	1,900	862	457 mm (18 in.) dia x 5.49 m (18 ft)
Residual Heat Removal Hx. (2)	23,100	10,478	914 mm (36 in.) dia x 91.44 m (30 ft)
Auxiliary Building Sump Pump (2)	1,300	590	4.57 m (15 ft) long
Residual Heat Removal Pumps (2)	6,800	3,084	2.74 m (9 ft) long
Containment Spray Pumps (2)	6,800	3,084	2.74 m (9 ft) long
Service Water Booster Pumps (4)			
Spent Fuel Pool Cooling Pumps (2)	900	408	1.52 x 0.46 x 0.61 m (5 x 1.5 x 2 ft)
Spent Fuel Pool Purification Pump (1)	1,000	454	1.52 x 0.46 x 0.61 m (5 x 1.5 x 2 ft)
Spent Fuel Pool Skimmer Pump (1)	700	318	1.22 x 0.46 x 0.61 m (4 x 1.5 x 2 ft)
Component Cooling Water Pumps (3)	15,000	6,804	3.15 x 1.42 x 1.60 m (10.3 x 4.6 x 5.3 ft)
Component Cooling Makeup Pumps (2)	800	363	1.22 x 0.30 x 0.61 m (4 x 1 x 2 ft)
Primary Water Makeup Pumps (2)	800	363	1.22 x 0.30 x 0.61 m (4 x 1 x 2 ft)
Reactor Coolant Drain Tank Pumps (2)	500	227	1.22 x 0.30 x 0.60 m (4 x 1 x 2 ft)
Auxiliary Building Drain Tank Pumps (2)	200	91	0.91 x 0.30 x 0.30 m (3 x 1 x 1 ft)

**TABLE A.3-6. Equipment List: Spent Fuel Storage
and Auxiliary Building (cont'd)**

Name	Weight, (each)		Overall Dimensions, (each)
	lb	kg	
Dirty Waste Drain Tank Pumps (2)	500	227	1.22 x 0.30 x 0.61 m (4 x 1 x 2 ft)
Dirty Waste Monitor Tank Pumps (2)	200	91	0.91 x 0.30 x 0.30 m (3 x 1 x 1 ft)
Clean Waste Rec. Tank Pumps (2)	500	227	1.22 x 0.30 x 0.61 m (4 x 1 x 2 ft)
Chemical Waste Drain Tank Pumps (2)	200	91	0.91 x 0.30 x 0.30 m (3 x 1 x 1 ft)
Treated Water Monitor Tank Pumps (2)	230	104	0.91 x 0.30 x 0.30 m (3 x 1 x 1 ft)
Waste Concentrates Holding Tank Pumps (1)	230	104	0.91 x 0.30 x 0.30 m (3 x 1 x 1 ft)
Safety Injection Pumps (2)	8,600	3,901	4.37 x 4.02 x 1.07 m (14.3 x 3.3 x 3.5 ft)
Centrifugal Charging Pumps (2)	17,090	7,752	5.44 x 1.27 x 1.40 m (17.8 x 4.2 x 4.6 ft)
Boric Acid Transfer Pumps (2)	618	280	1.32 x 0.38 x 0.53 m (4.3 x 1.25 x 1.75 ft)
Holdup Tank Recirc. Pump (1)	636	288	1.32 x 0.38 x 0.86 m (4.3 x 1.25 x 2.8 ft)
Gas Stripper Feed Pumps (2)	500	227	0.66 x 0.56 x 0.36 m (2.17 x 1.83 x 1.17 ft)
Chem. Vol. Cont. Sys. Mont. Tank Pumps (2)	600	272	1.32 x 0.38 x 0.53 m (4.3 x 1.25 x 1.75 ft)
Posit. Displace. Charging Pump (1)	17,700	8,029	4.27 x 1.73 x 1.32 m (14.0 x 5.67 x 4.33 ft)
Concent. Holding Tank Transfer Pumps (2)	200	91	0.46 x 0.48 x 0.25 m (1.5 x 1.58 x 0.83 ft)
Strain Generator Blowdown Tank (1)	2,000	907	1.83 dia x 3.05 m (6 x 10 ft)
Component Cooling Water Surge Tanks (2)	2,000	907	2.13 dia x 2.44 m (7 x 8 ft)
Contain. Spray Additive Tanks	2,500	1,134	2.74 dia x 3.05 m (9 x 10 ft)
Component Cooling Water Chem. Add. Tank (1)	1,000	454	0.61 dia x 1.52 m (2 x 5 ft)
Fuel Pool Demineralizer	2,200	998	1.22 dia x 3.05 m (4 x 10 ft)
Reactor Coolant Drain Tank	1,700	771	0.91 dia x 2.44 m (3 x 8 ft)
Auxiliary Drain Tank	2,100	953	1.83 dia x 2.74 m (6 x 9 ft)
Clean Waste Rec. Tanks (2)	10,958	4,970	3.05 dia x 9.14 m (10 x 30 ft)
Chem. Waste Drain Tank (1)	5,400	2,449	3.05 dia x 1.52 m (10 x 5 ft)
Spent Resin Storage Tank (1)	6,800	3,084	2.74 dia x 3.35 m (9 x 11 ft)
Treated Waste Monitor Tank (2)	11,200	5,080	3.05 dia x 7.92 m (10 x 26 ft)
Dirty Waste Drain Tank (1)	6,540	2,966	3.05 dia x 3.96 m (10 x 13 ft)
Dirty Waste Monitor Tank (1)	5,800	2,631	3.05 dia x 3.66 m (10 x 12 ft)
Waste Concent. Holding Tank (2)	2,100	953	1.82 dia x 3.05 m (6 x 10 ft)
Waste Gas Surge Tank	900	408	0.91 dia x 1.83 m (3 x 6 ft)
Waste Gas Decay Tank	10,800	4,899	3.05 x 4.88 m (10 x 16 ft)
Boron Inj. Tank (1)	28,500	12,927	1.67 dia x 3.81 m (5.5 x 12.5 ft)
Chem. Vol. Cont. Sys. Demin. (3)	1,050	476	0.66 dia x 1.65 m (2.17 x 5.40 ft)
Resin Fill Tank (1)	260	118	1.63 dia x 1.88 m (5.33 x 6.17 ft)
Volume Control Tank (1)	4,900	2,223	2.29 dia x 3.18 m (7.50 x 10.4 ft)
Boric Acid Tanks (2)	20,000	9,072	3.66 dia x 10.36 m (12 x 34 ft)
Boric Acid Batching Tank (1)	1,450	658	1.22 dia x 1.78 m (4 x 5.8 ft)
Evap. Ion Exch. (5)	1,050	476	0.66 dia x 1.65 m (2.17 x 5.40 ft)
Concentrates Holding Tank	3,500	1,588	1.68 dia x 2.36 m (5.5 x 7.8 ft)
CVCS Holdup Tanks (3)	30,000	13,608	5.49 dia x 10.36 m (18 x 34 ft)
CVCS Monitor Tanks (2)	20,000	9,072	6.10 dia x 3.05 m (20 x 10 ft)
Electric Steam Boiler (1)	13,000	5,897	3.05 x 2.13 x 6.71 m (10 x 7 x 22 ft)
Clean Radioactive Waste Evap.	40,000	18,144	5.79 x 2.74 x 3.66 m (19 x 9 x 12 ft)
Boric Acid Evap. and Gas Stripper (Skid Mounted)	21,000	9,525	4.57 x 3.35 x 2.79 m (15 x 11 x 9.2 ft)

A.3.7 Reactor Building Major Equipment

Table A.3-7 lists the major equipment located in the Reactor Building.

TABLE A.3-7. Equipment List: Reactor Building

Name	Weight, (each)		Overall Dimensions, (each)
	lb	kg	
Regenerative Hx	6,600	2,994	0.36 dia x 5.49 m (1.17 x 18 ft)
Excess Letdown Hx (1)	1,600	726	0.28 dia x 3.35 m (0.92 x 11 ft)
Containment Sump Pumps (2)	1,400	635	1.83 m (6 ft) long
Reactor Cavity Drain Pump	800	363	4.57 m (15 ft) long
Pressurizer Relief Tank (1)	27,200	12,338	3.25 dia x 8.25 m (10.7 x 27 ft)
Safety Inj. Sys. Accumulator (4)	76,500	34,700	3.66 dia x 6.4 m (12 x 21 ft)

A.4 HEATING, AIR CONDITIONING, AND VENTILATION SYSTEMS

The heating, ventilation, and cooling systems are designed to provide adequate temperature control as required and to maintain equipment temperatures at levels compatible with normal equipment operating requirements.

The information contained in Figures A.4-1 through A.4-7 have not been converted to metric units to avoid unnecessary confusion, since these data are included for illustration only and are not essential to the study results.

A.4.1 HVAC-Control Building

The Control Building is served by the following heating, ventilating, and cooling systems shown in Figures A.4-1 and A.4-2.

- 1) The Control Room Emergency Cooling System is designed to provide cooling to the control room under certain emergency conditions.
- 2) The Control Room Normal Air-Conditioning System provides cooling to the control room during normal plant operation.
- 3) The Control Room Special Exhaust System is designed to remove smoke or other contaminants from the room.

- 4) The Control Building Ventilation Supply System is designed to provide supply air, which can be heated if necessary, to spaces in the building that do not require air-conditioning.
- 5) The Control Building Air-Conditioning Supply System supplies conditioned air to the telephone equipment room and the potentially contaminated controlled access areas in the Control Building.
- 6) The Control Building Exhaust System exhausts the potentially contaminated controlled access areas in the Control Building.
- 7) The Battery Room Exhaust System ensures against the accumulation of hydrogen gas in the battery rooms.
- 8) The Electrical Auxiliary Room Coolers cool the rooms containing emergency electrical switchgear.
- 9) The Cable Spreading Room Coolers cool the cable spreading room.
- 10) The Computer Room Air-Conditioning Unit cools the computer room.

A.4.2 HVAC-Fuel and Auxiliary Buildings

The Fuel and Auxiliary Buildings are served by the following heating, ventilating and cooling systems shown in Figures A.4-3, A.4-4, and A.4-5.

- 1) Pump cooling units designed to cool rooms containing pumps required to function during emergency conditions, and which consists of fancoil units.
- 2) A pump cooling unit designed to cool a room containing a pump which has no safety function.
- 3) Fuel and Auxiliary Building Supply System which is a single system supplying outdoor air and heating to both the Fuel and Auxiliary Buildings.
- 4) Fuel and Auxiliary Building Exhaust System which exhausts all spaces except the spent fuel pool area in both the Fuel and Auxiliary Buildings. The radioactive waste treatment area is included in this space.
- 5) Spent Fuel Pool Exhaust System which exhausts the spent fuel pool area in the Fuel Building.
- 6) Maintenance Area Supply Cooling System which air-conditions the maintenance area.

FIGURE A.4-1. Air Flow Diagram Control Building

GENERAL NOTES:

1. ALL FLAT TYPE FILTERS SHALL BE CLASS I FIRE RETARDANT
2. SELF-CONTAINED AIR-CONDITIONING UNIT

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LIST OF SYSTEMS FOR AND REACTIONARY BUILDINGS

1. SMOKE DETECTION SYSTEMS
- AB 1. SMOKE DETECTION SYSTEMS
- AB 2. SMOKE DETECTION SYSTEMS
- AB 3. SMOKE DETECTION SYSTEMS
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- AB 100. SMOKE DETECTION SYSTEMS

FIGURE A.4-3. Air Flow Diagram
Fuel Building

- GENERAL NOTES
1. SYSTEM AB 3 FANS VC BDA B C & D DESIGN IS BASED ON ANY THREE EXHAUST FANS NORMALLY OPERATING WITH ONE FAN AS STANDBY.
 2. SYSTEM AB 4 FANS VC BDA B & D DESIGN IS BASED ON ANY ONE (1) EXHAUST FAN NORMALLY OPERATING WITH ONE FAN AS STANDBY.
 3. FOR SYSTEM IDENTIFICATION REFER TO FIGURE A-4.3 PAGE 3 OF LIST OF SYSTEMS.

FIGURE A.4-4. Air Flow Diagram
Auxiliary Building

FIGURE A.4-6. Air Flow Diagram Turbine Building

[illegible]

FIGURE A.4-7. HVAC Equipment List:
Turbine Building

The Fuel and Auxiliary Building Heating and Ventilation System consists of a single outdoor air supply system and two exhaust systems. One exhaust system exhausts all areas of the building except the spent fuel pool area and the other exhausts only the spent fuel pool area. All air is exhausted through a common vent at the top of the Containment. Automatic continuous radiation monitoring is provided for the exhaust ducts. The systems are designed for continuous operation with the controls located in the Auxiliary Building.

The common supply system consists of two parallel 50-percent capacity, vane-axial, direct-driven fans, which take in air from an outdoor air intake through an automatic roll filter and electric heating coil. The incoming air is directed to all parts of the buildings through a ductwork system. The supply system is designed to supply approximately $2350 \text{ m}^3/\text{min}$ (83,000 cfm) of outside air with both fans operating. This system includes a fan-coil unit which draws air from the area containing the holdup tanks in the Fuel Building and recirculates cooled air through ductwork to the areas having high heat loads. Cooling water for the coil in the unit is supplied from the Service Water System.

The exhaust system which serves the spent fuel pond area consists of exhaust inlets located around the perimeter of the spent fuel pool which are connected by ductwork to two parallel exhaust air plenums each of which is designed to exhaust $594 \text{ m}^3/\text{min}$ (19,375 cfm). Each plenum contains an exhaust fan, a bank of carbon adsorbers, two banks of high efficiency particulate air (HEPA) filters, an automatic roll filter, and motorized isolation dampers at each end of the filter trains. Backdraft dampers are installed in each plenum to prevent reverse flow through an idle plenum. The system is designed for 100 percent capacity operation with one exhaust plenum in service while the other serves as a standby.

The exhaust system which serves the remaining parts of the Fuel and Auxiliary Buildings consists of exhaust inlets located throughout the buildings which are connected by ductwork to four parallel exhaust air plenums each of which is designed to exhaust approximately $793 \text{ m}^3/\text{min}$ (28,000 cfm). Each plenum contains a bank of HEPA filters, an automatic roll pre-filter, motorized isolation dampers at each end of the filter trains and backdraft dampers to

minimize backflow through an idle plenum. The system is designed to exhaust approximately $2379 \text{ m}^3/\text{min}$ (84,000 cfm) with three of the four exhaust plenums in service while the fourth plenum serves as a standby.

A portion of the total exhaust flow originates from spaces of the Auxiliary Building which contain CVCS equipment located between the Reactor Coolant System (RCS) and the letdown heat exchanger. This is the only equipment outside the Containment which processes hot, unpurified reactor coolant and therefore presents an increased airborne radioactivity potential. The $2.8 \text{ m}^3/\text{min}$ (100 cfm) ventilation exhaust from these spaces is drawn by a separate booster fan through a filtration unit consisting of a prefilter, a HEPA filter, a deep-bed charcoal filter and a final HEPA filter. The exhaust is then directed to normal building exhaust ducts.

A.4.3 HVAC-Turbine Building

The Turbine Building is served by the following heating, ventilating and cooling systems (Figures A.4-6 and A.4-7).

- 1) Turbine Building Ventilation System cools the main parts of the Turbine Building during normal plant operation.
- 2) Auxiliary feedwater pump, turbine and diesel drive, Room Cooling Systems cool the emergency feedwater pumps when they operate and provide combustion air for the diesel engine.
- 3) Lube oil room Ventilation System cools the lube oil room.
- 4) Systems to exhaust various miscellaneous spaces.
- 5) Infrared and unit heaters heat the Turbine Building.
- 6) Emergency generator rooms Cooling and Exhaust System cools the emergency generator rooms when the generators are operating and provides combustion air for the diesel engines.
- 7) Switchgear room Cooling and Exhaust System cools the switchgear.

The areas served by these Turbine Building systems are not considered to be radioactive because the reactor is of the pressurized water type which does not normally produce radioactive steam. The air flows are not monitored for radiation level, and there is no treatment of exhaust air.

A.5 LIQUID RADIOACTIVE WASTE SYSTEMS

The Liquid Waste Systems collect, process, monitor and discharge all plant liquid effluents that may contain radioactive nuclides.

The information contained in Figures A.5-1, -2, -3 and -6 have not been converted to metric units to avoid unnecessary confusion since these data are included for illustration only and are not essential to the study results.

A.5.1 Dirty Radioactive Waste Treatment System (DRW)

The DRW System collects, processes, and monitors liquids having low radioactivity and high particulate content. The design objectives for the DRW System are:

- 1) To provide a means of collecting the liquid effluent from the floor drains and sumps from the Containment, the Fuel Building, and the Auxiliary Building.
- 2) To provide sufficient storage capacity to contain the maximum anticipated liquid flow from the floor drains and sumps in the Containment, the Fuel Building, and the Auxiliary Building, and to provide a minimum holdup time for the decay of short-lived nuclides.
- 3) To remove radioactive particulates from the process streams by filtration.
- 4) To monitor the process streams prior to discharge or recycle to determine the necessity of further processing in the Clean Radioactive Waste System (CRW).

The sources of liquids for this system are floor drains from all structures housing equipment containing radioactive fluids and the drains from the sample sink and radioactive chemical laboratory. The liquid collected in this system is not of a reusable quality and, depending on the quantity and nature of contamination, will be either processed or discharged. The only method of radioactivity reduction provided by the system is the decay of radionuclides while in storage pending discharge. Waste volumes found to exceed allowable chemical or radiological limits for discharge are transferred to the CRW System for processing.

FIGURE A.5-1. Containment Building Sumps Flow Diagram

FIGURE A.5-2. Auxiliary Building Sumps Flow Diagram

FIGURE A.5-3. Dirty Waste Drain Tank Flow Diagram

The DRW System process flow paths are shown in flow diagrams in Figures A.5-1 through A.5-5. The system collects waste in various tanks, the physical characteristics of which are given in Table A.5-1. System pump characteristics are summarized in Table A.5-2.

The DRW System is designed to collect, store, and monitor potentially radioactive wastes having activities of 10^{-5} Ci/m³ or less. If a particular volume of waste is found unacceptably high in chemical or radioactivity content, provisions are made in the system to direct them to the CRW system for further processing. Input sources to this system are basically equipment drains and floor drains which are highly variable in volume.

Four types of sumps collect drainage in those areas that cannot be drained directly to the dirty waste drain tank. They are the reactor cavity sump, the Containment Building sumps, the Auxiliary Building sumps, and the Auxiliary Building passageway sump.

The reactor cavity sump, located below the incore neutron detector instrument tunnel at the 4.9 m level, collects only leakage, washdown water, and condensation in the lowest section of the Containment Building. The reactor cavity drain pump is then used to transfer the drains to Containment Building sump "B".

Containment Building sumps "A" and "B" are at elevation 13.7 m outside the missile barrier between the north-south axis of the Containment centerline and the reactor centerline. These sumps collect floor drains and Component Cooling Water System drainage within the Containment. Each sump has an associated pump used to transfer drainage to the dirty waste drain tank through one air-operated and one motor-operated Containment isolation valve and then through air-operated dirty waste drain tank inlet selector valves.

The Auxiliary Building sumps "A" and "B", at elevation 1.5 m, collect drainage within the Auxiliary Building. All floor and equipment drains from elevation 1.5 m go to either Auxiliary Building sump "A" or "B" if from rooms containing ESF equipment. The sumps are completely covered and are continuously purged by the vent collection header. One Auxiliary Building sump pump in each sump is used to transfer sump drainage to the dirty waste drain tank on service through air-operated inlet selector valves.

DIRTY WASTE FILTERS

FIGURE A.5-4. Dirty Waste Monitor Tank Flow Diagram

FIGURE A.5-5. Dirty Radioactive Waste Treatment System

TABLE A.5-1. Dirty Radioactive Waste Treatment System Tank Data

<u>Characteristics</u>	<u>Dirty Waste Drain Tank</u>	<u>Dirty Waste Monitor Tank</u>
Quantity	One	One
Type	Vertical	Vertical
Stored material	Water, 4.2 <pH< 9.0	Water, 6.2 <pH< 9.0
Design pressure, psig	Atmospheric	Atmospheric
Normal operating pressure, psig	Atmospheric	Atmospheric
Design temperature	355°K (180°F)	355°K (180°F)
Normal operating temperature	322°K (120°F)	289-311°K (60-100°F)
Volume	22.1 m ³ (780 ft ³)	25.5 m ³ (900 ft ³)
Capacity	19.7 m ³ (5200 gal)	23.8 m ³ (6300 gal)
Material of construction	Stainless steel T304	Stainless steel T304
Design code	API 650	API 650

TABLE A.5-2. Dirty Radioactive Waste Treatment System Pump Data

Characteristics	Auxiliary Building Sump Pump	Containment Sump Pump	Dirty Waste Drain Tank Pump	Dirty Waste Monitor Tank Pump	Reactor Cavity Drain Pump
Quantity	Two	Two	Two	Two	One
Type	Vertical, single stage, centrifugal	Vertical, single stage, centrifugal	Horizontal, single stage, centrifugal	Horizontal, single stage, centrifugal	Suction lift, air- operated diaphragm
Pumped fluid	Water w/boric acid and dirt	Water w/boric acid and dirt	Water @ 311°K (100°F) 4.2<pH<9.0	Water @ 300°K (80°F) 6.2<pH<9.0	Water w/boric acid and dirt
Design pressure, Mg/m ² (psig)	105.4 (150)	105.4 (150)	105.4 (150)	105.4 (150)	105.4 (150)
Design temperature, °K (°F)	333 (140)	333 (140)	322 (120)	311 (100)	333 (140)
Design flow rate, m ³ /min (gpm)	0.284 (75)	0.189 (50)	0.189 (50)	0.189 (50)	0.189 (50)
Design head, m (ft)	7.62 (25)	15.24 (50)	48.77 (160)	24.38 (80)	18.29 (60)
NPSH required at design flow rate, m (ft)	0.457 (1.5)	0.457 (1.5)	1.22 (4)	1.83 (6)	N/A
Maximum operating flow rate, m ³ /min (gpm)	0.416 (110)	0.416 (110)	0.257 (68)	0.212 (56)	0.378 (100)
Head at maximum operating flow rate, m (ft)	4.57 (15)	9.14 (30)	33.53 (110)	23.16 (76)	9.14 (30)
NPSH required at maximum operating flow rate, m (ft)			1.52 (5)	2.29 (7.5)	
Minimum available NPSH, m (ft)			6.40 (21)	6.40 (21)	
Shutoff head, m (ft)	9.75 (32)	16.76 (55)	54.86 (180)	28.04 (92)	88.39 (290)
Materials of construction:					
Casing	S.S. T316	S.S. T316	ASTM A108-69	ASTM A108-69	S.S. T316
Impeller	S.S. T316	S.S. T316	Stainless Steel A296-68 GRCF8M	Stainless Steel A296-68 GRCF8M	N/A
Shaft	S.S. T316	S.S. T316	ASTM A296-67	ASTM A296-67	N/A
Driver:					
Type	Induction	Induction	Induction	Induction	Air (125 psi)
Service factor	1.0	1.0	1.15	1.15	N/A
Nameplate rating, watts (hp)	745 (1.0)	2237 (3)	5593 (7.5)	2237 (3)	N/A
Voltage	460	460	460	460	N/A
RPM	1750	1750	3500	3500	N/A
Design Code			ASME Class III Section C	ASME Class III Section C	

The Auxiliary Building passageway sump, at elevation 1.5 m, is a completely covered sump with a vent to the vent collection header. The sump is drained by gravity flow to Auxiliary Building sump "A" through a normally shut manual valve. This valve is equipped with a reach rod to allow valve operation from the floor above. The valve is normally shut to isolate the passageway drain from Auxiliary Building sump "A".

The dirty waste drain tank, at elevation 1.5 m of the Auxiliary Building, collects drainage as shown in Figure A.5-3. The tank is divided into two sections each having a capacity of 9.84 m^3 .

The tank sections are connected by an unvalved overflow line which has a 1 m loop seal drain connection to the Auxiliary Building passageway sump. This provides tank overpressure protection while preventing the possible spread of contamination. Each tank section has a sparger with a service air connection supplying air. The sparger is used to thoroughly mix tank contents in order to ensure representative tank sampling and prevent setting of particulate matter. Two dirty waste drain tank pumps, also at elevation 1.5 m, take suction on the tank sections and deliver collected drainage to the dirty waste monitor tank through the dirty waste filters.

The dirty waste monitor tank, at elevation 13.7 m of the Auxiliary Building, receives input flow as shown in Figure A.5-4. The tank is divided into two sections, each with sufficient capacity to receive the contents of a single dirty waste drain tank section in order to provide process compatibility. When one side is filled, it is isolated for analysis and the other side placed in service.

All DRW System piping is fabricated from seamless ASTM A-312, type-304 stainless steel and is rated for 105 Mg/m^2 (150 psig) service. All system joints are welded except at flanged equipment connections.

A.5.2 Clean Radioactive Waste Treatment System (CRW)

The CRW System collects waste in various tanks, the physical characteristics of which are given in Table A.5-3. The CRW System process flow paths are shown on flow diagrams in Figures A.5-6, A.5-7 and A.5-8.

FIGURE A.5-6. Auxiliary Building Drain Tank Flow Diagram

FIGURE A.5-7. Treated Waste Monitor Tank Flow Diagram

FIGURE A.5-8. Clean Radioactive Waste Treatment System

TABLE A.5-3. Clean Radioactive Waste Treatment System Tank Data

<u>Characteristic</u>	<u>Reactor Coolant Drain Tank</u>	<u>Auxiliary Building Drain Tank</u>	<u>Clean Waste Receiver Tank</u>
Quantity	One	One	Two
Type	Horizontal	Vertical	Vertical
Stored material	Borated water, $4.2 < \text{pH} < 10.5$	Borated water, $4.2 < \text{pH} < 10.4$	Borated water, $4.2 < \text{pH} < 13$
Design pressure, Mg/m^2 (psig)	Internal - 35 (50) External - 49 (70)	35 (50)	0.7 (1)
Normal operating pressure, Mg/m^2 (psig)	1.4 (2)	1.4 (2)	Atmospheric
Design temperature, $^{\circ}\text{K}$ ($^{\circ}\text{F}$)	422 (300)	355 (180)	355 (180)
Normal operating temperature, $^{\circ}\text{K}$ ($^{\circ}\text{F}$)	283 - 366 (50 - 200)	289 - 322 (60 - 120)	289 - 322 (60 - 120)
Volume, m^3 (ft^3)	1.3 (46)	5 (175)	57 (2010)
Capacity, m^3 (gal)	1.3 (347)	5 (1300)	57 (15,322)
Material of construction	ASTM SA240 T304	ASTM SA240 T304	ASTM SA240 T304
Design code	ASME III-C	ASME VIII	API 620
<u>Characteristic</u>	<u>Chemical Waste Drain Tank</u>	<u>Treated Waste Monitor Tank</u>	<u>Waste Concentrate Holding Tank</u>
Quantity	One	Two	Two
Type	Vertical	Horizontal	Vertical
Stored material	Water w/NaOH, $6.2 < \text{pH} < 13$	Water, $6.2 < \text{pH} < 9.0$	12% Boric acid concentrate
Design pressure, Mg/m^2 (psig)	Atmospheric	0.7 (1)	10.5 (15)
Normal operating pressure, Mg/m^2 (psig)	Atmospheric	Atmospheric	Atmospheric
Design temperature, $^{\circ}\text{K}$ ($^{\circ}\text{F}$)	355 (180)	355 (180)	230
Normal operating temperature, $^{\circ}\text{K}$ ($^{\circ}\text{F}$)	289 - 322 (60 - 120)	322 (120)	289 - 366 (60 - 200)
Volume, m^3 (ft^3)	24.5 (860)	57 (2010)	5 (175)
Capacity, m^3 (gal)	24.5 (6500)	57 (15,322)	5 (1300)
Material of construction	ASTM SA240 T304	ASTM SA240 T304	ASTM SA240 T304
Design code	API 650	API 620	ASME VIII

The CRW System design capacity is based on a relatively small and steady rate of liquid waste generation during normal plant operation with peak waste generation occurring during shutdown and equipment maintenance periods. System drain tanks are sized to accept in excess of 3 days of normal operational drainage before processing is required. In addition, tanks are sized to accept wastes associated with equipment maintenance.

Liquids processed by the CRW System will contain varying amounts of boric acid and other chemicals. To minimize corrosion, all piping, valves, and major components in contact with the process fluid are stainless steel.

The Auxiliary Building drain tank, at elevation 1.5 m of the Auxiliary Building, collects drains by gravity flow from nuclear steam supply system components within the Auxiliary Building.

The chemical waste drain tank, at elevation 7.6 m of the Auxiliary Building, collects chemically contaminated drainage from the Steam Generator Blow-down System. The chemical waste drain tank receives normal operational and maintenance drainage volumes which are negligible when compared to tank capacity.

The reactor coolant drain tank, at elevation 12.8 m of the Containment inside the Containment recirculation sump, normally collects contaminated drainage that originates within the Containment.

The reactor coolant drain tank receives a normal operational design drainage of 0.034 m^3 (9 gal) an hour, based primarily on the volume of reactor coolant pump seal leakage. This input rate will require tank pump-down for processing approximately every 1-1/2 days.

The clean waste receiver tanks, at elevation 1.5 m of the Auxiliary Building, collect waste from the previously mentioned pump discharges as well as from drains with a higher probability of being contaminated with particulate matter of chemicals.

The CRW evaporator, at elevation 18.5 m of the Auxiliary Building, is designed to receive radioactive wastes from the collecting tanks previously mentioned and concentrate these wastes, producing distillate in the process

whose radioactive contamination levels have been reduced by a factor of at least 10^3 . The evaporator will process $0.057 \text{ m}^3/\text{min}$ (15 gpm) of radioactive waste to produce distillate and concentrated waste products consisting primarily of approximately 12 weight percent boric acid.

Contaminated wastes are pumped to the evaporator through the clean waste filter which removes particulate matter that could reduce evaporator efficiency.

Two waste concentrates holding tanks, at elevation 23.5 m of the Auxiliary Building, receive evaporator concentrates residue from both the CRW evaporator and the CVCS boric acid evaporators. System piping which processes waste concentrates is electrically heat traced and will maintain solution temperature above 347°K (165°F) to insure boric acid solubility up to 17 weight percent.

Two treated waste monitor tanks, at elevation 13.7 m of the Auxiliary Building, receive the evaporator distillate and store it for monitoring before further processing. The tanks can also receive Steam Generator Blowdown System effluents for monitoring.

The treated waste monitor tanks are each sized to receive the condensate resulting from evaporator processing of one clean waste receiver tank. Similarly, the waste concentrates holding tanks are each sized to receive the concentrates resulting from evaporator processing of one clean waste receiver tank.

The spent resin storage tank, at elevation 18.6 m of the Auxiliary Building, receives slurries of spent resin from plant demineralizers for storage prior to solidification processing for disposal.

All CRW System piping is fabricated from seamless ASTM A-312, type-304 stainless steel and is rated for 105 Mg/m^2 (150 psig) service. All system joints are welded except at flanged equipment connections.

A.6 RADIONUCLIDE CONTAINMENT SYSTEMS

The Containment System that encloses the reactor and other portions of the plant constitutes a design feature provided primarily for the protection of

public safety. The ability to perform its intended role, if called upon, of acting as a barrier to confine potential releases of radioactivity, depends upon maintaining leaktightness within specified bounds throughout its operating lifetime.

The Containment System is composed of the Containment Structure and the directly associated systems upon which the Containment function depends. These systems include the Containment Heat Removal Systems, Containment Air Purification and Cleanup Systems, Containment Isolation Systems, and Containment Combustible Gas Control Systems.

This section addresses itself only to the ventilation and exhaust systems concerned with dismantling.

The information contained in Figures A.6-1 and A.6-2 have not been converted to metric units to avoid unnecessary confusion since these data are included for illustration only and are not essential to the study results.

System Descriptions

The Purge Supply System (CS-1 in Figure A.6-1) and the Purge Exhaust and Refueling Cavity Supply and Exhaust System (CS-2 in Figure A.6-1) are designed to provide one and one-half Containment air changes of outside air per hour $1416 \text{ m}^3/\text{min}$ (50,000 cfm). The purge exhaust passes through HEPA filters to remove airborne contaminants.

The Purge Supply System (CS-1) consists of an outside air intake, a roll-type prefilter, a bank of HEPA filters, two $1416 \text{ m}^3/\text{min}$ (50,000 cfm) fans arranged in parallel, anti-backdraft dampers, a Containment penetration, inboard and outboard isolation valves, and ductwork. The fans and filters are located outside of Containment. At the penetration, the supply system is equipped with quick-closing purge isolation valves. The system is operated before and during personnel occupancy of the Containment.

The Purge Exhaust and Refueling Cavity Supply and Exhaust Systems (CS-2) consists of a roll-type prefilter, a bank of HEPA filters, two 1416 m^3 (50,000 cfm) fans arranged in parallel (100% redundancy), anti-backdraft dampers, a Containment penetration, inboard and outboard isolation valves, and ductwork. The fans and filters are located outside of the Containment. At the

penetration, the exhaust system is equipped with quick-closing purge isolation valves. The system exhausts to the Containment purge vent at the top of the Containment Building.

Exhaust air is monitored for radioactivity by high range and low range radio-gas monitors, a particulate monitor, and an iodine monitor. The high range gas monitor is a G-M tube type, while the low range gas, particulate, and iodine monitors are of the beta-gamma scintillation type. The Purge Exhaust System is operated before and during personnel occupancy of the Containment.

The refueling cavity supply consists of two fans which draw approximately $227 \text{ m}^3/\text{min}$ (8,000 cfm) of air from the Containment atmosphere and discharge it horizontally across the refueling cavity. The refueling cavity exhaust consists of two fans which draw approximately $425 \text{ m}^3/\text{min}$ (15,000 cfm) of air from inlets at the surface of the refueling cavity and discharge to the Purge Exhaust System (CS-2).

The ventilating equipment list is shown in Figure A.6-2.

FIGURE A 6-1 Air Flow Diagram
Containment Building

SEISMIC CLASS	SYSTEM NUMBER	EQUIPMENT NUMBER	AREA SERVED	LOCATION OF EQUIPMENT	FANS														COILS														FILTERS										CONF IG	INITIAL	PD					
					CHARACTERISTICS														COIL DATA														TEST EFFICIENCY																	
					CAPACITY CFM EACH	QV FPM	DIA	TYPE	DISCH	ROTN	ANG 1	FAN RPM	BRK HP	HP	VOLTS	PHASE	HZ	COIL NUMBER	TYPE	CFM	ENT TEMP	LV TEMP	RAV (F00)	ENT TEMP	LV TEMP	CFM	DIA	TUBES	NOMINAL SIZE	ROWS	FLAT	TYPE	ROL	HEPA	FACE VELOCITY FPM	FILTER EQUIPMENT NUMBER	TEST EFFICIENCY	DUST SPOT	WT METHOD	OSP										
1	CS 1	VC 201 A VC 201 B VC 201 C VC 201 D	CONTAINMENT AIR COOLERS SYSTEM	EL 200 0 AREAS A&B	75 000	4200		AXIAL FLOW	DOWN BLAST			4	510	70	125	400	3	60	VE 201 A VE 201 B VE 201 C VE 201 D	COMPONENT COOLING WATER	100 000	120°	80°	200 5	95°	101.4°	150	3/8	9	37766	CACHEM BIANI TOTAL	6																		
1		VC 201 A VC 201 B VC 201 C VC 201 D		EL 200 0 AREAS A&B															VE 201 A VE 201 B VE 201 C VE 201 D																															
11	CS 2	VC 201 A VC 201 B	REFUELING POOL SUPPLY FANS	EL 99 0 AREA 4	3 400	400	27	CENT AIR FLOW	TOP HORIZONTAL	CCW	8	640	0.42	94	400	3	60																																	
11	CS 8	VC 201 A VC 201 B VC 211 A VC 211 B	COOLANT PUMP VENT SYSTEM	EL 111 0 AREA 4 EL 79 0 AREA 4	62 000	1500	40	AXIAL FLOW	DOWN			4	1170	20	10	400	3	60																																
		VC 201 A VC 201 B VC 201 C VC 201 D		EL 100 0 AREA 5 EL 100 0 AREA 5				HORIZONTAL				1170	20	25																																				
		VC 201 A VC 201 B VC 201 C VC 201 D		EL 100 0 AREA 5 EL 100 0 AREA 5				HORIZONTAL				1170	4.5	20																																				
11	CS 2	VC 201 A VC 201 B	REFUELING POOL EXHAUST FANS	EL 87 3 AREA 4 EL 97 0 AREA 4	1 500	2000	79	AXIAL FLOW	UP BLAST HORIZONTAL			4	1750	4.5	3	400	3	60																																
11	CS 6	VC 201 A VC 201 B VC 201 C	REACTOR CAVITY COOLING SYSTEM	EL 40 0 AREA 5 EL 50 0 AREA 5	11 000	2700	37	AXIAL FLOW	HORIZONTAL			4	1750	7.3	10	400	3	60	VE 201 A VE 201 B	CHILLED WATER	13 000	100°	101.5°	170	90°	60°	3/8	1	10100	EACH FAN	8																			
11	CS 1	VC 210 A VC 210 B	PURGE SUPPLY SYSTEM	EL 97 0 AREA 5	50 000	3400	20	AXIAL FLOW	HORIZONTAL			4	1750	100	125	400	3	60																																
11	CS 2	VC 211 A VC 211 B	PURGE EXHAUST SYSTEM	EL 99 0 AREA 4	50 000	3400	20	AXIAL FLOW	HORIZONTAL			4	1750	100	125	400	3	60																																
11	CS 4	VC 212	INCORE INSIDE RM COOLING	EL 80 1 AREA 5	1 500		18	AXIAL FLOW	DOWN BLAST			4	1750	0.7	1	400	3	60	VE 201	CHILLED WATER	1 500	120°	80°	100 5	100°	60°	25	3/8	1	10170		8																		
11	CS 4	VC 213	PRESSURIZER COMPARTMENT COOLING SYSTEM	EL 130 6 AREA 4	8 150	2000	27	AXIAL FLOW	HORIZONTAL			4	1750	6	5	400	3	60	VE 201	CHILLED WATER	8 150	120°	70°	352 0	90°	60°	47	3/8	1	10170		4																		
11	CS 3	VC 201 A VC 201 B VC 201 C VC 201 D	CONTINUED DRIVE VENT SYSTEM	EL 100 0 AREA 5	30 000	4800	61 1/2	CENT BLOWER	UP BLAST	CCW	8	870	118	125	400	3	60																																	
11		VC 201 A VC 201 B	CLEAN UP REC. REGULATING UNITS																																															
1	CS 8	VC 201 A VC 201 B	HYDROGEN VENT SYSTEM	EL 97 0 AREA 4	170		12 1/2	CENT BLOWER				4	3500		1	400	3	60																																
1	CS 8	VC 201 A VC 201 B	HYDROGEN INLEAK SYSTEM	EL 200 0 AREAS A&B	2 300	1000	21	AXIAL FLOW	DOWN BLAST			4	1750	0.15	1 1/2	400	3	60																																

FIGURE A.6-2. Equipment List:
Containment Ventilation and Cooling

A.7 FACILITY DESCRIPTION GLOSSARY

This glossary is divided into two parts. The first part contains abbreviations directly related to PWR equipment, systems, and related terminology, including acronyms of technical societies associated with facility construction. The second part contains abbreviations common to the reference PWR facility drawings found in Appendix A.

A.7.1 FACILITY ABBREVIATIONS

<u>Abbreviation</u>	<u>Name</u>
AEC	Atomic Energy Commission
AFS	Auxiliary Feedwater System
AGA	Automatic gas analyzer
AIEE	American Institute of Electrical Engineers
ASME	American Society of Mechanical Engineers
ASME B&PV Code	American Society of Mechanical Engineers Boiler and Pressure Vessel Code
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASTM	American Society for Testing Materials
AWS	American Welding Society
BCL	Beginning of Core Life
BCWS	Bearing Cooling Water System
BECHTEL	Bechtel Corporation (San Francisco)
BRS	Boron Recycle System
CACS	Containment Air Cooler System
CCMS	Containment Condensate Measuring System
CCWS	Component Cooling Water System
CFS	Condensate and Feedwater System
CHDS	Containment Humidity Detection System
CIS	Chemical Injection System or Containment Isolation Signal
CMS	Containment Monitoring System

<u>Abbreviation</u>	<u>Name</u>
CPES	Containment Purge and Exhaust System
CRDM	Control Rod Drive Mechanism
CRWS	Clean Radioactive Waste Treatment System
CSAS	Containment Spray Actuation Signal
CSS	Containment Spray System
CSTS	Condensate Storage and Transfer System
CWRT	Clean Waste Receiver Tank
CWS	Chilled Water System
CVCS	Chemical and Volume Control System
DFOS	Diesel Fuel Oil System
DGS	Diesel Generator System
DPT	Dye Penetrant Test
DRWS	Dirty Radioactive Waste Treatment System
DWS	Domestic Water System
ECCS	Emergency Core Cooling System
ECL	End of Core Life
EFPY	Equivalent full power years
ESF	Engineered Safety Features
EVS	Emergency Ventilation System
EWEB	Eugene Water and Electric Board
FHS	Fuel Handling System
FS	Feedwater System
FTS	Fuel Transfer System
FWS	Fire Water System
HEPA	High Efficiency Particulate Airfilter
HMS	Hydrogen Mixing System
HX	Heat Exchanger
IEEE	Institute of Electrical and Electronic Engineers
LOCA	Loss of Coolant Accident

<u>Abbreviation</u>	<u>Name</u>
MCC	Motor control center
MSS	Materials Standards Society or Main Stream System
MT	Magnetic Particle Test
MWTS	Makeup Water Treatment System
NDT	Nondestructive Testing
NEMA	National Electrical Manufacturers Association
NSSS	Nuclear Steam Supply System
NUS	NUS Corporation
OBE	Operating Basis Earthquake
PGE	Portland General Electric Company
PMWP	Primary makeup water pumps
PP&L	Pacific Power & Light Company
PSAR	Preliminary Safety Analysis Report
PSS	Primary Sampling System or Process Sampling System
QA	Quality Assurance
QAP	Quality Assurance Procedure or Quality Assurance Program
QC	Quality Control
RCCA	Rod Cluster Control Assembly
RCS	Reactivity Control System or Reactor Coolant System
RESAR	Reference Safety Analysis Report (<u>W</u>)
RGWS	Radioactive Gaseous Waste System
RHRS	Residual Heat Removal System
RT _{NDT}	Radiographic, Nil Ductility Transition Temperature
RWST	Refueling Water Storage Tank

<u>Abbreviation</u>	<u>Name</u>
SC I or II	Seismic Category I or II
SFP	Spent fuel (storage) pool
SG	Safety Guide (AEC)
SIS	Safety injection signal
SNUPPS	Standardized Nuclear Unit Power Plant System
SSE	Safe shutdown earthquake
SSSS	Solid State Shutdown System
SWS	Service Water System
TCWS	Turbine Building Cooling Water System
TNP	Trojan Nuclear Plant
UT	Ultrasonic Testing
<u>W</u>	Westinghouse Electric Corporation

A.7.2 FACILITY DRAWING ABBREVIATIONS

<u>Abbreviation</u>	<u>Description</u>
A	Amber, ammeter or ampere
AC	Alternating current
ACB	Air circuit breaker
AIR COND	Air conditioning unit
AL	Alarm or aluminum
AMP	Amplifier or ampere
ANALY	Analyzer
ANNUN	Annunciator
APPROX	Approximate
AR	Alarm relay
ARM	Armature
ARR	Arrester or arrange
ASSY	Assembly
A/T	Ampere transducer
ATMOS	Atmosphere
AUTO	Automatic
AUTO TRANS	Auto transformer

<u>Abbreviation</u>	<u>Description</u>
AUX	Auxiliary
AVG	Averaging or average
AWG	American Wire Gauge
BAR	Barrier
BAT	Battery
BCT	Bushing current transformer
BD	Board
BIL	Basic impulse insulation level
BKR	Breaker
BLDG	Building
BLO	Blower
BLR	Boiler
B/M	Bill of material
BRG	Bearing
BSTR	Booster
BYP	Bypass
CAB	Cabinet
CAP	Capacitor or capacity
CAS	Casing
CAUS	Caustic
CAV	Cavity
CB	Control board
CC	Control cabinet or closing coil
CCW	Component Cooling Water
CE	Conductivity Cell
CENTRIF	Centrifugal
CH	Control House (Switchyard)
CHEM	Chemical
CHG	Change or charger, charging
CHLOR	Chlorinator or chlorine
CIRC	Circulating

<u>Abbreviation</u>	<u>Description</u>
CKT	Circuit
CL	Center line
CNCT	Concentrate
CND	Conduit
CNDS	Condensate
COL	Column
COMM	Communication or common
COMP	Computer or component
COMPR	Compressor
CON	Console
COND	Condenser or conductor
CONT	Control
CONTAIN	Containment
CONTC	Contractor
CONTD	Continued
COOL	Cooling or cooler
CP	Cable pit or condensate pump
CR	Control Room
CS	Control switch or control station
CTR	Center
CUB	Cubicle
CVCS	Chemical Volume Control System
CW	Cooling water, circulating water or clockwise
DC	Direct Current
DEMIN	Demineralizer
DET	Detector
DIA	Diameter
DIAG	Diagram
DIFF	Differential
DISC	Disconnect
DISCH	Discharge

<u>Abbreviation</u>	<u>Description</u>
DIST	Distribution
DP	Differential pressure
DR	Drive
DRN	Drain
DWG	Drawing
EA	Exhaust Air
EL	Elevation
ELEC	Electric or electrical
EMER	Emergency
EMT	Electrical metallic tubing
ENCL	Enclosure
ENG	Engineer
E/P	Electro to pneumatic transducer
EQUIP	Equipment
ESF	Engineered Safety Features
EXC	Exciter or excitation
EXH	Exhaust
EXHTR	Exhauster
EXT	Extractor
EXTN	Extraction
F	Flow
FDR	Feeder
FIG	Figure
FIL	Filter
FL	Flame
FLD	Field
FLEX	Flexible
FS	Flow switch
F/T	Frequency transducer
FW	Feedwater
G	Green
GEN	Generator

<u>Abbreviation</u>	<u>Description</u>
GOV	Governor
GR	Gear
GRD	Ground
GS	Gland seal
GSS	Gland seal steam
GUI	Guide
H ₂	Hydrogen
H&V	Heating and ventilation unit
HI	High
HP	High pressure or horsepower
HSE	House
HT	Heat tracing
HTG	Heating
HTR	Heater
HV	High voltage
HVAC	Heating, ventilation, air conditioning
HWC	Hot water circulation
HYD	Hydraulic
HYDZ	Hydrazine
ID	Inside diameter
IN	Inlet, input or intake
INC	Incoming
IND	Indicator
INFIL	Infiltrate
INST	Instrument
INV	Inverter
IP	Intermediate pressure
ISO	Isolation or isolated
ITL	Interlock
JB	Junction box
KV	Kilovolt
KVA	Kilovolt-ampere
KW	Kilowatt

<u>Abbreviation</u>	<u>Description</u>
L	Left or level
LA	Lightning arrester
LAB	Laboratory
LAL	Level alarm, low
LC	Load Center
LCS	Local control station
LIM	Limit
LIQ	Liquid
LO	Lubrication oil, low or lockout
LP	Low pressure
LS	Level switch or limit switch
LTG	Lighting
M	Main
MACH	Machine
MAN	Manual
MCC	Motor Control Center
MECH	Mechanic or mechanical
MFR	Manufacture or manufacturer
MG or M-G	Motor-generator
MH	Manhole
MISC	Miscellaneous
MO	Motor operated
MOV	Motor operated vane or valve
MS	Main steam
MT	Main Transformer
MTR	Motor or meter
NC	Normally closed
NEC	National Electrical Code
NEG	Negative
NES	Non-engineered Safety Features
NEUT	Neutral

<u>Abbreviation</u>	<u>Description</u>
NO	Number or normally open
NORM	Normal
NP	Name plate
NPS	Negative phase sequence
OA	Outside Air
O ₂	Oxygen
OC	Overcurrent
OCB	Oil circuit breaker
OD	Outside diameter
OL	Overload
OUT	Outlet or output
P	Pressure
PAL	Pressure alarm
PB	Pull box or pushbutton
PH	Phase
PMG	Permanent magnet generator
PNL	Panel
PO	Position or positioner
POS	Position switch or positive
POT	Potential
PP	Pump
PRESS	Pressure
PRZR	Pressurizer
PS	Pressure switch
PT	Potential transformer or pressure transmitter
PW	Pilot wire
PWR	Power
R	Right or red
RAB	Reactor Auxiliary Building
RAD	Radiation
RCP	Reactor Coolant Pump
RE	Roof exhauster

<u>Abbreviation</u>	<u>Description</u>
REAC	Reactor
REC	Receiver
RECIP	Reciprocator or Reciprocating
RECIRC	Recirculation or recirculating
RECP	Receptacle
RECT	Rectifier
REG	Regulator
REF	Reference
RH	Reheat
RHEO	Rheostat
RHR	Residual Heat Removal
RHTR	Reheater
REINF	Reinforced
RLY	Relay
RM	Room
RR	Railroad
RS	Reversing starter
RTD	Resistance temperature detector
SAMP	Sampling
SCAV	Scavenging
SCHEM	Schematic
SCRN	Screen
SD	Steam load drain or dump valve
SEC	Secondary
SECT	Section
SEL	Selector
SEP	Separator
SERV	Service
SEW	Sewage
SFT	Shaft
SHED	Shedding

<u>Abbreviation</u>	<u>Description</u>
SHLD	Shield
SI	Seal in relay or contact
SIG	Signal
SIS	Safety injection signal
SL	Seal
SO	Seal oil
SP	Speed or spare
SPA	Spaced
S.S.	Stainless steel
STA	Station
STBY	Standby
STD	Standard
STG	Storage
STM	Steam
STR	Starter
STUP	Startup
SUBSTA	Substation
SUCT	Suction
SUPHTR	Superheater
SUPT	Support
SV	Solenoid valve
SW	Switch or service water
SWBD	Switchboard
SWGR	Switchgear
SYN	Synchronous, synchronizing or synchronism
SWYD	Switchyard (i.e., switching station)
SYS	System
T	Temperature, transducer or transmitter
TACH	Tachometer
TAL	Temperature alarm
TB	Terminal box

<u>Abbreviation</u>	<u>Description</u>
TC	Thermocouple or trip coil
TD	Time delay
TDC	Time delay (time delay to close)
TDD	Time delay (on deenergizing)
TDE	Time delay (on energizing)
TDO	Time delay (time delay to open)
TE	Thermal element
TEL	Telephone
TERM	Terminal
TG or T-G	Turning gear or turbine generator
THERMO	Thermostat
THROT	Throttle
TK	Tank
TOS	Torque switch
TRANS	Transfer or transformer
TRAV	Traveling
TS	Temperature switch or test switch
TURB	Turbine
TV	Television (receiver)
TVC	Television camera (transmitter)
TYP	Typical
UF	Underfrequency
UH	Unit Heater
UP	Upstream or upper
UV	Undervoltage
V	Volt, voltage, voltmeter or valve
VAC	Vacuum
VAP	Vapor
VB	Vibrometer
VENT	Ventilation
VERT	Vertical
V/T	Volt transducer

<u>Abbreviation</u>	<u>Description</u>
W	West, white or wattmeter
WHSE	Warehouse
WP	Weatherproof
W/T	Watt transducer
WTR	Water

REFERENCES

1. TROJAN, Final Safety Analysis Report, Portland General Electric Co., Portland, OR, September 1973.
2. RESAR-3 Preliminary Safety Analysis Report, Westinghouse Electric Co., Pittsburgh, PA, November 1973.
3. Standardized Nuclear Unit Power Plant System - SNUPPS, Preliminary Safety Analysis Report, Volume 1, November 1974.
4. TROJAN Nuclear Plant Information Center Brochure, Portland General Electric Co., Rainier, OR.
5. TROJAN, Final Safety Analysis Report, Volume 6, Amendment 3, Portland General Electric Co., Portland, OR, September 1973.
6. TROJAN, Final Safety Analysis Report, Volume 3, Amendment 3, Portland General Electric Co., Portland, OR, September 1973.

APPENDIX B

REFERENCE PWR SITE DESCRIPTION

A reference environment was developed to aid in assessing the public safety and potential environmental effects of conceptually decommissioning a PWR by various alternative methods. Some of the information, namely the meteorology parameters and population distributions, were taken from Appendix I of the ALAP Study⁽¹⁾ for the river site expected for the year 2000.

The ecological information was derived from the environment of an operating nuclear reactor.⁽²⁾ The remainder of the information was obtained from a variety of sources or it was developed specifically for this study. All of the information is representative of potential sites for PWR facilities in the midwestern or south mideastern United States.

The animal species identified in the environs of this reference site may not be the same as those identified in site specific statements, however, the impact on species including those identified as endangered is not considered to be so markedly different as to obviate the usefulness of this analysis.

Individual features of an actual site for a given nuclear facility may vary from those of the reference site. However, it is believed that this generic approach will result in a more meaningful overall analysis of potential impacts associated with most nuclear reactor facilities. Site specific assessments would be required for individual PWR power stations.

⁽¹⁾ U.S. AEC, Final Environmental Statement Concerning Proposed Rule-making Action: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criteria "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, WASH-1258, Directorate of Regulatory Standards, Volume 1 of 3, Figure 6B-1, p. 6B-43 and Figure 6C-8, p. 6C-12, July 1973.

⁽²⁾ U.S. AEC, Final Environmental Statement Related to Operation of Monticello Nuclear Generating Plant, Docket No. 50-263, pp. II-15 through II-26, November 1972.

Naturally occurring radionuclides and radionuclides expected from nuclear weapons testing will be present on any nuclear facility site consistent with the level generally found in the local region. A short discussion on the magnitude of these two sources of radiation is presented in Volume 1, Section 7 for perspective and is not repeated here. The information presented in Section 7 is a distillation of a detailed treatise on the subject found in the UNSCEAR report.⁽³⁾ Such information on the levels and nature of the radioactive contamination present on the site at the time of decommissioning is necessary, however, in order to make decisions about alternative uses for a site. Estimates of the PWR residual radioactivity are presented in Appendix C. Expected site radioactivity contamination levels at the time of plant shutdown are presented Appendix E.

B.1 SITE LOCATION

The reference site is located in a rural area with characteristics similar to those found in midwestern or south mideastern United States. The site occupies 4.7 km² (1160 acres) in a rectangular shape of 2 km (1.24 miles) by 2.35 km (1.46 miles). A moderately-sized river runs through one corner of the site.

B.2 DEMOGRAPHY

The site is located in a rural area that has relatively low population density, with highest population densities occurring at distances of 16 to 64 km (10 to 40 miles) away, and gradually reducing population densities out to 177 km (110 miles). The population numbers are given in Table B.2-1. The total population out to 80 km (50 miles) is 3.52 million.

⁽³⁾United Nations Scientific Committee on the Effects of Atomic Radiation, Ionizing Radiation: Levels and Effects, Volume 1, United Nations, pp. 29-63, 1972.

TABLE B.2-1. Population Distribution Around the Reference Site

Distance from Facilities (miles) (kilometers)		Population Density (persons/mi ²)	Total Population in Annulus(a)	Cumulative Population
0 - 1	0 - 1.6	---	10	10
1 - 2	1.6 - 3.2	226	2,130 ^(b)	2,140 ^(b)
2 - 3	3.2 - 4.8	333	5,230	7,370
3 - 4	4.8 - 6.4	361	7,940	15,300
4 - 5	6.4 - 8.0	415	11,700	27,000
5 - 10	8.0 - 16	379	89,300	116,000
10 - 20	16 - 32	398	375,000	491,000
20 - 30	32 - 48	559	878,000	1,370,000
30 - 40	48 - 64	469	1,030,000	2,400,000
40 - 50	64 - 80	398	1,120,000	3,520,000

(a) It is assumed that 1/16 of these persons reside within each of the 16 sectors.

(b) Totals are rounded to three significant figures.

B.3 LAND USE

The use of any part of the total site area of 4.7 km² (1160 acres) for anything other than for the reactor facility is prohibited. The plant facilities are located inside a much smaller fenced-in portion (0.12 km², or 30 acres) of the site, with a minimum distance of one kilometer (0.6 miles) from the point of plant airborne releases to the outer site boundary (fenced and marked).

In most of the surrounding area about 80% of the land is used for farming. The main crops in this area, which include all land within 16 km (10 miles) of the site, are soybeans (60%), corn, oats, and other grain (30%) and hay (10%). It is expected that this area will remain largely agricultural and that the population will not change significantly.

A wildlife refuge is located about 14 to 20 km (9 to 12 miles) from the site. A State Park is located about 10 km (6 miles) in the opposite direction, and a State Forest and campground are about 14 km (9 miles) away, near the wildlife refuge.

There are large truck gardens in the area. The nearest dwelling is a farm house located 1.3 km (0.8 miles) from the site. A milk cow is kept at this farm and is maintained on fresh pasture 7 months of the year. A family garden is kept for vegetables with a growing season of 5 months. River water is used for irrigation of this farm.

B.4 WATER USE

Complete access to natural waterways (except to onsite shorelines) by the public is assumed. The river is used for dispersal of acceptable liquid effluents from the reactor facility. The river is used for irrigation, fishing, boating and other aquatic recreational activities. Its shores are used for hunting and for marinas and docks. It is assumed that one-sixth of the average person's diet comes from food crops irrigated by the river water downstream of the site, and one-third of the maximum exposed individual's food intake is from the same source.

In the vicinity of the plant it is assumed that a limited number of individuals may spend 50 hr/yr swimming, 50 hr/yr boating, and may spend 70 hr/yr fishing on the shoreline obtaining 20 kg of fish. Essentially no fresh water molluscs or crustacea are taken from the river. Aquatic foods are assumed to be consumed within 24 hours of the time they are caught.

Drinking water is taken from the river by larger towns and cities. Individuals living immediately downstream near the river derive their potable water from wells and not directly from the river. For development of population radiation doses it is assumed that 20% of the residents within 80 km (50 miles) of the site obtain their drinking water from the reference river. Travel time from the site to the urban water consumer is taken to be 48 hours. It is assumed that each person not in the vicinity of the plant will spend on the average 2 hr/yr swimming, 4 hr/yr boating, and 4 hr/yr on the shoreline

downstream of the site.⁽⁴⁾ The average per capita fish consumption for this area has been estimated to be 6.9 kg/yr.⁽⁵⁾ It is assumed that 15% of this consumption is from fish obtained downstream from the site.

The region has no large commercial fishing operations. Most of the fish catch is caught by local recreational anglers for home consumption.

B.5 GEOLOGY

The reference site occupies a low bluff that forms a bank of the river running through one corner. Several flat alluvial terraces comprise the main topographic features on the property. These terraces lie at average elevations of 280 to 284 m (918 to 932 feet) above sea level and, in general, slope away from the river at grades of 2 to 3%. The topography at the site is typical of that in the area and the region.

The rocks that underlie this region, including the reference site, are relatively old. Glaciation probably less than 1,000,000 years ago as well as recent alluvial desposition, has mantled the older rocks with a variety of unconsolidated materials in the form of glacial moraines, glacial outwash plains, glacial till and river bed sediments. This cover of young soils rests upon a surface of glacially carved bedrock consisting of sandstone, shale and granitic rocks. The upper surface of underlying rock can support unit foundation loads up to 73,200 kg/m² (15,000 lb/ft²). The bedrock surface is irregular and slopes generally to the east or southeast.

The nearest known or inferred fault is 37 km (23 miles) southeast of the site. There is no indication that faulting has affected the area of the site in the last few million years. Within the last 110 years, only two earthquakes were recorded as having occurred within 161 km (100 miles) of the site. The first occurred in 1917 and had an intensity of 6 on the Modified

(4) U.S. AEC, Final Environmental Statement Concerning Proposed Rule-Making Action: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criteria "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, WASH-1258, Directorate of Regulatory Standards, Volume 1 of 3, Table 6B-6, p. 6B-36, July 1973.

(5) U.S. NRC, Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50," Appendix I, Table E-5, pp. 1.109-40, October 1977.

Mercalli scale. The epicenter was located about 97 km (60 miles) from the site. The second occurred in 1950 with an estimated intensity of 5 to 6, and the epicenter was located about 130 km (80 miles) from the site. For construction of facilities in this area, the Design Basis Earthquake relates to a horizontal acceleration of 0.25 g.

B.6 HYDROLOGY

The site lies on the outwash plain of the river running across its corner. This plain is bounded by a glacial moraine containing numerous lakes and swampy areas. Drainage from the moraine converges with drainage from the terraces and swales, and flows generally southeast into the river. The river tributaries close to the site are a creek, 8 km (5 miles) northwest, and another creek, 4.8 km (3 miles) southeast. Another river flows approximately parallel to the plant site river along a line 6.4 km (4 miles) north of the site, joining the plant-site river 24 km (15 miles) downstream from the site.

The natural surface drainage of the immediate site area is mainly to the southwest, away from the river. Surface runoff tends to collect in the depression at the south end of the terrace, where it is bounded by higher ground.

Large supplies of groundwater are available from the plant site river outwash plain alluvium, glacial moraine, and from underlying sandstones in the area. The general course of deep groundwater flow is to the southeast. The regional gradient broadly parallels the trend of the topography and the surface drainage.

The groundwater levels near the site are relatively flat and slope toward the river during normal river stages. During periods of high river flow, there may be some reversal of groundwater flow. These reversals would be of short duration and infiltration of water from the river would be limited. The gradient toward the river is reestablished after the high water recedes.

River flow information based on 40 years of data from the plant site river gaging station shows an annual average flow rate of $1420 \text{ m}^3/\text{sec}$ (50,000 cfs). River flow and temperature data pertinent to the reference site are shown in Figures B.6-1 and B.6-2, respectively.

The river is assumed to be reasonably free of pollution as shown in Table B.6-1. It is moderately turbulent during the spring. Little or no stratification of water occurs. There is no tidal influence.

The average river velocity at the site varies between 0.46 and 0.76 m/sec (1.5 and 2.5 fps) for flows below $2000 \text{ m}^3/\text{sec}$ (72,000 cfs). The river drops about 3 m (10 ft) from 2.4 km (1.5 miles) upstream to 2.4 km (1.5 miles) upstream to 2.4 km (1.5 miles) downstream of the site. Rapids frequently exist in this stretch of the river.

The 1-in-1000-years flood would be expected to reach 280 m (921 ft) MSL (mean sea level), and the maximum flow of record (1965) was estimated to have reached 279 m (916 ft) MSL. Normal river stage at the site is about 275 m (905 ft) MSL, and the site grade is 287 m (942 ft) MSL.

FIGURE B.6-1. Recorded Daily Average and Extreme River Flows at the Reference Site

FIGURE B.6-2. Daily Average and Extreme Water Temperatures at the Reference Site

A study was conducted to determine the predicted flood discharge flow and water level at the site resulting from the "maximum probable flood" as defined by the U.S. Army Corps of Engineers. The "maximum probable flood" was estimated to reach a peak stage elevation of 286 m (939 ft) MSL at the reference site. The peak level at the site would be reached in about 12 days from the onset of the worst combination of conditions resulting in the "maximum probable flood."

The nearest domestic water supply reservoir is fed by the river from an intake about 60 km (37 miles) downstream. Under emergency conditions, withdrawal of river water could be suspended for about 48 hours without curtailment of nonessential use. With curtailment of nonessential use, this period could be extended to about 100 hrs.

The groundwater table under normal conditions is higher than the river; thus, groundwater and runoff drain to the river. There are numerous shallow

TABLE B.6-1. River Water Chemistry at the Reference Site (Summary of 12 Monthly Samples)

	<u>Minimum</u>	<u>Maximum</u>	<u>Average</u>	<u>Standard Deviation</u>	<u>Number of Samples</u>
<u>Solids (mg/l)</u>					
Total	143	216	185	23.2	1212
Dissolved	125	208	178	27.8	12
Suspended	1.2	18.4	7.5	6.2	12
<u>Hardness (mg/l)</u>					
(as CaCO ₃)					
Total	98	174	147	24.8	12
Calcium	70	120	99	15.6	12
Magnesium	28	58	48	9.9	12
<u>Alkalinity (mg/l)</u>					
(as CaCO ₃)					
Total	91	165	140	24.3	12
Phenolphthalein	0	12	1.8	4.1	12
<u>Gases (mg/l)</u>					
Free Carbon Dioxide	--	--	--	--	--
Ammonia-Nitrogen (N)	0	0.09	0.02	0.03	12
<u>Anions (mg/l)</u>					
Carbonate (CO ₃ ⁻²)	0	14.4	2.10	4.96	12
Bicarbonate (HCO ₃ ⁻)	111	201	166	29.1	12
Hydroxide (OH ⁻)	--	--	--	--	--
Chloride (CL ⁻)	0.30	5	1.43	1.48	12
Nitrate-Nitrogen (NO ₃ ⁻)	0.07	0.55	0.26	0.15	12
Sulfate (SO ₄ ⁻²)	6.3	13.5	9.5	2.2	12
Phosphorus-Soluble (PO ₄ ⁻³)	0.012	0.057	0.030	0.012	12
Silicate (SiO ₃ ⁻²)	3.2	12.5	7.7	3.3	12
<u>Cations (mg/l)</u>					
Calcium (Ca ⁺²)	28	48.1	39.7	6.28	12
Magnesium (Mg ⁺²)	6.8	14.1	11.6	2.4	12
Sodium (Na ⁺²)	2.8	6.4	5.0	1.1	12
Total Iron (Fe ⁺²)	0.04	0.52	0.23	0.13	12
Total Manganese (Mn ⁺²)	--	--	--	--	--
Potassium (K ⁺¹)	--	--	--	--	--
<u>Miscellaneous</u>					
Color (APHA units)	20	80	39	22	12
Turbidity (JTU)	1	4.5	2.53	1.48	12
Ryznar Index (AT 25 C)	6.64	7.86	7.21	0.377	12
Conductivity (mmhos)	192	350	292	49.8	12
pH	7.4	8.6	8.15	0.308	12
BOD (mg/l)	0.9	2.5	14	0.58	12
Dissolved Oxygen (mg/l)	8	15	10.6	2.1	11
Temperature (°C)	0.5	30	9.69	9.03	12

public wells, 23 to 46 m (75 to 150 ft) deep, supplying residences and farms along the river terrace. The closest public water supply well is a small city well obtaining water 72 m (237 ft) below ground level, about 4.8 km (3 miles) downstream.

The amount and rate of dispersion of effluent materials discharged into a river depend on the flow rate, geometry, bottom roughness, and density gradients. As a result, no single dilution-distance relationship exists that will account for all the possible variations.

The point-source dilution curve initially is inversely proportional to distance and then, after vertical mixing has occurred, the dilution is inversely proportional to the square root of the distance until the limiting effect of the near-side boundary becomes important at about 1,000 m (0.6 miles). The fully mixed situation occurs at a distance of about 50,000 m (30 miles).

Values indicated in Figure B.6-3 are assumed for evaluations of the reference river site in this study, using the case for one nuclear reactor. Adjustments can be made directly to these graphs for other liquid effluent flow rates.

FIGURE B.6-3. Liquid Effluents Dilution for the Reference River

The estimates to be developed for the concentration of radionuclides in the water downstream of the liquid effluent discharge point in a river are based on the analytical model given in WASH-1258.⁽¹⁾

B.7 METEOROLOGY

The site has a typical continental climate. It is characterized by wide variations in temperature, modest winter precipitation, normally ample spring and summer rainfall, and a general tendency to extremes in all climatic features. January is the coldest month and July is the warmest month. Table B.7-1 shows monthly temperature statistics.

TABLE B.7-1. Monthly Air Temperature Statistics (°C) at the Reference Site

	<u>Jan</u>	<u>Feb</u>	<u>Mar</u>	<u>Apr</u>	<u>May</u>	<u>Jun</u>	<u>Jul</u>	<u>Aug</u>	<u>Sep</u>	<u>Oct</u>	<u>Nov</u>	<u>Dec</u>
Maximum	-6.1	-4.4	3.3	12.8	20	25	28.3	26.7	22.2	15	4.4	-3.3
Minimum	-16.1	-14.4	-6.7	1.7	7.8	13.3	16.1	15	10	3.9	-4.4	-12.2
Mean	-11.1	-9.4	-1.7	7.2	13.9	18.9	22.2	21.1	16.1	9.4	0.0	-7.8
Extreme Maximum	15	16.1	27.8	32.8	40.6	39.4	41.7	40	40.6	32.2	23.9	17.2
Extreme Minimum	-38.9	-36.7	-34.4	-15.6	-6.7	0.6	5.6	3.3	-5.6	-13.3	-27.8	-33.9

The number of days with maximum temperatures of 32°C (90°F) and above is estimated to be 12. The number of days with a minimum temperature of 0°C (32°F) or below and -18°C (0°F) or below is estimated to be 168 and 40, respectively. The January relative humidities at 7:00 a.m., 1:00 p.m., and 7:00 p.m. are estimated to be 76, 68, and 70%, respectively. The corresponding humidities for July are 86, 55, and 55%. Monthly average humidities are shown in Table B.7-2.

TABLE B.7-2. Mean Monthly Relative Humidity (%) at the Reference Site

<u>Jan</u>	<u>Feb</u>	<u>Mar</u>	<u>Apr</u>	<u>May</u>	<u>Jun</u>	<u>Jul</u>	<u>Aug</u>	<u>Sep</u>	<u>Oct</u>	<u>Nov</u>	<u>Dec</u>
74	75	73	66	62	66	68	70	70	66	73	78

⁽¹⁾ U.S. AEC, Final Environmental Statement Concerning Proposed Rule-making Action: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criteria "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, WASH-1258, Directorate of Regulatory Standards, Volume 1 of 3, Figure 6B-1, p. 6B-43 and Figure 6C-8, p. 6C-12, July 1973.

The months of May through September have the greatest amounts of precipitation, with an average rainfall during this period of 432 to 457 mm (17 to 18 in.), 70% of the 610-mm (24-in.) annual rainfall. The maximum 24-hr total rainfall for the period 1894 through 1965 was 127 mm (5 in.) and occurred in May. Thunderstorms have an annual frequency of 36 and are the chief source of rain from May through September. Snowfall in the area averages 1070 mm (42 in.) annually, with occurrences recorded in all months except June, July and August. The extremes in annual snowfall of record are 152 mm (6 in.) minimum and 2235 mm (88 in.) maximum.

Annually, the winds are predominantly bimodal. This bimodal distribution is characteristic of the seasonal wind distributions as well. The average wind speed for spring is 11 km/hr (7 mph) and for the other seasons is about 16 km/hr (10 mph). The maximum reported wind speed of 148 km/hr (92 mph) was associated with a tornado. Tornadoes and other severe storms occur occasionally. Eight tornadoes were reported from 1916 through 1967 in the county where the site is located. The probability of a tornado striking a given point in this area is about 5×10^{-4} per year. For design purposes a wind velocity of 480 km/hr (300 mph) is assumed to be associated with tornadoes.

It is estimated that natural fog restricting visibility to 0.4 km (0.25 mile) or less occurs about 30 hr/yr. Icing due to freezing rain can occur between October and April, with an average of one to two storms per year. The mean duration of icing on utility lines is 36 hr.

Diffusion climatology comparisons with other locations indicate that the site is typical of the region, with relatively favorable atmospheric dilution conditions prevailing. Frequency of thermal inversion is expected to be about 32% of the year, and the frequency of thermal stabilities is 19% slightly stable, 27% stable, 20% neutral, and 34% unstable.

Data from a number of river sites used for nuclear reactors were used to calculate the "typical" annual atmospheric dispersion pattern in an average 22-1/2-degree sector from the site. Dispersion factors at selected distances for the average sector were determined from joint-frequency distributions for each site. This was done by calculating the dispersion factor, $\bar{X}/Q'$ for each sector at selected downwind distances and then calculating the average dispersion factor at each distance. In other words, the dispersion factors in

those sectors corresponding to overland trajectories were added without regard to direction and divided by the number of sectors involved. For river sites all sixteen sectors were used. Thus, an average dispersion factor was obtained for each selected downwind distance for all sixteen sectors.⁽¹⁾

Standard groups of meteorological data were interpolated from the specific site data. This grouping provided four stability classes based on vertical temperature gradient and five wind speed classes based on the Beaufort wind scale. Vertical temperature gradient classes were based on Regulatory Guide 1.23 (Safety Guide 23);⁽⁶⁾ Pasquill classes A, B, and C were classified as B (unstable), Pasquill classes D (neutral) and E (slightly stable) were not reclassified, and Pasquill classes F and G were classified as F (moderately stable).

Wind speed data were frequently available at only one height. In this event, the measured values were extrapolated to the 10-m level for building and vent release calculations and to the 100-m level for stack release calculations. Where measurements at two heights were available, the highest was extrapolated to 100 m and the lowest to 10 m using a standard power law extrapolation procedure.⁽⁷⁾

Calculations were made for a number of specific sites to yield average air concentrations and average radiation doses versus distance from the release point, i.e., the values represent the average over all of the wind directions that carry the airborne effluents over land. As previously stated, average sector values were used for this study.

Data on the maximum sector and the average of all 16 sectors were available for sixteen sites.⁽¹⁾ These data were used to determine the ratio of the

⁽¹⁾ U.S. AEC, Final Environmental Statement Concerning Proposed Rule-Making Action: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet Criteria "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, WASH-1258, Directorate of Regulatory Standards, Volume 1 of 3, Figure 6B-1, p. 6B-43 and Figure 6C-8, p. 6C-12, July 1973.

⁽⁶⁾ U.S. NRC, Regulatory Guide 1.23, "Onsite Meteorological Programs (Safety Guide 23)", February 1972.

⁽⁷⁾ Meteorology and Atomic Energy 1968, TID 24190, Edited by D. H. Slade, p. 73, July 1968.

minimum effective dilution (maximum dispersion factor) to the mean sector value, and are applied to the present study.

The value obtained for the ratio of dispersion factor in the maximum sector to the average sector dispersion factor was 2.5, and this value was used for all release heights in this study. Investigation of changes in this ratio with distance from the site determined that it remained essentially constant with distance from the point of release. The dispersion factors for the average sector as a function of height of release and downwind distance are shown in Figure B.7-1.

To assess the effect of increased stack height, atmospheric dispersion factors for stack heights of 150, 200 and 300 m (400, 660, and 980 ft) were estimated from the original joint frequency distributions of the 16 reactor sites. These values are also presented in Figure B.7-1.

Where large volumes of heated air are being ejected, the plume rise constant for momentum is estimated to be about $50 \text{ m}^2/\text{sec}$ ($540 \text{ ft}^2/\text{sec}$). Assuming an annual average wind speed of 2 to 3 m/sec, (4.5 to 6.7 mph) the increase in effective stack height due to momentum would be 20 to 25 m (66 to 82 ft). Plume rise due to buoyancy (heat effect) would add at least another 25 to 100 m (82 to 330 ft) of effective stack height, depending upon the temperature of the exhaust gases. Thus, the $\bar{X}/Q'$ values illustrated in Figure B.7-1 are larger than they would be if credit had been taken for buoyancy and momentum.

B.8 ECOLOGY

The description of the ecology at the site is divided into terrestrial and aquatic ecology. These are discussed below under separate subsections.

B.8.1 Terrestrial Ecology

Less than 20% of the land within 40 km (25 miles) of the plant site is covered with pristine vegetation.

Vegetation in the area was originally identified as supporting a climax deciduous forest. Remnants of the native climax hardwood forest are found on the larger islands, with some lesser stands in isolated pockets along the river bank. Nearly all accessible virgin woodland areas in the region have been

FIGURE B.7-1. Average ($\bar{X}/Q'$) Values Versus Distance in a Sector from the Reference River Site

burned, cut and plowed. Trees and shrub cover remain stable from year to year except for disturbances from cattle grazing or browsing, clearing, and cutting. Because annual grasses and shrubs originate each year from seed, their abundance shows pronounced fluctuation from seasonal and annual changes in environmental conditions, such as rainfall.

The area bordering the river in the vicinity of the plant consists predominantly of a willow-scrub thicket interspaced with cottonwood and poplar, whereas the areas adjacent to the border are largely agricultural crop and grazing land. Many of the terrestrial animals occur in vegetative cover bordering the river, and periodically forage in the adjacent agriculture areas. Some of the areas near the river are inundated by water from small swamps or marshes characterized by cattails (Typhus sp.) and other forms of marsh vegetation.

The shoreline and islands in this reach of the river are heavily utilized by waterfowl and shorebirds. Most waterfowl and shorebirds are migratory, and are present only seasonally, primarily during the summer and fall. The mallard and blue-winged teal are common spring breeders and summer residents. Common marsh birds are the red-winged blackbird, marsh wren, and great-blue heron. A variety of upland game birds and nongame birds utilize riparian habitats for nesting and feeding, as well as for protection from predators and adverse weather.

Bird hunting is mainly directed at waterfowl, both jump shooting and hunting from blinds. This portion of the river is not a preferred hunting area, nor is it used significantly as a resting area for migrating water-fowl. Ruffed grouse are occasionally hunted, but there is little hunting for other birds.

A list of wildlife considered to be rare, endangered, or threatened within the study state is shown in Table B.8-1. The southern bald eagle nests along the Atlantic and Gulf Coasts but moves northward after the nesting season and is occasionally sighted in the study area. The Arctic peregrine falcon nests in the treeless tundra and migrates southward through the midwestern states to the Gulf Coast and South America. The prairie falcon, which nests in other areas, is occasionally sighted in the area. The northern greater prairie chicken is also present at the reference site.

TABLE B.8-1. Wildlife Considered to be Rare,
Endangered, or Threatened Within
the Reference Site⁽⁸⁾

<u>Birds</u>	
Southern Bald Eagle	
Arctic Peregrine Falcon	
Prairie Falcon	
Northern Greater Prairie Chicken	

Some important mammals in the area are listed in Table B.8-2. Squirrel is the principal animal hunted in the area. A one-day season for hunting deer with a gun and a prolonged season for bow and arrow hunting suggest a limited deer population. There is also some hunting of fox and raccoon.

TABLE B.8-2. Some Important Mammals That Are Found
Near the Reference Site

Badger	Gray Squirrel	Raccoon
Bats	Ground Squirrels	Red Fox
Beaver	Harvest Mouse	Red Squirrel
Bobcat	Jumping Mice	Shrews
Chipmunks	Mink	Skunk
Cottontail Rabbit	Moles	Voies
Coyote	Muskrat	Weasels
Deer Mouse	Opossum	Whitetail Deer
Fox Squirrel	Packet Gopher	Whitetail Jackrabbits

Some important fur bearing animals include raccoon, mink, muskrat, beaver, and fox. Other wide-ranging fur-bearing mammals will probably occur in the vicinity from time to time; especially the coyote, bobcat and possibly the river otter.

⁽⁸⁾Bureau of Sport Fisheries and Wildlife Redbook, U.S. Department of Interior.

Broad-leaved trees characteristic of the study area include oak, elm, basswood, maple and hackberry. These species occur abundantly on the larger islands, with lesser stands in isolated pockets along the river banks. The climax community was probably comprised of maple and basswood. Farming, grazing and logging activities, however, have caused considerable change from the climax vegetation. Most areas that would be in active use at the reference site (e.g., buildings, parking lot) occupy land formerly cultivated. The portion of the site on the far side of the river is in various stages of recovery from weedy species. There is no evidence of the existence of rare or endangered vegetation species at the reference site.

The soil in the area is thin and varies from sand to silty loam, with an underlay of glacial till. The water table in lower areas is close to the surface, and during river flood these areas are frequently inundated.

The numerous ponds, lakes and swampy areas bounding the site provide nesting areas for waterfowl. Most nest fairly close to water.

B.8.2 Aquatic Ecology

The shoreline near the site consists of a sand and gravel base that merges with soil and riparian vegetation near the high water line.

The smallest organisms of concern in the river near the proposed site are planktonic forms. The plankton consists of both the photosynthetic phytoplankton and nonphotosynthetic zooplankton. The phytoplankton form the first major link in the aquatic food chain. At the site, phytoplankton originate largely from upstream areas, including impoundments, with some contribution from the periphyton in the immediate area. Diatoms are the principal form of phytoplankton.

The abundance of diatom populations varies in an annual cycle, with the greatest density occurring in late spring and the lowest density in midwinter. Phytoplankton production is influenced by seasonal changes in light intensity, water temperature, and quantities of dissolved nutrients.

The abundance of zooplankton populations also varies in an annual cycle with the greatest density occurring during the summer and fall. The primary

habitat of these forms is in the quiet and protected parts of the river. Zooplankton consume primarily phytoplankton and both forms of plankton are utilized as food by certain larger invertebrates and fish.

Diatoms and filamentous green algae dominate the periphyton, or algae communities attached to submerged substrates in the river. The periphyton population appears to be maintained in a subclimax condition, that is, typically in a viable state of recolonization and growth because of erosion by current flows and grazing by aquatic fauna. Seasonal productivity of periphyton corresponds to the annual cycle of phytoplankton growth, being highest in late spring and lowest in midwinter, and is also influenced by light, water temperature, and quantities of dissolved nutrients. Periods of high river turbidity and flow have a pronounced impact on periphyton growth, primarily because of siltation and scouring by high velocity currents. Benthic organisms in the river near the site include larvae of aquatic insects, molluscs, and crustacea as the most important producers and consumers. Chironomoids (Diptera, Chironomidae), caddisflies, and mayflies are the dominant insects. Molluscs and crustacea are present.

The density of the benthos varies with substrate type, which in turn is influenced by current flow. Populations include organisms characteristic of the clean, stone riffle areas and others characteristics of sand silt areas. Generally, the density of the benthos is lower in deep water than in shallow water. Benthic organisms form secondary or intermediate links in the food chain leading to the production of more valuable forms. Aquatic insects are the major food items in the diet of many river fish, and insect larvae are the dominant components of the river drift.

Rooted macrophytes occur in certain areas along the margin of the river. Most of these are emergents such as Thypha and Juncus but some are floating plants such as Potamogeton and Lemna, while a few are submerged plants such as Anacharis and Ceratophyllum. Biomass of macrophytes varies seasonally, with the greatest concentrations occurring during the summer and fall and the smallest concentrations during the winter. Macrophytes provide food for waterfowl and riparian mammals, shelter for small fish, hiding areas for large fish, and production areas for many microscopic river invertebrates.

Fish form the upper trophic level of aquatic organisms near the proposed site. There are several species of fish known to be present. No significant commercial fishery exists. Sport fishing intensity is greatest in the spring and fall. Most fish spawn in the spring and early summer. Suckers and others have short upriver spawning migrations. Andromous species are present.

None of the fish present is considered to be a rare or endangered species. Sewage wastes released upriver have been subjected to primary and secondary treatment and do not contribute a large amount of pollution in the river. However, the water may contain higher than normal amounts of phosphates and nitrates following periods of high rains due to the influx of agricultural fertilizers. Insecticides from crop spraying are known to be present occasionally in the water at the site, although fish kills from toxic chemicals have yet to be recorded.

REFERENCES

1. U.S. AEC, Final Environmental Statement Concerning Proposed Rule-making Action: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criteria "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, WASH-1258, Directorate of Regulatory Standards, Volume 1 of 3, Figure 6B-1, p. 6B-43 and Figure 6C-8, p. 6C-12, July 1973.
2. U.S. AEC, Final Environmental Statement Related to Operation of Monticello Nuclear Generating Plant, Docket No. 50-263, pp. II-15 through II-26, November 1972.
3. United Nations Scientific Committee on the Effects of Atomic Radiation, Ionizing Radiation: Levels and Effects, Volume 1, United Nations, p. 29-63, 1972.
4. U.S. AEC, Final Environmental Statement Concerning Proposed Rule-making Action: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criteria "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, WASH-1258, Directorate of Regulatory Standards, Volume 1 of 3, Table 6B-6, p. 6B-36, July 1973.
5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.109. "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50," Appendix I, Table E-5, p. 1.109-40, October 1977.
6. U.S. NRC, Regulatory Guide 1.23, "Onsite Meteorological Programs (Safety Guide 23)," February 1972.
7. Meteorology and Atomic Energy 1968, TID 24190, Edited by D. H. Slade, p. 73, July 1968.
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APPENDIX C

ESTIMATES OF RESIDUAL RADIOACTIVITY

The radionuclide inventory present in a shutdown, defueled-reactor facility is contained in two principal sources: 1) the activated reactor vessel, internals and biological shield; and 2) the activated corrosion products and fission products collected in filters and ion exchange resins, and deposited throughout the piping systems. In this appendix, the details of the calculations made to estimate the quantities of radionuclides contained in the activated reactor vessel, internals and biological shield at the time of reactor shutdown are presented. The principal radionuclides are characterized in terms of the parent isotope, the means of production, and the nature of the emitted radiation. The models, methods and results of calculations used to estimate radiation dose rates from the activated materials are presented. Radiation exposure data from operating PWRs are presented to provide a base for calculating occupational radiation exposures during decommissioning operations. The quantities of activated corrosion products deposited on the internal surfaces of the reactor coolant system components are also calculated, based on radiation data measured on the surfaces of the piping systems.

C.1 CALCULATION OF THE INVENTORY OF NEUTRON-ACTIVATED MATERIALS

The types and quantities of radionuclides present in the activated reactor vessel, vessel internals, and biological shield at the end of the operating life of a PWR have been calculated. The calculational model employed contained three radial zones of fuel having compositions typical of a PWR bundle after full power operation for 6, 18 and 30 months, respectively, simulating the mid-cycle conditions in an equilibrium core, with 500 weight parts per million boron in the light water coolant. The atomic densities of the fuel isotopes and associated material in the three fuel zones, given in Table C.1-1,

TABLE C.1-1. Material Densities Used in ANISN Calculations

Atomic Densities in Homogenized Fuel Zone (atoms/cm³)
After Full Power Operation For:

	6 Mo.	18 Mo.	30 Mo.
²³⁵ U	2.0183×10^{20}	1.3090×10^{20}	8.2941×10^{19}
²³⁶ U	8.2672×10^{18}	2.0970×10^{19}	2.8190×10^{19}
²³⁸ U	7.3204×10^{21}	7.2686×10^{21}	7.1961×10^{21}
²³⁹ Pu	1.6402×10^{19}	3.3812×10^{19}	3.7362×10^{19}
²⁴⁰ Pu	2.5687×10^{18}	9.4543×10^{18}	1.4696×10^{19}
²⁴¹ Pu	5.3073×10^{17}	3.4910×10^{18}	7.8452×10^{18}
²⁴² Pu	3.5402×10^{16}	5.1519×10^{17}	2.3532×10^{18}
¹³⁵ Xe	4.1820×10^{14}	5.8010×10^{14}	6.2330×10^{14}
¹⁰ B	3.0878×10^{18}	3.0878×10^{18}	3.0878×10^{18}
Zr	4.2732×10^{21}	4.2732×10^{21}	4.2732×10^{21}
O	2.9101×10^{22}	2.9101×10^{22}	2.9101×10^{22}
H	2.7765×10^{22}	2.7765×10^{22}	2.7765×10^{22}

Atomic Densities in Surrounding Regions (atoms/cm³)

	Coolant	304 Stainless Steel	Carbon Steel	Void	Concrete
¹⁰ B	5.5744×10^{18}				
O	2.5073×10^{22}			9.03×10^{18}	4.208×10^{22}
H	5.0146×10^{22}				7.513×10^{21}
Ni		8.542×10^{21}			
Cr		1.662×10^{22}			
Fe		6.074×10^{22}	8.461×10^{22}		3.708×10^{21}
Ca					3.462×10^{21}
Si					1.530×10^{22}
Al					3.297×10^{21}

were obtained from an ORIGEN⁽¹⁾ calculation for a PWR bundle at the three exposures given above. Epithermal neutron cross sections were obtained from the ENDF/B-IV⁽²⁾ library using ETOG.⁽³⁾ Thermal neutron cross sections were obtained from the same library and appropriately averaged over spectrum and space by doing fuel pin cell calculations with the Battelle Revised THERMOS⁽⁴⁾ code.

The homogenized fuel region cross sections and the material cross sections for the reactor vessel, internals, and biological shield were used in one-dimensional, 30 energy group ANISN⁽⁵⁾ calculations to obtain the flux distributions throughout the reactor vessel and its adjacent structures.

The models used in the ANISN calculations performed for this study are shown in Figure C.1-1. The rectangular core shroud (baffle) was converted to a cylinder at the equivalent radius for the radial flux calculation. The fuel bundle tie plates, core grid plates, support structure and control rod guide tubes, etc., were simulated by homogeneous regions containing the appropriate stainless steel and water fractions for the axial flux calculation. The compositions assumed for the steel components and for the biological shield are also given in Table C.1-1.

(1) M. J. Bell, ORIGEN - The ORNL Isotope Generation and Depletion Code. ORNL-4628, Oak Ridge National Laboratory, Oak Ridge, TN, May 1973.

(2) H. C. Honeck, ENDF/B Specifications for an Evaluated Nuclear Data File for Reactor Applications. BNL-50066 (T-467), Brookhaven National Laboratory, May 1966, Revised July 1967, and O. Ozer and D. Garber, ENDF/B Summary Documentation. ENDF/B-201, May 1973.

(3) D. E. Kusner, R. A. Dannels, and S. Kellman, ETOG-1, A FORTRAN IV Program to Process Data from the ENDF/B File to the MUFT, GAM, and ANISN Format. WCAP-3845-1, Westinghouse Electric Corp., December 1969.

(4) C. L. Bennett, W. I. Purcell, BRT-I: Battelle-Revised-THERMOS. BNWL-1434, Battelle, Pacific Northwest Laboratories June 1970.

(5) W. E. Engle, Jr., A Users Manual for ANISN, A One Dimensional Discrete Ordinates Transport Code With Anisotropic Scattering. K-1693, Oak Ridge National Laboratory, Oak Ridge, TN, March 1967.

FIGURE C.1-1. Models for ANISN Calculations

The calculated radial thermal neutron flux distribution is shown in Figure C.1-2, and the axial flux distribution normalized at the core mid-plane is shown in Figure C.1-3. The 30 energy group fluxes obtained from the ANISN calculations were collapsed into three energy groups and were used as input for a series of ORIGEN calculations to predict the buildup of the various activation products at different locations throughout the reactor vessel and its surrounding biological shield during 30 effective full power years of operation. The resulting estimates are maximized in that no allowances were made for decay and nonactivation during refueling and maintenance outages.

The material isotopic compositions and elemental weight percentages in the reactor structures that were used in the ORIGEN calculations are given in Table C.1-2. The values listed in Table C.1-2 for the major constituents of the reactor materials are based on nominal handbook compositions for these materials. The isotopic fractions given for each element are those present in nature.

Two elements on the list are worthy of special note, ^{59}Co and ^{93}Nb . Cobalt occurs naturally as an impurity in iron and nickel with concentration depending upon the geographic source of the ore used in producing the metal. The concentration of the cobalt impurity is known to vary from 0.05 to 0.15% by weight in 304 stainless steel alloys, and from 0.006 to 0.012% by weight in the SA533 steel alloy in the pressure vessel. Since the exact composition of the 304 stainless steel alloys used in the core components of a given reactor is generally not known, the upper and lower boundary values for the ^{59}Co concentrations were used in the activation calculations to establish the range of possible values for the concentration of the ^{60}Co radionuclide in the reactor components.

The interest in niobium arises from a study at Cornell⁽⁶⁾ in which the possible trace elements in stainless steel and in steel were surveyed for likely sources of long-lived activation products. The naturally occurring

⁽⁶⁾ J. J. Stevens, Jr. and R. O. Pohl, Trace Elements in Reactor Steels: Implications for Decommissioning, Report #2882, Laboratory of Atomic and Solid State Physics and Material Science Center, Cornell University, August 1977.

FIGURE C.1-2. Radial Thermal Neutron Flux Distribution at Core Axial Midplane

FIGURE C.1-3. Normalized Axial Activation Distribution

TABLE C.1-2. Material Compositions Used in ORIGIN Calculations

Isotope	Initial Isotope Abundance	Nominal Elemental Weight Percent in		
		Internals 304 Stainless	Pressure Vessel 5A533 Steel	Bio Shield Concrete
¹ H	99 985	-	-	0 53
² H	0 015			
¹² C	98 89	0 08	0 24	0 03
¹³ C	1 11			
¹⁴ N	99 64	0 04	0 01	0 0013
¹⁶ O	99 756	-	--	47
¹⁷ O	0 039			
¹⁸ O	0 205			
²³ Na	100 0	-	-	1 6
²⁴ Mg	78 99	-	--	0 24
²⁵ Mg	10 0			
²⁶ Mg	11 01			
²⁷ Al	100 0	-	--	4 4
²⁸ S _i	92 2	1 00	0 31	30
²⁹ S _i	4 7			
³⁰ S _i	3 1			
³¹ P	100 0	0 0004	0 035	0 0045
³² S	95 0	0 03	0 04	0 12
³³ S	0 75			
³⁴ S	4 2			
³⁶ S	0 015			
³⁹ K	93 3	--	-	1 8
⁴⁰ K	0 012			
⁴¹ K	6 7			
⁴⁰ Ca	96 94	-	--	7 9
⁴² Ca	0 65			
⁴³ Ca	0 14			
⁴⁴ Ca	2 08			
⁴⁶ Ca	0 003			
⁴⁸ Ca	0 19			
⁵⁰ Cr	4 35	17 5	0 51	0 068
⁵² Cr	83 79			
⁵³ Cr	9 5			
⁵⁴ Cr	2 36			
⁵⁴ Fe	8	68 5	94 63	5 8 (includes rebar)
⁵⁶ Fe	91 7			
⁵⁷ Fe	2 14			
⁵⁸ Fe	0 31			
⁵⁵ Mn	100 0	2 0	2 80	0 20
⁵⁸ Ni	68 3	10 1	0 72	0 094
⁶⁰ Ni	26 1			
⁶¹ Ni	1 1			
⁶⁴ Ni	0 9			
⁵⁹ Co	100 0	0 05-0 15	0 006-0 012	0 00077
⁶³ Cu	69 1	0 15	0 076	0 014
⁹² Mo	14 8	0 05	0 63	0 082
⁹⁴ Mo	9 1			
⁹⁵ Mo	15 9			
⁹⁶ Mo	16 7			
⁹⁷ Mo	9 5			
⁹⁸ Mo	24 4			
¹⁰⁰ Mo	9 6			
⁹³ Nb	100 0	0 016		

isotope ^{93}Nb was identified to be the only trace element likely to produce an activation product of any significance to decommissioning. Data given by Heger⁽⁷⁾ and later measurements at Cornell⁽⁶⁾ and at Battelle-Northwest⁽⁸⁾ showed a reasonable upper boundary value for the concentration of niobium in 304 stainless steel to be about 0.016% by weight in the alloy.

The ORIGEN code computes the production and loss of radionuclides based on initial concentrations of parent isotopes, neutron flux level, length of time exposed to the neutron flux, the isotopic cross sections for radioisotope production and removal, and the rates of radioactive decay. The neutron flux levels computed for the various components at the axial mid-plane of the fuel core were used in the ORIGEN code to predict the concentrations of the radioactive species that would be present in those components at shutdown after 30 effective full-power years (EFPY) of reactor operation. The calculated buildups of radionuclides in the reactor vessel internals, shown in Figure C.1-4, indicate that the ^{55}Fe concentration saturates after about 16 years at full power. The ^{60}Co radionuclide saturation level is more strongly flux-dependent, reaching saturation in the core shroud after about 22 years but not quite reaching saturation in the core barrel and vessel cladding after 30 effective full-power years. The ^{59}Ni and ^{14}C radionuclide concentrations increase linearly with time and do not saturate within the time span of interest. The ^{63}Ni and ^{94}Nb radionuclide buildups exhibit a slight curvature, indicating a small saturation effect over the 30-year time span. Thus, the levels of ^{55}Fe and ^{60}Co radioactivity are relatively independent of the number of years of operation (beyond ~20 years) while the

(6) J. J. Stevens, Jr. and R. O. Pohl, Trace Elements in Reactor Steels: Implications for Decommissioning, Report #2882, Laboratory of Atomic and Solid State Physics and Material Science Center, Cornell University, August 1977.

(7) J. J. Heger, "Residual Elements in Stainless Steel-General Characteristics," Effects of Residual Elements on Properties of Austenitic Stainless Steels, ASTM STP 418, Am. Soc. Testing Mats., p. 120, 1967.

(8) Private Communication: Letter from M. D. Merz to R. I. Smith, dated January 18, 1978, Subject: Review of Data on Nb Concentrations in Stainless Steel.

FIGURE C.1-4. Buildup of Activation Products in PWR vessel Internals as a Function of EFPY

levels of ^{59}Ni , ^{94}Nb , ^{63}Ni and ^{14}C radioactivity continue to increase almost linearly with years of reactor operation.

C.1.1 Radioactivity Induced In Steel Components

The concentrations of the various radionuclides in neutron-activated steel components of the reactor vessel at the axial mid plane of the fuel core, computed at the time of reactor shutdown after 30 EFPY of operation, are listed in Table C.1-3. Radionuclides with half-lives shorter than thirty-five days have been omitted from the table since those nuclides will decay with sufficient rapidity to have little impact on decommissioning.

To compute the total inventory of these radionuclides in the activated components, graphical integrations were performed over the normalized flux distribution shown in Figure C.1-3, for the axial regions indicated on the

TABLE C.1-3. Radioactivity Levels in Major Activated Reactor Components at Time of Reactor Shutdown

Isotope	Half-Life	Core Mid-Plane Radioactivities (Ci/m ³)					Upper Grid Plant (a)	Lower Grid Plate (a)
		Shroud	Lower 4.72 m of Core Barrel	Thermal Shields	Vessel Inner Cladding	Lower 5.02 m of Vessel Wall		
⁹⁵ Nb	35 day	2.0 x 10 ³	7.6 x 10 ⁰	3.5 x 10 ⁰	5.6 x 10 ⁻³	1.7 x 10 ⁻³		
⁵⁹ Fe	45 day	4.6 x 10 ⁴	4.4 x 10 ³	2.0 x 10 ³	1.0 x 10 ²	2.7 x 10 ¹		
⁵⁸ Co	72 day	1.5 x 10 ⁵	1.0 x 10 ⁴	4.6 x 10 ³	3.3 x 10 ²	6.6 x 10 ⁰		
⁹⁵ Zr	65 day	1.1 x 10 ⁻¹	6.2 x 10 ⁻³	2.9 x 10 ⁻³	2.0 x 10 ⁻⁴	7.2 x 10 ⁻⁴		
⁶⁵ Zn	245 day	1.2 x 10 ²	1.1 x 10 ⁰	5.0 x 10 ⁻¹	6.7 x 10 ⁻⁴	3.5 x 10 ⁻⁵		
⁵⁴ Mn	~300 day	6.8 x 10 ⁴	3.7 x 10 ³	1.7 x 10 ³	1.2 x 10 ²	4.7 x 10 ¹		
⁵⁵ Fe	2.7 yr	1.3 x 10 ⁶	1.5 x 10 ⁵	6.7 x 10 ⁴	3.5 x 10 ³	7.2 x 10 ²		
⁶⁰ Co (b)	upper lower	9.6 x 10 ⁵	9.3 x 10 ⁴	4.7 x 10 ⁴	2.5 x 10 ³	7.5 x 10 ¹		
		3.2 x 10 ⁵	3.1 x 10 ⁴	1.6 x 10 ⁴	8.2 x 10 ²	2.5 x 10 ¹		
⁶³ Ni	~100 yr	1.2 x 10 ⁵	1.5 x 10 ⁴	6.8 x 10 ³	3.6 x 10 ²	3.8 x 10 ⁰		
⁹³ Mo	~3000 yr	3.6 x 10 ⁻¹	5.2 x 10 ⁻²	2.4 x 10 ⁻²	1.2 x 10 ⁻³	1.3 x 10 ⁻³		
¹⁴ C	~5,750 yr	1.5 x 10 ²	1.8 x 10 ¹	8.3 x 10 ⁰	4.0 x 10 ⁻¹	1.9 x 10 ⁻²		
⁹⁴ Nb	~20,000 yr	5.4 x 10 ⁰	2.6 x 10 ⁻¹	1.2 x 10 ⁻¹	9.5 x 10 ⁻³	--		
⁵⁹ Ni	~80,000 yr	7.4 x 10 ²	1.3 x 10 ²	5.0 x 10 ¹	3.0 x 10 ⁰	3.2 x 10 ⁻²		
Sum (Ci/m ³)		2.97 x 10 ⁶	3.07 x 10 ⁵	1.45 x 10 ⁵	7.73 x 10 ³	9.04 x 10 ²	2.97 x 10 ⁶	2.97 x 10 ⁶
Average/Peak		0.755	0.637	0.778	0.637	0.637	0.003 x 4.74 (c)	0.08 x 4.74 (c)
Ci/kg (d)		2.787 x 10 ²	2.433 x 10 ¹	1.403 x 10 ¹	7.621 x 10 ⁻¹	7.164 x 10 ⁻²	5.254 x 10 ⁰	1.403 x 10 ²
Weight of Material (kg)		12,312	26,783	10,413	2,074	245,582	4,627	3.946
Sum (Ci) (e)		3.431 x 10 ⁶	6.516 x 10 ⁵	1.461 x 10 ⁵	1.581 x 10 ³	1.759 x 10 ⁴	2.431 x 10 ⁴	5.534 x 10 ⁵
TOTAL - Radioactivity					4.826 x 10 ⁶ Curies	1.786 x 10 ¹⁷ becquerels		

(a) Normalized to shroud

(b) Upper and lower bounds were computed using the maximum and minimum levels of ⁵⁹Co contaminant in the materials. All totals were computed using the upper bound values.

(c) $\frac{\text{Activity (Plate Average)}}{\text{Activity (Plate Edge)}} = 4.74$, $\frac{\text{Activity (Shroud at Plate Location)}}{\text{Activity (Shroud at Axial Midplane)}} = \begin{cases} 0.005, \text{ upper plate} \\ 0.08, \text{ lower plate} \end{cases}$

(d) Conversion factor assumes stainless steel density of 8.038 x 10³ kg/m³ (0.29 lb/in.³).

(e) The number of significant figures carried is for computational accuracy and does not imply precision to four places.

figure and listed in the column headings in Table C.1-3 to obtain the Average-to-Peak ratios, also listed in Table C.1-3. For the upper and lower grid plates, a graphical integration was performed in the radial direction. Converting the radioactivity in curies/m³ to curies/kg of material and multiplying by the weight of material in each component in the region of interest yields the total number of curies contained in that component at the time of reactor shutdown, as given on the last line of Table C.1-3. Thus, the total number of curies contained in the activated vessel and internals was calculated to be approximately 4.8 million, not including contributions from the reactor control rod elements, utilizing the upper bound values for ⁶⁰Co activity.

Activations in the core structures above and below the regions indicated on Figure C.1-3 were considered to contribute relatively insignificant amounts to the total inventory. A summary of the estimated radioactivity levels in the components of the reactor vessel is given in Table C.1-4.

C.1.2 Radioactivity Induced In The Concrete Bio-Shield

The same ORIGEN calculations that produced the activated steel data in Table C.1-3 also produced values for activations in the concrete biological shield. Based on the initial compositions listed in Table C.1-2, the specific activities of the radionuclides in the concrete were calculated for 30 EFPY of reactor operation. The values calculated at the inner most surface of the concrete are given in Table C.1-5, for the time of reactor shutdown and for times after shutdown of 1, 10, and 100 years. The distribution of radioactivity as a function of radial position through the shield is shown in Figure C.1-5 for three of the principal radionuclides present. A graphical integration of these distributions, modified for the axial average to peak flux ratio from Table C.1-3, resulted in an estimation of 1,200 curies of radioactivity being present in the total concrete shield.

An analysis of a core sample from the Elk River Reactor biological shield was made during the dismantlement of that facility. The reactor had been shutdown for about 3 1/2 years at the time of measurements. Using gamma-ray

TABLE C.1-4. Summary of Estimated Radioactivity Levels in Activated Steel Components

<u>Component</u>	<u>Estimated Radioactivity (Ci)</u>
Pressure Vessel,	
Cylindrical Wall	19,171
Vessel Head	<10
Vessel Bottom	<10
Upper Core	
Support Assembly	<10
Upper Support	
Columns	<100
Upper Core Barrel	<1000
Upper Core Grid	
Plate	24,310
Guide Tubes	<100
Lower Core Barrel	651,600
Thermal Shields	146,100
Core Shroud	3,431,000
Lower Core Grid	
Plate	553,400
Lower Support	
Columns	<10,000
Lower Core Forging	<2,500
Misc. Internals	<2,000
Reactor Cavity	
Liner	<10
Totals	<4,841,321

TABLE C.1-5. Calculated Radioactivity in the Bio-Shield Concrete at the Inner Surface of the Shield

Radio-Nuclide	Principal Decay Mode	Radioactivity (Ci/m ³) With Decay Since Shutdown of:			
		0 Years	1 Year	10 Years	100 Years
³ H	β ⁻	2.29 x 10 ⁻⁵	2.17 x 10 ⁻⁵	1.31 x 10 ⁻⁵	8.20 x 10 ⁻⁸
¹⁴ C	β ⁻	6.94 x 10 ⁻⁴	6.94 x 10 ⁻⁴		
³³ P	β ⁻	3.24 x 10 ⁻¹	1.30 x 10 ⁻⁵	--	--
³⁵ S	β ⁻	3.17 x 10 ⁻²	1.79 x 10 ⁻³	1.02 x 10 ⁻¹⁴	--
³⁶ Cl	β ⁻ , γ	8.40 x 10 ⁻⁶	8.40 x 10 ⁻⁶	8.40 x 10 ⁻⁶	8.40 x 10 ⁻⁶
³⁷ Ar	γ	2.15 x 10 ⁻¹	1.55 x 10 ⁻⁴	--	--
³⁹ Ar	β ⁻	3.96 x 10 ⁻²	3.95 x 10 ⁻²	3.86 x 10 ⁻²	3.06 x 10 ⁻²
⁴⁰ K	β ⁻ , β ⁺ , γ	3.76 x 10 ⁻⁵	3.76 x 10 ⁻⁵	3.76 x 10 ⁻⁵	3.76 x 10 ⁻⁵
⁴¹ Ca	γ	7.00 x 10 ⁻³	7.00 x 10 ⁻³	7.00 x 10 ⁻³	6.99 x 10 ⁻³
⁴⁵ Ca	β ⁻	3.66 x 10 ⁰	7.90 x 10 ⁻¹	7.97 x 10 ⁻⁷	--
⁴⁶ Sc	β ⁻	1.86 x 10 ⁻⁴	8.11 x 10 ⁻⁶	1.47 x 10 ⁻¹⁷	--
⁵¹ Cr	γ	1.04 x 10 ⁻¹	1.15 x 10 ⁻⁵	--	--
⁵⁴ Mn	γ	1.68 x 10 ⁻¹	7.28 x 10 ⁻²	3.95 x 10 ⁻⁵	--
⁵⁵ Fe	γ	3.01 x 10 ¹	2.31 x 10 ¹	2.10 x 10 ⁰	8.01 x 10 ⁻¹¹
⁵⁹ Fe	β ⁻ , γ	9.99 x 10 ⁻¹	3.60 x 10 ⁻³	--	--
⁵⁸ Co	β ⁺ , γ	2.15 x 10 ⁻²	6.17 x 10 ⁻⁴	8.21 x 10 ⁻¹⁸	--
⁶⁰ Co	β ⁻ , γ	6.69 x 10 ⁻¹	5.87 x 10 ⁻¹	1.79 x 10 ⁻¹	1.27 x 10 ⁻⁶
⁵⁹ Ni	γ	1.19 x 10 ⁻³	1.19 x 10 ⁻³	1.19 x 10 ⁻³	1.18 x 10 ⁻³
⁶³ Ni	β ⁻	1.40 x 10 ⁻¹	1.39 x 10 ⁻¹	1.30 x 10 ⁻¹	6.61 x 10 ⁻²
⁶⁵ Zn	β ⁺ , γ	4.47 x 10 ⁻⁷	1.59 x 10 ⁻⁷	1.46 x 10 ¹¹	--
^{93m} Nb	γ	2.77 x 10 ⁻⁵	2.82 x 10 ⁻⁵	3.93 x 10 ⁻⁵	5.63 x 10 ⁻⁵
⁹⁵ Nb	β ⁻ , γ	5.40 x 10 ⁻⁶	1.05 x 10 ⁻⁷	6.06 x 10 ⁻²³	--
⁹³ Mo	γ	6.69 x 10 ⁻⁵	6.69 x 10 ⁻⁵	6.68 x 10 ⁻⁵	6.64 x 10 ⁻⁵
⁹⁹ Tc	β ⁻	4.93 x 10 ⁻⁵	4.93 x 10 ⁻⁵	4.93 x 10 ⁻⁵	4.93 x 10 ⁻⁵
Totals		36.5	24.7	2.45	0.106

Note: No rare earths are included in the calculation since definitive data on probable initial concentrations of rare earths in concrete are not available.

FIGURE C.1-5. Radial Distribution of Radioactivity in the Concrete Biological Shield at Reactor Shutdown

spectroscopy, radioactivity levels per gram of concrete of 0.38 $\mu\text{Ci/g}$ (^{60}Co), 0.31 $\mu\text{Ci/g}$ (^{152}Eu) and 0.0144 $\mu\text{Ci/g}$ (^{22}Na) were found at the inner surface of the shield.⁽⁹⁾ The radiation dose rate at the inner surface of the concrete, due to the concrete, was found to be about 135 mR/hr.

A review of our activation calculation and of the possible ways to produce ^{22}Na has led this author to the conclusion that it is unlikely that any ^{22}Na was present. The 511 KeV gamma ray associated with ^{22}Na is also associated with a great many other radioactive species, and was probably mistakenly attributed to ^{22}Na .

The presence of ^{152}Eu is not surprising, considering the probability of there being trace amounts of rare earths in the concrete. Unfortunately, no quantitative data are available on the probable concentrations of rare earths in concrete. A review of the (n,γ) reactions for the isotopes of cerium, samarium, europium, gadolinium, dysprosium, and lutetium showed only two that have sufficiently large capture cross sections and half-lives long enough to be of interest or concern in the dismantlement of the concrete bio-shield. These were:

- The 47.8% abundant ^{151}Eu (n,γ) ^{152}Eu ($T_{1/2} = 12.7$ years), and
- The 52.2% abundant ^{153}Eu (n,γ) ^{154}Eu ($T_{1/2} = 18$ years).

The thermal neutron capture cross section for ^{151}Eu is about 8,700 barns and for ^{153}Eu is about 320 barns. The remaining reactions have either very small capture cross sections or very short half lives, or both. Considering the measured strength of the ^{152}Eu radionuclide in the concrete at Elk River, it is somewhat surprising that no ^{154}Eu was detected, since the parent isotopes are about equally abundant and the ^{153}Eu capture cross section is only about a factor of 30 smaller than the ^{151}Eu capture cross section.

If a definitive estimate of the inventory of radionuclides in the bioshield concrete is needed for decommissioning, it will be necessary to either analyze

⁽⁹⁾ AEC-Elk River Reactor Dismantling Plan, SS-836, Rural Cooperative Power Association, August 27, 1971.

a sample of the concrete for trace elements and calculate the probable activation levels or to obtain a core sample from the shield after reactor shutdown and to analyze that sample using both beta and gamma-ray spectroscopy. The potentially large impact of a minor trace constituent such as europium on the total radionuclide inventory in the concrete makes an a priori prediction of that inventory to within a factor of ten unlikely .

C.2 CHARACTERIZATION OF THE PRINCIPAL ACTIVATION PRODUCTS

Three steps are necessary to compute the radiation dose rate from the surface of activated material after the levels of activation have been determined. First, the radiation emissions must be characterized in terms of energy and probability of decay. Secondly, the transport of the radiation through the bulk material must be calculated, and thirdly, the resultant dose rate from the surface emanations must be computed. The characterizations of the principal activation products are given in the following paragraphs.

Production of ^{14}C in the structural portions of a PWR takes place via two routes, by multiple neutron capture in ^{12}C and ^{13}C , and by neutron capture - proton emission in ^{14}N , with the $^{14}\text{N} (n,p) ^{14}\text{C}$ reaction being the predominant one. Both carbon and nitrogen are present in stainless steel and in carbon steel in trace quantities. ^{14}C decays with a half life of about 5750 years via β^- emission with a maximum beta energy of 156 keV. The contribution of ^{14}C to external occupational radiation exposure is insignificant since the radioactivity is bound tightly in the metal alloys. The principal concern is to prevent its release into the biosphere where it could enter biological systems.

^{55}Fe is produced principally by thermal neutron capture in 5.84% abundant ^{54}Fe in iron. The neutron capture cross section is small, about 3 barns. ^{55}Fe decays by electron capture (EC) to $^{55}_{25}\text{Mn}$ with a half-life of about 2.7 years. The associated manganese x-rays and Auger electrons were characterized as described for ^{59}Ni from the same data sources to produce a source as:

Energy	Events Per Electron Capture		
	X-Rays	Electrons	Sum
5.895 keV	0.236	0.668	0.904
6.51 keV	0.025	0.071	0.096

Other radiations occur with energies less than 0.72 keV. The external exposure from the electron radiations do not contribute to the skin dose. The ^{55}Fe transition is an allowed one and thus the s-electron capture contribution calculated by Glauber and Martin has been confirmed by measurements by Biavati⁽¹⁰⁾ extending down to 17 keV gamma energy. A pointwise probability distribution was constructed to describe IB gamma spectrum up to the maximum 220 keV gamma energy.

$^{59}_{28}\text{Ni}$

^{59}Ni is produced principally by thermal neutron capture in 67.76% abundant ^{58}Ni in nickel. The neutron capture cross section is small, about 4.4 barns. ^{59}Ni decays by capture of an orbital electron (EC) by the nucleus to form the nucleus $^{59}_{27}\text{Co}$ which decays with a half-life of approximately 80,000 years. Each EC event is accompanied by the emission of x-rays and electrons characteristic of cobalt. In this region of atomic number, most of the x-rays are not emitted and Auger electrons of the same energy appear instead. The K-shell x-rays were characterized by the energies and probabilities tabulated in Reference 11. The fractional yields of K x-rays were taken from the Table of Isotopes.⁽¹²⁾ The k_{α} and k_{β} doublets were combined and represented as a single radiation at a weighted energy in the usual fashion to obtain the following characterization:

- (10) M. H. Biavati, S. J. Nassif and C. S. Wu, "Internal Bremsstrahlung Spectrum Accompanying 1S Electron Capture in Decay of ^{55}Fe , ^{131}Cs and ^{204}Tl ." Physical Review. 25:1364-1372, 1962.
- (11) CRC Handbook of Chemistry and Physics. 53rd Edition, 1972-3, CRC Press, Cleveland, Ohio, 1972.
- (12) C. M. Lederer, J. M. Hollander and I. Perlman, Table of Isotopes, Sixth Edition, John Wiley & Sons, New York, 1967.

Energy	Events Per Electron Capture		
	X-Rays	Electrons	Sum
6.925 keV	0.289	0.596	0.885
7.67 keV	0.038	0.077	0.115

Other x and e^- radiations occur with energies less than 0.87 keV. These are almost totally absorbed in the source. The maximum electron energies are lower than the 68 keV required to penetrate the outer dead layers of skin and do not contribute to a skin dose.

The EC decay events are accompanied by a continuous spectrum of gamma rays to 1.07 MeV from the inner bremsstrahlung (IB) process. Because of the spin and parities of the nuclear states involved ($3/2^- \rightarrow 7/2^-$), the transition is classed as second nonunique forbidden. The energy dependence of the IB gammas from capture of s-electrons for this type of transition has been calculated by Cutkosky.⁽¹³⁾ Saraf⁽¹⁴⁾ measured the IB shape from gamma energies greater than 100 keV to be in good agreement with the theory. This shape was then chosen for that due to s-electron capture but was normalized to the higher absolute yield obtained later by Schmorak⁽¹⁵⁾ for energies greater than 600 keV. At lower energies the IB contributions calculated for p-electron capture by Glauber and Martin⁽¹⁶⁾ with relativistic and atomic screening corrections were added. The gamma ray spectrum used was expressed as pointwise probability distribution for gamma energies up to 1.07 MeV. The methodology employed to derive the IB gamma spectrum is consistent with that given in a recent article in Reviews of Modern Physics.⁽¹⁷⁾

(13) R. E. Cutkosky, "Internal Bremsstrahlung." Physical Review. 95:1222-1225, 1954.

(14) Babula Saraf, "Inner Bremsstrahlung Emitted in Electron Capture Decay of ^{59}Ni , ^{37}A , and ^{55}Fe ." Physical Review. 102:46-473, 1956.

(15) Marcel Schmorak, "Virtual Electron Capture in ^{59}Ni ." Physical Review. 129:1668-1672, 1963.

(16) R. T. Glauber and P. C. Martin, "Radioactive Capture of Orbital Electrons." Physical Review. 104:158-175, 1956.

(17) W. Bambynek, H. Behrens, M. H. Chen, B. Crasemann, M. L. Fitzpatrick, K. W. D. Ledingham, H. Genz, M. Mutterer and R. L. Intemann, "Orbital Electron Capture by the Nucleus." Review of Modern Physics, 49:1, January 1977.

$^{60}_{27}\text{Co}$

^{60}Co is produced principally by thermal neutron capture in 100% abundant ^{59}Co . The neutron capture cross section is about 37 barns. ^{60}Co decays by β^- emission to excited nuclear levels of $^{60}_{28}\text{Ni}$ with a half-life of 5.27 years. Over 99% of the decays are to a state at 2.5057 MeV which decays to the ground state with the sequential emission of two gamma rays of 1.173 MeV and 1.332 MeV.⁽¹²⁾ The continuum allowed beta spectrum with a maximum energy of 0.32 MeV contributes a minor fraction to the dose rate.

$^{63}_{28}\text{Ni}$

^{63}Ni is produced principally by thermal neutron capture in 3.66% abundant ^{62}Ni in nickel. The neutron capture cross section is about 15 barns. ^{63}Ni undergoes beta decay to $^{63}_{29}\text{Cu}$ with a half-life of about 100 years.⁽¹⁸⁾ The beta transition is allowed and has a maximum energy (E_m) of about 67 keV.⁽¹²⁾ For an allowed transition of $E_m \ll M_0 c^2$ (0.51 MeV), the Fermi function $F(Z,E)$ has an energy dependence as $E^{-1/2}$.⁽¹⁹⁾ The normalized beta energy distribution was thusly calculated to be

$$N(E)dE = 2.55 \times 10^{-5} E^{-1/2} (67-E)^2 dE \text{ (betas/keV)}.$$

The maximum energy of the beta spectrum is less than that required to penetrate the outer dead layers of skin (68 keV). Thus, external exposure from ^{63}Ni radiations do not contribute to the skin dose.⁽²⁰⁾

(12) C. M. Lederer, J. M. Hollander and I. Perlman, Table of Isotopes, Sixth Edition, John Wiley & Sons, New York, 1967.

(18) S. F. Mughabghab and D. I. Garber, Neutron Cross Sections, 3rd Edition, USAEC Report, BNL-325, June 1973.

(19) R. D. Evans, The Atomic Nucleus, McGraw-Hill Book Co., New York, NY, p. 548, 1955.

(20) U.S. NRC, Nuclear Regulatory Guide 1.109, Table E-6, p. 1.109-41, October 1977.

$^{93}_{42}\text{Mo}$

^{93}Mo is formed by neutron capture in 15.84% abundant ^{92}Mo in molybdenum. The neutron capture cross sections are very small: about 0.045 barns for thermal neutrons and a resonance capture integral of about 0.52 barns. ^{93}Mo can be formed in an excited state of 2.45 MeV which decays to the ground state with a 6.85 hour half life by gamma decay. The ground state of ^{93}Mo decays by electron capture (e.c.) with a rather poorly established half-life of 3,500 years. An estimated 96^{+4}_{-6} percent of the e.c. events proceed to an excited state of ^{93}Nb at an excitation energy of 30.4 keV. The e.c. to the 30.4 keV state is accompanied by an inner bremsstrahlung (IB) gamma ray spectrum with a maximum energy of 0.368 MeV. The IB spectra to the excited or ground states are of the same shape. The yield and spectral shape of the IB are not amenable to definitive theoretical calculation for the type of transition involved and have not been measured experimentally. The IB yield is the order of 3.2×10^{-4} gammas per e.c. event and the relative yield of the higher energy end of the IB spectrum is expected to be about half of that which would be given by an allowed spectrum. The 30.4 keV state in ^{93}Nb decays with a half-life of 13.6 years. Unless the niobium is chemically separated from the parent ^{93}Mo isotope, the apparent decay life will be that of the ^{93}Mo parent at long times ($\gg 13.6$ yrs) after neutron irradiation. The 30.4 keV decay gamma rays are essentially totally internally converted and none have been observed experimentally as expected. The radiation products per decay of the 30.4 keV state are primarily 10.4 keV Auger electrons and 16.6 niobium x-rays, approximately 0.8 of each per decay. These radiations will dominate the radiation dose of the 30.4 keV state decay.

$^{94}_{41}\text{Nb}$

^{94}Nb is formed by thermal neutron capture in 100% abundant ^{93}Nb . The thermal neutron capture cross section is about 1.2 barns. Neutron capture leads either to the ground state of ^{94}Nb or to an isomeric state which is 41 keV above the ground state. The isomeric state decays with a half life of 6.26 minutes with 99.5% of the decays going to the ground state. Thus,

for long-time radiation dose rate considerations, all neutron captures lead to the ground state. ^{94}Nb decays by β^- emission with a maximum beta energy of 471 keV and a half life of 20,300 years. All of the decays go to a single excited state of ^{94}Mo at 1574 keV excitation. All of the decays of this state go through an excited state at 871 keV. Thus, each ^{94}Nb decay yields two gamma rays with energies of 703 and 871 keV. These two gamma rays will dominate the dose rate due to the beta, other electrons, x-rays, and inner bremsstrahlung emissions.

C.3 CALCULATION OF RADIATION DOSE RATES FROM THE PRINCIPAL ACTIVATION PRODUCTS

The radiative transport of the monoenergetic gamma rays from ^{60}Co and the inner bremsstrahlung gamma spectra from ^{59}Ni and ^{55}Fe orbital electron capture were calculated using the ANISN⁽⁵⁾ code, modeled in the physical geometry of the reactor vessel and internals. Similar calculations were performed using the PUSHLD⁽²¹⁾ code, in slab geometry, and the results transformed to cylindrical geometry, using the standard methods given by Rockwell⁽²²⁾ to check on possible differences due to method and built-in features of the codes such as attenuation factors and buildup factors. Both codes produce as an output the dose rate to biological tissue at a distance of one centimeter from the surface of the activated material.

The transport dose rate calculations for the low energy x-rays were made using PUSHLD, with appropriate corrections for the fact that the actual source medium is stainless steel or carbon steel, not iron as is built into the PUSHLD code.

(5) W. E. Engle, Jr., A Users Manual for ANISN, A One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering. K-1693, Oak Ridge National Laboratory, Oak Ridge, TN, March 1967.

(21) J. N. Strode and H. H. Van Tuyl, PUSHLD - A Code for Calculation of Gamma Dose Rates from Plutonium in Various Geometries. USAEC Report HEDL-TME, pp. 73-89, 1973.

(22) Theodore Rockwell III, Reactor Shielding Design Manual. D. Van Nostrand Co., Inc., Princeton, NJ, 1956.

The computed dose rates from the activated PWR reactor components at the time of shutdown are given in Table C.3-1. These dose rates include only those radiations which contribute significantly to the dose to biological tissue. As stated earlier, ^{63}Ni does not contribute to the biological dose since the energies of the beta-rays from ^{63}Ni are insufficient to penetrate the surface dead skin layers of the human body. The external radiation dose rates from ^{14}C and ^{93}Mo are not significant in comparison with those listed in Table C.3-1.

A comparison of the time-dependent levels of radioactivity and dose rate in the activated reactor components is given in Figure C.3-1. The total dose rate and the total number of curies per unit volume from the principal sources are normalized to unity at the time of reactor shutdown. The fractional contribution from each of the principal source radionuclides is also presented.

TABLE C.3-1. Computed Dose Rates From Activated Components^(a)

Component	Calculated Dose Rates in R Per Hour			
	^{60}Co (Gamma)	^{94}Nb (Gamma)	^{55}Fe (IB, Gamma)	^{59}Ni (IB, Gamma)
Shroud				
Inner Wall	$(1.9 \text{ to } 5.6) \times 10^5$	2.0×10^0	1.1×10^{-1}	9.1×10^{-2}
Outer Wall	$(1.0 \text{ to } 3.0) \times 10^5$	1.0×10^0	5.6×10^{-2}	4.7×10^{-2}
Core Barrel				
Inner Wall	$(2.6 \text{ to } 7.9) \times 10^4$	1.7×10^{-1}	5.7×10^{-3}	2.1×10^{-2}
Outer Wall	$(9.7 \text{ to } 2.9) \times 10^3$	5.9×10^{-2}	1.3×10^{-3}	6.4×10^{-3}
Pressure Vessel				
Inner Wall	$(2.3 \text{ to } 5.4) \times 10^2$	--(b)	3.4×10^{-4}	1.6×10^{-4}
Outer Wall	$(2.3 \text{ to } 5.3) \times 10^0$	--	1.6×10^{-5}	1.0×10^{-6}
Biological Shield				
Inner Liner	$(1.9 \text{ to } 3.7) \times 10^0$	--	2.0×10^{-5}	6.0×10^{-7}
Concrete (Max.)	$(0.3 \text{ to } 0.7) \times 10^0$	--	8.0×10^{-8}	--

(a) Radiation Dose rates computed at a distance of one centimeter from the surface of the activated material, at the vertical center line of the reactor core.

(b) A dash means that radionuclide was not present in that component.

NOTE: IB means inner bremsstrahlung

It can be seen that while the decay rate of ^{63}Ni controls the reduction of the number of curies present after the first ten years, the decay rate of ^{60}Co controls the reduction of the radiation dose rate for the first 70 to 80 years. After that time, the dose rate is dominated by ^{94}Nb . The radiation dose rate is the significant factor in decommissioning work, since it directly affects occupational radiation exposure and has a strong influence on work plans and methods.

FIGURE C.3-1. Time Dependence of Radioactivity Levels and Dose Rates in the Activated Reactor Components

C.4 ESTIMATES OF RADIATION DOSE RATES THROUGHOUT THE REFERENCE PWR

This section presents estimates of the anticipated radiation dose rates throughout the reference PWR at shutdown. Since radiation exposure of the decommissioning personnel is a matter for concern, these estimates are necessary to calculate occupational exposure. It must be recognized, however, that these radiation dose rate estimates, while based on actual field measurements for operational PWRs, are indeed just estimates. They could, in some instances, vary by as much as a factor of 10 depending upon the plant's operating history and the extent and success of prior decontamination programs.

Key Assumptions

The following assumptions are made concerning dose rate conditions throughout the plant:

- All dose rates are with the reactor vessel head off and in the head storage area (Operating Floor elevation)
- The fuel transfer canal and the refueling cavity are full of water (unless indicated otherwise)
- The plant is in a shutdown/refueling mode (no fuel movements are in progress, however)
- Dose rates around steam generators are highly dependent on the secondary side water level; i.e., generally high secondary water levels mean more shielding and thus lower dose rates
- Unless indicated otherwise (contact, general area, etc.) dose points are at one meter height for areas and where the dose rate is at a maximum for objects
- In some areas where pipes located near equipment dominated the dose rate levels in that general area, an attempt is made to specify this condition; i.e., the pipe dose rate is greater than that from other sources and is the dominant factor
- It should be noted that crud traps exist at valves, valve stations, and sharp bends in piping due to vertical legs and represent potential high dose rate areas that have by no means been investigated within the scope of this study.

C.4.1 Estimated Dose Levels At Shutdown

Shutdown radiation dose rates and contamination levels were obtained from one 2-loop and five 3-loop operational PWRs. The plants had histories ranging from 3 to 6 years of commercial operation. Actual data on the most significant sources of radiation and the corresponding dose rates and/or contamination levels were provided by the Health Physics organizations at four sites.⁽²³⁻²⁶⁾ Specific area radiation levels generally did not vary greatly from plant to plant. Therefore, all data are presented as composite values, within fairly wide ranges in some cases but representative of their corresponding sources. The data are presented in Figures C.4-1 through C.4-5. The locations of the source references are indicated on the left side of the figures with area designations as follows:

<u>Measurement Point Prefix</u>	<u>Corresponding Elevation, m (ft)</u>
1 A - xx	13.72 (45)
2 A - xx	18.59 (61)
3 A - xx	23.47 (77)
4 A - xx	28.35 (93) Operating Level
5 A - xx	1.52 (5)
6 A - xx	7.62 (25)

Measured radiation dose rates for a "typical" steam generator are shown in Figure C.4-6.

(23) S. Crocker, Carolina Power and Light Company, H. B. Robinson No. 2, October 1976, Personal Communication.

(24) H. L. Story, Florida Power and Light Company, Turkey Point 2 & 3, October 1976, Personal Communication.

(25) R. C. Mecredy, Rochester Gas and Electric Company, R. E. Ginna, October 1976, Personal Communication.

(26) W. W. Cameron, Virginia Electric Power Company, Surrey 1 & 2, October 1976, Personal Communication.

• Dose Rates and Contamination Levels
by Source Location

Dose Point	Dose Rate, R/hr	gm/100 cm ²	Source Reference
1A-01	12 - 30		Reactor Coolant Pump Bowl (contact)(b)
1A-02	0.5 - 0.6		RCS Piping, cold leg (contact)
1A-03	0.05 - 0.4		Steam Generators (general area)(c)
1A-04	0.001 - 0.012		Emergency Personnel Lock
1A-05	0.1 - 0.6	>100 K	Floor Drains (contact)
1A-06	0.025 - 0.5	>100 K	Recirculating Sump (general area)
1A-07	0.01 - 0.15	1 - 32 g(a)	General Area (typical)
1A-08	0.025 - 0.1		RCS Pumps (general area)
1A-09	0.35 - 0.8		Steam Generator Loop to RCS Pump (typical)
1A-10	0.075 - 0.3	>500 K	Pressurizer Relief Tank (contact)
1A-11	0.035 - 0.4		General Area
1A-12	18 - 22		Steam Generator at the Manway (typical)
1A-13	0.5 - 0.7		Steam Generator at 0.9 m (typical)
1A-14	<0.3		Steam Generator at 1.8 m (typical)
1A-15	<0.2		General Area (typical)
1A-16	0.05 - 0.15		General Area (typical)
1A-17	<0.4		Pressurizer Spray Line
1A-18	<0.4		Piping (systematic)
1A-19	0.15 - 3	>100 K	Sump, Reactor Building
1A-20	<0.5		Reactor Coolant Drain Tank
1A-21	<0.1		Pressurizer Area (General Field)
1A-22	<0.2		Excess Letdown Hx
1A-23	0.1 - 0.3		Laundry Room
1A-24	<0.005		Radiation Sampling Room
1A-25	<0.06	<20 K	Radiological Chemistry Laboratory
1A-26	<0.007	<100	Waste Holdup Tank Room (typical)
1A-27	2 - 5	<300	Component Cooling Water Hx Area
1A-28	0.07 - 0.14	<100	Gas Stripper Feed Pumps Area
1A-29	>0.025		Component Cooling Water Pumps
1A-30	<0.15		Holdup Tanks Pump Drain
1A-31	0.5 - 1		Waste Tank Room
1A-32	0.2 - 0.4		Spent Fuel Pool Pumps
1A-33	<0.05		Service Water Booster Pumps Area
1A-34	<0.005		Treated Waste Monitor Tanks (contact)
1A-35	0.01 - 0.3		Personnel Hatch (outside C 1)
1A-36	<0.01		

(a) 1-32 K = 1,000 to 32,000 dpm/100 cm², 0.01 - 0.15 = a range from 0.01 to 0.15 R/hr
(b) Contact means as close as the measuring instrument can be conveniently placed next to the source (i.e. a surface dose rate)
(c) General Area refers to the radiation field in a room or area, not specifically from one specified source or direction, although a specific source or object may be the principal contributor to the General Area level measurement

FIGURE C.4-1. Radiation Source
Locations - 13.72 m
Elevation

• Dose Rates and Contamination Levels
by Source Location

Dose Point	Dose Rate, R/hr	Dose/100 cm ²	Source Reference
2A-01	>300		Upper Internals
2A-02	1 - 15		Regenerative He (contact)(b)
2A-03	0.05 - 0.25		Regenerative He, Laboratory
2A-04	<0.025		General Area (typical)
2A-05	<0.2		Pressure - (contact)
2A-06	<0.2		HC Pump (in duct)
2A-07	0.1 - 0.9		General Area (typical)(c)
2A-08	0.01 - 0.2		Steam Generator Area
2A-09	<0.1		Spent Fuel Ho
2A-10	0.2 - 1.5	500 - 5000 ^(a)	Drumming Room
2A-11	<0.02		Water Gas Decay Tank Room
2A-12	0.05 - 0.15	100 - 1000	Pipework
2A-13	<0.4		Heats Storage Tank Area
2A-13A	0.4 - 0.5		Heats Storage Tank (Fuel Pile is resting contact)
2A-13B	100 - 200		Heats Storage Tank (DCS stand bed resin contact)
2A-14	<0.505		Volume Control Tank (through the wall)
2A-15	1 - 3	100 - 1000	Volume Control Tank Area
2A-16	<0.01		Valve Gallery Area
2A-17	<0.02		Drumming Room Entrance
2A-18	<0.2		CHCS Monitor Tanks (floor drains)
2A-19	<0.3		CHCS Monitor Tanks (contact)
2A-20	0.25 - 0.5	10 - 300 c	Heats Evaporator Room
2A-21	<0.805		Elevator Entrance
2A-22	<0.2		Radwaste Evaporator Feed Tank (contact)
2A-23	0.05 - 0.8		Waste Gas Compressors Room
2A-24	2 - 4		Radwaste Evaporator Feed Tank (bottom)
2A-25	0.08 - 0.1		Radwaste Evaporator Feed Lines
2A-26	1 - 10		Letdown He (contact)
2A-27	2 - 7		Letdown Line (contact)

(a) $100 \text{ cm}^2 = 1000 \text{ to } 5000 \text{ dose/100 cm}^2$, $0.2 - 1.5$ is a range from 0.2 to 1.5 R/hr
 (b) 100 cm^2 means as close as the measuring instrument can be conveniently placed next to the source (i.e. a surface dose rate)
 (c) 100 cm^2 area refers to the radiation field in a room or area, not specifically from one discrete source or direction, although a specific source or object may be the principal contributor to the General Area Level measurement

FIGURE C.4-2. Radiation Source Locations - 18.59 m (61 ft) Elevation

* Dose Rates and Contamination Levels
by Source Location

Dose Points	Dose Rate, R/hr	dpw/100 cm ²	Source Reference
AA-01	0.1 - 1.0	27000 K	Reactor Cavity (inside edge)
AA-02	> 15		Preturbizer (top)
AA-03	> 2	20 - 200 K ^(a)	Steam Generators (general area) ^(b)
AA-04	0.005 - 0.05 ^(a)		General Area
AA-05	> 10		Storage Area with Reactor Vessel (RV) Head
AA-06	0.005 - 0.05		Equipment Hatch
AA-07	> 2		Above RV on Cooling Duct (general area)
AA-08	0.02 - 0.1	100 - 500	Controlled Access Machine Shop (general area)
AA-09	> 6		RCS Piping
AA-10	> 0.01		General Area
AA-11	> 0.02		Water Sample Laboratory
AA-12	> 0.025		Spent Fuel Pit (general area)
AA-13	> 0.005		HEPA Exhaust Filters

^(a) 20-200 K = 20,000 to 200,000 dpw/100 cm²; 0.005 - 0.05 = a range of from 0.005 to 0.05 R/hr

^(b) General Area refers to the radiation field in a room or area, not specifically from one source or direction, although a specific source or object may be the principal contributor to the General Area level measurement

FIGURE C.4-4. Radiation Source Locations - Operating Floor and Above

• Dose Rates and Contamination Levels

by Source Location:

Dose Point	Dose Rate, R/hr	dpm/100 cm ²	Source Reference
6A-01	>0.01		Boron Injection Tank Room
6A-02	<0.01		Laundry Drain Tank Pump Room
6A-03	0.01 - 0.05		RHR Hx Piping (through wall)
6A-04	>0.05		NaOH Tank (contact) ^(b)
6A-05	<1.2	15 - 30 K ^(a)	Charging Pump Room
6A-06	0.04 - 0.05 ^(a)		Centrifugal Charging Pumps (contact)
5A-01	0.8 - 17	30 - 1000 K	Residual Heat Removal Hx Room
5A-02	1 - 10	>750 K	Auxiliary Building Sumps
5A-03	0.05 - 0.3	4 - 200 K	RHR Pumps (contact)
5A-04	0.005 - 0.015	<1000	Safety Injection Pump rooms
5A-05	1 - 10		Clean Waste Receiver Tanks

(a) 15-30 K = 15,000 to 30,000 dpm/100 cm²; 0.04 - 0.05 = a range of from 0.04 to 0.05 R/hr.

(b) Contact means as close as the measuring instrument can be conveniently placed next to the source (i.e. a surface dose rate).

FIGURE C.4-5. Radiation Source Locations in Auxiliary Building at Plan Elevations 7.62 m and 1.52 m (25 ft and 5 ft)

SOURCE POINT LOCATION

- 1 - IN MANWAY
- 2, 3, 4, 5 - WAIST HIGH IN CENTER OF AND NEXT TO PERFORATED PLATES
- 6, 7 - EDGE OF DECK PLATE (ONE FOOT ABOVE)
- 8, 9, 10, 11 - AT FEEDWATER RING
- 12, 13, 14, 15 - AT FLOW RESISTANCE PLATE
- 16, 17 - AT HAND HOLE
- 18 - TUBE SHEET
- 19 - TUBE SHEET
- 20 - HOT LEG
- 21 - COLD LEG
- 22 - MANWAY
- 23 - MANWAY
- 24 - WORK PLATFORM

- NOTE
- POINTS IN () ARE LOCATED 180° OPPOSITE THOSE SHOWN
 - STEAM GENERATOR WATER LEVEL AT ZERO IN PRIMARY AND AT APPROXIMATELY 47% IN THE SECONDARY SIDE

SOURCE POINT NO.	RADIATION LEVEL R/hr
1	0.05
2	0.2
3	0.2
4	0.2
5	
6	0.5
7	
8	1
9	2
10	2
11	
12	3.5
13	10.5
14	10.5
15	
16	10
17	10
18	30
19	37
20	22
21	30
22	18
23	22
24	1.2

FIGURE C.4-6. Steam Generator Radiation Level Data

C.4-2. Estimated Dose Rates From Internally Contaminated Piping

Calculations of internal contamination levels in the reactor coolant systems were made to estimate the potential for occupational exposure when the systems are cut up during dismantling. The impacts of the radioactive corrosion product species are not proportional to the weights of the species but rather, to the energies and intensities of the gamma rays emitted per disintegration. Thus, cobalt, with a small fraction of the total crud weight deposited, generally is the most offensive species.

The assumed makeup of the deposited material is given in Table C.4-1. The fractional abundance of each of the radionuclides at time of reactor shutdown is based on measurements of filterable crud from 13 operating PWRs.⁽²⁷⁾ The contribution to the surface dose rate from fission product deposition within the piping systems was considered to be insignificant, based on data⁽²⁸⁾ that show the deposition of ^{134}Cs and ^{137}Cs to be about a factor of 10,000 less than that of ^{60}Co .

A thin layer of activated corrosion products was assumed to be deposited on the inside surface of the system. These deposits produced the radiation dose rates measured at the exterior surfaces of the system components. Using normal shielding calculations, the strength of the deposited source was inferred from the measured dose rate at the outer surface of the component. The estimates for the large pipes (686 to 787 mm dia.) were obtained from hand calculations for a slab of steel and an empirical correction factor to convert from slab to cylindrical geometry. Attenuation and buildup factors were obtained from ANL-5800⁽²⁹⁾ and the Reactor Handbook⁽³⁰⁾, based on the spectra

(27) A. J. Kennedy, "PWR Crud Characterization and Transport Modeling." Proceedings of the System Contamination Workshop - EPRI, Atlanta, GA, March 1976.

(28) E. T. Chulick, Oconee Nuclear Station Unit One Cycle Two Refueling Shutdown Primary System Radiation Levels, EPRI NP-340, Table 6-2, pp. 6-25, April 1977.

(29) Reactor Physics Constants. ANL-5800, Section 8, 2nd Edition, July 1963.

(30) Reactor Handbook. Vol. III, Part B, "Shielding." Interscience Publishers, 1962.

TABLE C.4-1. Activated Corrosion Product Isotopes Used in Pipe Radiation Dose Calculations

Reaction	Half Life	Gamma Energy (MeV)	Number Per Disintegration	Fractional Dose At T = 0	Fractional Abundance of Radioactivity in The Coolant (27)	Usual Source
$^{54}\text{Fe} (n,p) ^{54}\text{Mn}$	310 d	0.8353	1.0	0.024	0.042	Carbon Steel
$^{58}\text{Ni} (n,p) ^{58}\text{Co}$	72 d	0.511	0.30	0.796	0.591	Stainless Steel, Inconel, Incoloy
		0.8105	0.99			
		0.865	0.014			
		1.675	0.0062			
$^{59}\text{Co} (n,\gamma) ^{60}\text{Co}$	5.3 y	1.173	0.99	0.18	0.082	All materials, but especially nickel alloys and cobalt based wear surfaces
		1.332	0.99			
		2.158	1.2×10^{-5}			
$^{50}\text{Cr} (n,\gamma) ^{51}\text{Cr}$	28 d	0.32	0.02	(a)	0.269	Stainless Steel
$^{58}\text{Fe} (n,\gamma) ^{59}\text{Fe}$	45 d	0.143	0.01	(a)	0.017	Carbon Steel
		0.192	0.03			
		1.095	0.56			
		1.295	0.44			

(a) These relatively short-lived activities have been neglected in the calculation of piping dose rates since they will have decayed to unimportant levels by the time actual dismantlement of reactor system begins.

(27) A. J. Kennedy, "PWR Crud Characterization and Transport Modeling." Proceedings of the System Contamination Workshop - EPRI, Atlanta, GA, March 1976.

and gamma source strengths obtained from the Table of Isotopes.⁽¹²⁾ The dose rate as a function of pipe wall thickness is shown in Figure C.4-7. This curve is normalized to an assumed radioactivity of 0.86 Ci/m^2 on the inside surface of a 737 mm. I.D. pipe that produced a dose rate of 600 mrem/hr on the outside surface of the pipe. The relative contributions of ^{58}Co and ^{60}Co to the total surface dose rate at reactor shutdown are also shown in Figure C.4-7. The pipe wall thicknesses for Typical Reactor Coolant System (RCS) pipe diameters are shown in Table C.4-2.

FIGURE C.4-7. Pipe Surface Dose Rates at Reactor Shutdown Versus Wall Thickness

The calculated dose rate as a function of radioactivity deposited on the inside surface of the pipe for different RCS pipe sizes is shown in Figure C.4-8. Because of the assumptions made in the calculations, the results for these pipes may not be applicable to other RCS pipe sizes unless the wall thickness is much smaller than the pipe diameter.

⁽¹²⁾ C. M. Lederer, J. M. Hollander and I. Perlman, Table of Isotopes. Sixth Edition, John Wiley & Sons, New York, 1967.

TABLE C.4-2. Typical RCS Pipe Diameters and Wall Thicknesses

<u>Inside Diameter</u>		<u>Wall Thickness</u>	
<u>mm</u>	<u>in.</u>	<u>mm</u>	<u>in.</u>
698	27.5	58.9	2.32
737	29	62.0	2.44
787	31	65.8	2.59

FIGURE C.4-8. Dose Rate on the Outside Surface of Large Pipe as a Function of Corrosion Product Radioactivity on the Inside Surface

Because of the greater effect of geometry on the measured dose rate from the smaller pipe sizes shown in Table C.4-3, the ANISN⁽⁵⁾ code and a standard

⁽⁵⁾ W. E. Engle, Jr., A Users Manual for ANISN, A One Dimensional Discrete Ordinates Transport Code With Anisotropic Scattering. K-1693, Oak Ridge National Laboratory, Oak Ridge, TN, March 1967.

16 group gamma cross-section set were used to calculate the dose rates from assumed levels of deposited radioactivity in the smaller pipe sizes. The calculated dose rates on the outside surfaces of small pipes as a function of deposited corrosion product radioactivity are shown in Figure C.4-9. The variations in the dose rate with distance from the outside surface of pipe for the different pipe sizes discussed in this section are shown in Figure C.4-10.

TABLE C.4-3. Typical Small Pipe Dimensions

Nominal Size, in.	Schedule	I.D.		Wall Thickness	
		mm	in.	mm	in.
2	160	42.85	1.687	8.738	0.344
6	160	131.75	5.187	18.263	0.719
10	140	222.25	8.750	25.400	1.000
18		444.50	17.5	25.400	1.000

FIGURE C.4-9. Dose Rate on the Outside Surface of Small Pipes as a Function of Corrosion Product Radioactivity on Inside Surface

FIGURE C.4-10. Relative Dose Rate as a Function of Distance from Pipe Outer Surface

The decay of dose rates from the principal activated corrosion products as a function of time are presented in Figure C.4-11, with the total dose rate normalized to unity at reactor shutdown. The fractional dose rates used are the same as shown in Table C.4-1. The curves on Figure C.4-11 roughly indicate the relative contribution to the dose rate by the different corrosion product isotopes and the total dose rate as a function of time after reactor shutdown. The fraction of the total dose rate due to ^{60}Co shown here may be lower than that which might be encountered in a plant which has operated for 30 to 40 years, but these values represent presently available information.

FIGURE C.4-11. Radioactive Decay of Activated Corrosion Products

C.4.3 Estimated Inventory Of Radionuclides Deposited on System Internal Surfaces

The activated corrosion product deposition in the steam generators was inferred from the measured dose rates at the external surfaces of the generators, using relationships developed in the Reactor Shielding Design Manual⁽²²⁾ for infinite cylinders filled with a uniformly distributed source. The model used in the calculation is shown in Figure C.4-12. The radiation dose rate at the external surface of a steam generator as a function of the level of radioactivity from internally deposited activated corrosion products is given in Figure C.4-13. Based on measured external radiation dose rates of 40 R/hr, the radioactivity level within the steam generator is about 0.23 Ci/m^2 . With an internal surface area of about $19,000 \text{ m}^2$ for all four steam generators, the estimated inventory of radioactive corrosion products in the steam generators is about 4,400 Ci.

FIGURE C.4-12. Model of Steam Generator Used in Estimating Inventory of Deposited Radionuclides

(22) Theodore Rockwell III, Reactor Shielding Design Manual, D. Van Nostrand Co., Inc., Princeton, NJ, 1956.

FIGURE C.4-13. Dose Rate on the Surface of the Steam Generator as a Function of Internal Corrosion Product Radioactivity

The surface depositions on the reactor vessel and internal components are assumed to be at the same concentrations as in the steam generators. With an approximate surface area of 570 m^2 , the inventory deposited on the reactor vessel and internals is about 130 Ci.

The inventory deposited on the system piping (except RCS piping) is estimated based on the quantities of piping of various sizes given in Table C.4-4. The three pipe sizes contributing the largest internal surface areas (51, 152 and 254 mm dia) were used in making the estimate, together with an assumed surface radiation dose rate of 400 mR/hr. An average internal deposition of about 0.06 Ci/m^2 for all piping except RCS piping was derived. With an internal surface area of about $1,100 \text{ m}^2$, the deposited inventory becomes about 60 Ci.

TABLE C.4-4. Auxiliary Systems-Contaminated Stainless Steel Piping, Nuclear Class Service^(a)

Nominal Size, in.	Schedule	Number of Lineal ft. (c)	Total Wt. lbs. (c)	Total Inside Surface Area, m ²
1/2	80	400	436	5.3117
	160	400	524	4.5140
3/4	40	400	452	8.0163
	80	600	882	10.8278
	160	1,900	3,686	28.2806
1	40	200	336	5.1026
	80	600	1,302	13.9652
	160	1,400	3,976	22.7504
1-1/2	40	400	1,088	15.6628
	80	1,100	3,993	40.1299
	160	1,800	8,748	58.5751
2	40	1,000	3,650	50.2718
	80	1,600	8,032	75.4540
	160	3,500	26,110	143.6043 ^(b)
3	160	460	6,583	29.3566
4	160	600	13,512	50.1697
6	160	1,020	46,246	128.6770 ^(b)
8	160	470	35,113	70.6775
10	140	630	65,602	134.0704 ^(b)
12	140	290	40,507	74.0579
14	140	330	56,173	92.2988
Totals				
		19,100 ft	326,951 lb.	1,061.7644
		5,822 m	148,302 kg	

^(a) Nuclear Power Piping Design Code, ANSI B31.7, Class I and II

^(b) Selected for calculations regarding internal corrosion product deposition

^(c) $0.3048 \text{ m} = 1 \text{ ft}$; $0.4535 \text{ kg} = 1 \text{ lb}$.

The deposited level of radioactivity in the pressurizer was assumed to be about 0.04 Ci/m^2 .⁽³¹⁾ With an internal surface area of about 87 m^2 , the inventory in the pressurizer is about 4 Ci.

The inventory in the RCS piping was estimated assuming surface radiation dose rates of 600 mR/hr, which, from Figure C.4-8, yields a deposition of about 0.86 Ci/m^2 . With an internal surface of about 190 m^2 , the inventory in the RCS piping is about 160 Ci.

The estimated inventory of deposited radioactive corrosion products is summarized in Table C.4-5.

TABLE C.4-5. Estimates of Quantities of Radioactive Corrosion Products Deposited on the Interior of Reactor Systems

<u>Systems</u>	<u>Surface (m^2)</u>	<u>Level of Radioactivity (Ci/m^2)</u>	<u>Total Ci</u>
Reactor Vessel and Internals	$\sim 5.7 \times 10^2$	~ 0.23	~ 130
Steam Generators	$\sim 1.9 \times 10^4$	~ 0.23	~ 4400
Pressurizer	$\sim 8.7 \times 10^1$	~ 0.04	~ 4
Piping (Except RCS)	$\sim 1.1 \times 10^3$	~ 0.06	~ 60
RCS Piping	$\sim 1.9 \times 10^2$	~ 0.86	~ 160
Totals	$\sim 219 \times 10^2$		~ 4800

⁽³¹⁾ Standardized Nuclear Power Plant System - SNUPPS, Preliminary Safety Analysis Report, Vol. 3, 1974.

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APPENDIX D

ALTERNATIVE METHODS FOR FINANCING DECOMMISSIONING

Several methods for providing the financing necessary to accomplish the decommissioning of a nuclear power reactor at the end of its life have been identified and examined to determine their relative advantages and disadvantages. Relationships have been developed to permit the determination of the present value cost of each of the alternative financing methods and these relationships have been evaluated for several sets of input parameters, to permit comparison of costs.

D.1. BASE ASSUMPTIONS AND DEFINITION OF TERMS

It is useful to identify the possible ownership arrangements for power reactors, since the cost of financing is related to the type of corporation involved. The ownership arrangements presently existing are: publicly-owned utilities such as the Washington Public Power Supply System, investor-owned utilities such as Commonwealth Edison, and state or federal government agencies or corporations such as the Power Authority of the State of New York or the Tennessee Valley Authority. Joint ownership by several utilities within a given category is quite common, and joint ownership by combinations of public and investor-owned utilities are becoming common. A review of the present ownership arrangements shows that about 81% of the nuclear power reactors operating or planned in the U.S. are investor-owned, while the remaining 19% are publicly-owned (including government corporations).⁽¹⁾ Thus, the financing costs should be weighted heavily toward those typical of investor-owned utilities.

In the developments which follow, the results are sensitive to the various costs of financing. It is, therefore, appropriate to define these various cost factors and other terms of interest.

⁽¹⁾ U.S. NRC, Owners of Nuclear Power Plants. NUREG-327, October 1977.

Interest Rate

The rate of return on capital invested in normal securities, i.e., bonds, certificates of deposit, and similar financial instruments.

Inflation Rate

The rate of increase in cost of goods and services, on an annual basis, as determined from the nation's economic indicators by the Federal Department of Labor.

Discount Rate

The rate of return on capital that could have been realized in alternative investments, if the money were not committed to the plan being evaluated, i.e., the opportunity cost of alternative investments. This cost is equivalent to the weighted average cost of capital.⁽²⁾ For an investor-owned corporation, the weighted average cost of capital should reflect the corporation's cost for debt and equity capital and the proportions of debt, equity, and retained earnings that are used for capital investments.⁽³⁾

Present Value of Money

When different business activities require disbursement of funds over different time frames, it is difficult to compare the actual cost of each activity to the sponsoring organization. One generally accepted method of placing these various disbursements on a common basis is to compute the value of those disbursements in terms of current dollars, i.e., the present value of money to be paid out or received at some time other than the present. For an investor, "the present value of a future payment or series of payments is the present investment necessary to secure the promise of that future payment or series of payments, with interest at a given rate."⁽⁴⁾

(2) R. W. Johnson, Capital Budgeting. Wadsworth Publishing Co., Inc., Belmont, CA, pp 49, 1970.

(3) W. G. Lewellen, The Cost of Capital. Wadsworth Publishing Co., Inc., Belmont, CA, 1969.

(4) E. L. Grant, W. G. Ireson and R. S. Leavenworth, Principles of Engineering Economy. 6th edition, The Ronald Press Co., New York, 1976.

D.2 FINANCING METHODS

The principal methods for funding decommissioning activities are:

(1) pay the costs when they are incurred, (2) collect the present value of the estimated future decommissioning costs before the facility begins to operate, a prepaid sinking fund, and (3) establish a sinking fund that will be increased by annual payments during a facility's expected life which can then be used to pay decommissioning costs. Various combinations of these methods could also be utilized.

D.2.1 Payment of Costs When Incurred

Delaying the commitment of funds for paying decommissioning costs until the work actually takes place has several potential disadvantages from the viewpoint of protecting the public safety and the public coffers. One problem may be that of getting the obligated party to actually perform the work and pay the costs. If legal proceedings became necessary to fix responsibility, many additional years and dollars could be expended in achieving that determination. Even if the owner's responsibility is established and accepted, the owner may not have sufficient financial strength to pay for the work. This problem is not likely to occur when the facility owner is a regulated utility since presumably the state public utility commission will consider decommissioning as a legitimate operating expense and allow appropriate rate adjustments to pay these expenses. Similarly, if the facility is owned by a governmental body, the necessary funds can be raised through that body's taxing authority.

An additional problem with the Pay When Incurred option is that it is inequitable to those future rate payers or taxpayers who will have to pay the decommissioning costs for a facility without having enjoyed the benefits of that facility's operation. While such intergenerational subsidies are not unknown in the U.S.,^(a) it is arguably more equitable to require that those who enjoy the products of the nuclear facilities also bear the decommissioning expenses.

^(a) One example of an intergenerational subsidy is the social security system. Certain persons who had little or no opportunity to earn work credits are still eligible for benefits. Workers who began their working career after the social security system was initiated are not eligible for this special benefit.

D.2.2 Prepayment of Costs

A second alternative to funding decommissioning expenses would be to require nuclear power reactor owners to pay the total estimated costs in current dollars into some type of trust account at the time of facility startup. The trustees of the account would manage the monies until the funds were needed.

Additional payments into or refunds from the trust account would be made annually as required to assure that the fund changes sufficiently to keep up with inflation and to cover changes in costs due to technological innovation and/or new regulatory requirements.

The Prepayment method will also require that decommissioning costs be considered when utilities are comparing the economics of coal versus nuclear power production. Under the Pay When Incurred method, the costs are so far in the future that they have little effect on the decision.^(a) Under the Annual Payment Sinking Fund, the costs occur throughout the facility's life-time and must be considered. Under the Prepayment method the costs are incurred as a lump sum at the beginning of facility life and can be considered as part of the capital cost of the facility. For a corporation using a return on investment strategy to make capital investment decisions, the initial cost required by this method will clearly make the return less attractive and may affect the decision of whether or not to build the facility.

The Prepayment method is somewhat inequitable because future power consumers may not pay their fair share of the decommissioning costs, depending on their time of consumption. If the facility owners pay the present value of the expected decommissioning costs out of retained earnings, the future consumers will only have to pay for adjustments to the fund such as new regulatory requirements. The facility owner is more likely to fund the cost from equity and long-term debt financing. In this case, future consumers will be ultimately charged through the pricing mechanism a sufficient amount to retire the interest and principal of the debt. The term of the debt financing

^(a) The present value of \$1 payable in 30 years with a discount rate of 10% is less than 6¢.

will probably be less than the facility life. Thus, during the period when the debt is being retired, the consumers may pay as much or possibly even more to fund decommissioning than consumers under the Annual Payment method. When the debt is retired, consumers would pay less.

D.2.3 Creation of an Annual Payment Sinking Fund

The third alternative for paying decommissioning costs is the establishment of an annual payment sinking fund to generate enough funds during the life of a facility to pay these anticipated costs. The sinking fund could be set up by estimating future costs in current dollars of the year of facility startup. An amount would be paid into the fund per unit of product so that the payments, plus compound interest earned by investing the fund during and after the plant's operating life, would be sufficient to pay all anticipated decommissioning expenses. The most convenient mechanism for collecting these funds would be by treating the decommissioning costs as a negative salvage value in the calculation of depreciation for the station. The decommissioning cost portion of the depreciation calculation could be adjusted annually to account for inflation, regulatory changes and technological advances. This approach is favored by the majority of public utility commissions who have considered the question.⁽⁵⁾

A variety of entities could be designated to provide stewardship for the collected funds. Possibilities include state government, the federal government, a private organization such as a bank, or the utility itself. An independent "Decommissioning Assurance Agency" could also be chartered by each state or by the federal government to retain and invest the sinking fund and perhaps oversee activities and disperse payments to those conducting the activities. Such an agency would act in a fiduciary capacity for the public. Its governing board might be composed of representatives of the public, government, power-consuming industries, and power-producing industries.

⁽⁵⁾ Docket PRM 50-22, Letters to Robert G. Ryan, Director, Office of State Programs, U.S. NRC, in response to his letter of inquiry to State Public Utility Commissions.

By including various interest groups, tendencies to overestimate or underestimate costs and the annual payments needed to fund the costs should be minimized.

An approach supported by at least one public utility commission⁽⁶⁾ is for the utility to invest the funds collected for decommissioning in new facilities in the utility system. When money is needed for decommissioning, the utility would issue securities against those unencumbered facilities. In this way, the funds are placed in a secure investment until needed, thus providing an effective rate of return on the funds equivalent to the utility's cost of borrowing, and the utility does not have to borrow additional money to build the needed facilities.

The sinking fund approach is equitable. If cost escalations are factored in, all consumers should pay their approximately proportional share of costs in constant dollars. Exact sharing of costs would be virtually impossible because of changes in the operational life of facilities and changes in expected decommissioning cost caused by technological innovation and/or new regulatory requirements.

D.3 COMPARISON OF COSTS

One factor affecting the relative desirability of the three methods that has not yet been discussed is cost. Although each method ultimately provides sufficient funds for decommissioning, the funds are generated over different time periods. Consequently, the total cost of each of the methods discounted to present value is different. Meaningful comparison of the present value costs is complex because assumptions regarding the appropriate discount rate, the rate of cost escalation, and the rate of interest obtainable from invested funds must be made. Nevertheless, reasonable assumptions can be made and the cost compared.

⁽⁶⁾ Docket PRM 50-22, Letter to Robert G. Ryan, Director, Office of State Programs, U.S. NRC, from Charles A. Zielinshi, Acting Chairman, Public Service Commission, State of New York, January 9, 1978.

D.3.1 Derivations of Relationships for Cost Comparisons

The goal of each of the three methods to providing financing for decommissioning is to have the appropriate amount of money available when it is needed for decommissioning. In this section, the expressions used to compare the costs of the different methods are derived.

Let i be the available interest rate/interest period, j be the inflation rate over the same time period, k be the discount rate over the same time period, n be the number of time periods considered, P be the present value of a sum of money, and S be the sum of money which is equivalent to P when P is compounded over n time periods with an interest rate i .

D.3.1.1 Payment of Decommissioning Costs After Station Shutdown (Pay When Incurred)

The present value of a sum of money, S , to be paid n years later is given by:

$$P = \frac{S}{(1+k)^n}$$

If the amount S (the estimated decommissioning cost) is escalated by inflation, the present value of S becomes:

$$P_I = \frac{S(1+j)^n}{(1+k)^n}$$

D.3.1.2 Payment of Estimated Decommissioning Costs into a Trust Fund at Station Startup (Prepaid Sinking Fund)

In this case the full amount of the estimated costs, S , is deposited into a trust account when the facility begins operation. After one year the amount in the account will be $S(1+i)$ and the amount required to meet the estimated costs will be $S(1+j)$. Thus, the difference between these amounts, $S(j-i)$, is refunded if $i > j$ or an additional payment is required

if $j > i$. The amount present after the adjustment is made is just $S(1+j)$ at the end of year one. The present value of the trust account is given by:

$$P_1 = S + \frac{S(j-i)}{(1+k)}$$

After the second year, the amount present after adjustments is $S(1+j)^2$, and the adjustment for that year is $S(1+j)(j-i)$. The present value of the adjusted account is:

$$P_2 = S + \frac{S(j-i)}{(1+k)} + \frac{S(1+j)(j-i)}{(1+k)^2}$$

Extension of this series to n years results in the amount present after n years being $S(1+j)^n$ as required, the adjustment for that year is $S(1+j)^{n-1}(j-i)$, and the present value of the adjusted account is

$$S + \frac{S(j-i)}{(1+k)} + \frac{S(1+j)(j-i)}{(1+k)^2} + \dots + \frac{S(1+j)^{n-1}(j-i)}{(1+k)^n}$$

or

$$P_{II} = S \left\{ 1 + (j-i) \sum_{a=1}^n \frac{(1+j)^{a-1}}{(1+k)^a} \right\}$$

When $i=j$, this expression reduces to just S . When $i > j$, P is less than S due to the refunds which are appropriately discounted, and when $j > i$, P is greater than S due to the discounted additional payments which are required. The formulation above does not include the cost of servicing and repaying the debt incurred to establish the fund initially.

D.3.1.3 Creation of Sinking Fund with Annual Payments (Annual Payment Sinking Fund)

In this case, the estimated costs, S , escalated for inflation are paid by the creation of a sinking fund that receives payments each year of facility operation. At the end of the first year of the plant's life an amount

R_0 is paid into the sinking fund. No interest is earned during the first year. At the end of the second year of operation $R_0(1+j)$ is paid into the fund. During the second year $(i)(R_0)$ interest is earned on the invested fund. Thus, at the end of the second year the fund has a total of $R_0 + R_0(1+j) + iR_0$ dollars.

A closed form expression for the amount of money in the fund can be derived by considering each year's payment as a separate sinking fund. Thus, the first year's payment will be $R_0(1+i)$ at the end of year two, $R_0(1+i)(1+i)$ at the end of year three, and so on. At the end of the thirtieth year of operations it will total $R_0(1+i)^{n-1}$, or $R_0(1+i)^{29}$, dollars. Similarly the second year payment $R_0(1+j)$ will earn compounded interest for 28 years and will ultimately total $R_0(1+j)(1+i)^{n-2}$, or $R_0(1+j)(1+i)^{28}$. The total value of the sinking fund at the end of year n must be equal to the anticipated decommissioning costs in dollars of the year of shutdown, as given by:

$$S(1+j)^n = R_0 \sum_{a=1}^n (1+j)^{a-1} (1+i)^{n-a}$$

The initial payment is:

$$R_0 = \left[\frac{S(1+j)^n}{\sum_{a=1}^n (1+j)^{a-1} (1+i)^{n-a}} \right],$$

and the b^{th} annual payment will be:

$$R_b = R_0 (1+j)^{b-1}.$$

Thus, the present value of the series of payments is:

$$P_{III} = R_0 \sum_{b=1}^n \frac{(1+j)^{b-1}}{(1+k)^b} = \left[\frac{S(1+j)^n}{\sum_{a=1}^n (1+j)^{a-1} (1+i)^{n-a}} \right] \sum_{b=1}^n \frac{(1+j)^{b-1}}{(1+k)^b}$$

D.3.2 Rationale for Selection of Interest and Discount Rates Used in Cost Comparison Calculations

A reasonable argument can be made that, for purposes of comparing the three methods, the appropriate discount rate to use is between 6% and 10%, and is probably near the high end of this range. The specific rate depends on whether the facility owner is an investor-owned utility, or a publicly-owned utility.

For investor-owned utilities, the cost of capital can be approximated by the overall rate of return allowed by the state utility regulatory commission,⁽⁷⁾ currently in the 8-10% range. For example, Minnesota Power and Light was recently allowed a 9.51% overall rate of return,⁽⁸⁾ Pacific Power and Light 9.17%,⁽⁹⁾ and Montana-Dakotas Utilities 8.66%.⁽¹⁰⁾

Publicly-owned utilities have no equity financing and their cost of capital can be approximated by using the interest rates in their long-term bonds. These rates are generally lower than can be obtained on investor-owned utility bonds since the interest payments are not subject to Federal income taxes. For example, Washington Public Power Supply System AAA bonds, due in the year 2012, pay 5.70% interest.

Since about 81% of all power reactors presently operating or planned in the U.S. are owned by investor-owned utilities, the discount rate should be generally in the 8-10% range. Interest rates obtainable from stable, secure investments are generally several percentage points less than the current discount rates, reflecting the lower risk involved, or in the 6-8% range. While there is little historical basis for projecting inflation

⁽⁷⁾ E. L. Grant, W. G. Ireson, and R. S. Leavenworth, Principles of Engineering Economy. 6th Edition, the Ronald Press Co., New York, pp. 487, 1976.

⁽⁸⁾ Electrical Week, McGraw-Hill, p. 4, January 3, 1977.

⁽⁹⁾ Electrical Week, McGraw-Hill, p. 7, January 17, 1977.

⁽¹⁰⁾ Electrical Week, McGraw-Hill, p. 9, May 23, 1977.

rates over long periods of time, there are data which indicate that the difference between the discount rate and the inflation rate is relatively constant, at about 4%, independent of the current rate of inflation. Thus, for purposes of this study, a discount rate of 10%, an interest rate of 8%, and an inflation rate of 6% have been selected as the base condition. The effect of varying these parameters is also examined.

D.3.3 Calculation of Present Value Costs for Sinking Funds

Table D.3-1 contains comparative present value cost information for the three methods under various assumptions. Annual compounding of interest is used in all cases. All cases are based on estimated total decommissioning cost (1978 dollars of \$44 million and a 40-year plant life. If the estimated cost in current dollars is S , the expected interest rate obtainable on invested funds is i , the anticipated rate of inflation is j , and the discount rate (the rate of return that could have been realized from alternative investments) is k , then for the Pay When Incurred method, the present value P of the monies to be expended in n years hence will be given by:

$$P_I = \frac{S(1+j)^n}{(1+k)^n}, \text{ as developed in Section D.3.1.1.}$$

For the Prepayment method, the present value of the initial payment, with annual adjustments to compensate for surpluses or deficits due to differences in interest and inflation rates, is:

$$P_{II} = S \left\{ 1 + (j-i) \sum_{a=1}^n \frac{(1+j)^{a-1}}{(1+k)^a} \right\},$$

as developed in Section D.3.1.2.

For the Annual Payment method, the present value of the sinking fund is:

$$P_{III} = \left\{ \frac{S(1+j)^n}{\sum_{a=1}^n (1+j)^{a-1} (1+i)^{n-a}} \right\} \sum_{b=1}^n \frac{(1+j)^{b-1}}{(1+k)^b},$$

as developed in Section D.3.1.3.

The costs of administering the trust accounts for the second and third methods have been neglected in computing the present values given in Table D.3-1.

TABLE D.3-1. Comparison of Present Value Cost for the Three Financing Methods, for a Range of Financial Parameters

Financial Parameters (b)	Financing Method (c)	Present Value Costs (Millions \$) (a)			
		Discount Rate			
		6%	8%	10%	12%
i = 4	I	9.445	4.472	2.147	1.044
j = 2	II	26.723	30.824	33.537	35.409
k = 6,8,10,12	III	14.712	11.220	8.910	7.316
i = 6	I		9.724	4.667	2.270
j = 4	II		26.862	30.889	33.568
k = 8,10,12	III		15.002	11.477	9.132
i = 8	I			10.000	4.864
j = 6	II			27.000	30.955
k = 10,12	III			15.287	11.730
i = 10	I				10.273
j = 8	II				27.136
k = 12	III				15.567

(a) S = \$44 million, n = 40 years

(b) i = interest rate, j = inflation rate, k = discount rate

(c) I = Pay When Incurred, II = Prepaid Sinking Fund. III = Annual Payment Sinking Fund.

D.3.4 Calculation of Annual Cash Flow for Each Financing Method

The size of the annual payments or adjustments required for each of the three financing methods has been calculated for one set of financial parameters. These payments, at five points in time are given in Table D.3-2, together with the cumulative compounded fund total, in undiscounted dollars.

TABLE D.3- . Annual Cash Flow for Funding
Decommissioning (Millions \$)^(a)

<u>Year</u>	<u>Financing Method^(b)</u>	<u>Cash Flow in Year Shown</u>
1	I	0.0
	II	-0.880
	III	0.791
10	I	0.0
	II	-1.487
	III	1.337
20	I	0.0
	II	-2.663
	III	2.394
30	I	0.0
	II	-4.768
	III	4.288
40	I	452.572
	II	-8.539
	III	7.678

Final Fund Total = \$452.572 million in
undiscounted dollars.

^(a) Interest rate = 8%, inflation rate = 6%

^(b) I = Pay When Incurred, II = Prepaid Sinking
Fund, III = Annual Payment Sinking Fund.

D.4 FACTORS THAT INFLUENCE THE CHOICE OF A FINANCING METHOD

Three methods for financing future decommissioning costs have been discussed in this section: (I) Payment of Costs as they are Incurred, (II) Prepayment of the Present Value of Anticipated Costs, and (III) Creation of an Annual Payment Sinking Fund to Pay the Anticipated Costs. Some criteria that might be used to rank the desirability of these methods are:

- The extent to which decommissioning is financially assured,
- The present value cost of the method,
- The extent to which the beneficiaries of the nuclear reactor facility pay its decommissioning costs,
- The extent to which the method provides for consideration of decommissioning costs when selecting among alternative power generation options, and
- The ease of administering the method.

Some discussion of these criteria is appropriate.

The situation of a utility as a regulated monopoly that is assured of a reasonable return on investment significantly reduces any concern about its financial ability to pay for decommissioning. Thus, criteria I is eventually satisfied regardless of the method of financing selected. Should a station be forced to terminate production prematurely, the utility is forced into the Pay When Incurred mode if it was using the Annual Payment Sinking Fund approach.

The characteristics of the calculation of the present value of a future expenditure makes the Pay When Incurred approach look desirable since the commitment of funds is delayed as long as possible. For the same reason, the Prepaid Sinking Fund is the most expensive since the money must be committed at the beginning of station operation.

Equity in the distribution of decommissioning costs to the consumers of the station's electricity is an important consideration. Assuming the station operates for its projected lifetime, the Annual Payment Sinking Fund

is the most equitable, followed closely by the Prepaid Sinking Fund. The Pay When Incurred approach clearly places the costs on future electricity consumers, not on the consumers of the electricity from the station under consideration.

When decommissioning funds have to be committed at the start of operations as in the Prepaid Sinking Fund, these funds are essentially a part of the capital cost of the station and clearly will be considered when selecting between alternative power generation options. Using the Annual Payment Sinking Fund approach, these funds become a part of the operating cost and again will be considered in selecting alternatives. However, when the commitment of funds is 30 to 40 years in the future, its impact on consideration of alternatives will be rather minimal.

Administration of the Pay When Incurred approach is clearly the easiest since there are no funds to collect or administer until decommissioning takes place. The Annual Payment Sinking fund is the next easiest to administer since the funds come from operating revenue and are handled by normal accounting procedures. The efforts associated with the investment of the money in both the Annual Payment and Prepaid Sinking funds are approximately equivalent. The Prepaid Sinking fund has the additional complication of having to issue securities to obtain the initial funds.

In the final analysis, unless otherwise dictated by federal regulations, the approach that will be selected will be the one most favored by the public utility commissions since they control the ways that the funds can be collected from the electricity consumer. On that basis, it is likely that the Annual Payment Sinking fund approach will be the most prevalent, with some variations from state to state in how the investment of these funds is handled.

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APPENDIX E

RADIATION DOSE METHODOLOGY

Radiation dose calculations were performed for Sections 8 and 11 of this report using the radionuclide inventories in the reference PWR facility and on its site given in Section 7. The calculated doses were used to estimate the acceptable contamination levels for public use of the decommissioned reference PWR, and to estimate the safety impacts of decommissioning operations. This appendix contains details of the assumptions, models, and parameters used in the radiation dose calculations.

The following definitions and terminology apply for the airborne release of radionuclides:

- Chronic Release - An airborne release that occurs at a constant rate for a year.
- Acute Release - An airborne release of short duration, usually 0 to 8 hours. The accident analyses are based on acute releases.
- Organs of Reference - The organs for which radiation doses are calculated are the thyroid, lungs, total body, and bone.
- Exposure Pathways - The radiation exposure pathways of concern to personnel in the facility are inhalation, air submersion, and external exposure from surface contamination. In the environment, the same pathways are considered, plus the ingestion of contaminated food products.
- Maximum Exposed Individual - The individual who receives the maximum radiation exposure to any of his organs of reference. The maximum exposed individual is assumed to reside at the location of the highest airborne radionuclide concentration, and maximized exposure pathway parameters are used.
- Population Doses - Population doses use average individual parameters and the total population and population distribution discussed in Section 7 and Appendix B.
- First Year Doses - The dose accumulated during the year of the release period is reported as the first year dose.

- Fifty Year Dose Commitment - The first year dose plus the dose accumulated for the next 49 years from the radionuclides that were deposited within the human body during the release period is reported as the fifty year dose commitment.
- Annual Doses - The dose calculated during any year following continuous exposure. It is the sum of the doses received during the year of interest from all pathways including the doses resulting in that same year from intake of radionuclides during the previous years.
- Maximum Annual Dose - The maximum annual dose is the largest of the fifty annual doses calculated for the fifty year period.

These definitions apply for all dose calculations performed for and reported in this study.

E.1 MODELS FOR ESTIMATING RADIATION DOSE TO MAN

The fundamental relationship for the calculation of radiation dose to man from environmental pathways is given as follows for any radionuclide:

$$R_{ipr} = C_{ip} U_p D_{ipr} \quad (E.1)$$

where:

- R_{ipr} = the radiation dose or dose commitment from nuclide i via pathway p to organ r
- C_{ip} = the concentration of nuclide i in the media of pathway p ;
for airborne radionuclides, the average airborne concentration is defined by $\bar{X}_i$, which replaces C_{ip} in this equation
- U_p = the exposure rate or intake rate associated with pathway p , and
- D_{ipr} = a radiation dose or dose commitment factor: a factor for a given nuclide i , pathway p and organ r that converts a given concentration of the radionuclide and the intake rate of that radionuclide to the radiation dose or dose commitment.

Specific equations tailored to each exposure pathway are derived from Equation E.1. The principal difference among pathways is the manner in

which the radionuclide concentrations in air, soil, or food products are calculated as an integral part of the computerized models used in this study.^(1,2) They are functions of such parameters as the radionuclide release rates, deposition rates, root uptake parameters, and atmospheric dispersion.

The usage parameters (U_p in Equation E.1) assign hours of external exposure and intake rates of contaminated foods to the radiation dose equation. Average adult usage parameters are the basis for the population radiation dose estimates.

Equations for calculating the internal radiation dose or dose commitment factors (D_{ipr} from Equation E.1) were derived from those given by the International Commission on Radiological Protection (ICRP) for body burdens and Maximum Permissible Concentrations (MPC) of each radionuclide.^(1,4,5,6) Internal dose factors for the adult not found in the literature were calculated in a manner similar to that discussed in Reference 6. Effective decay energies for the nuclides were calculated from the ICRP model, which assumes

- (1) J. K. Soldat, et al., Models and Computer Codes for Evaluating Environmental Radiation Doses. USAEC Report BNWL-1754, Pacific Northwest Laboratory, Richland, WA, February 1974.
- (2) D. A. Baker, G. R. Hoenes and J. K. Soldat, "FOOD - An Interactive Code to Calculate Internal Radiation Doses from Contaminated Food Products." in Environmental Modeling and Simulation. Proceedings of a conference held in Cincinnati, OH, April 20-22, 1976, pp. 204-208, EPA, Washington, DC, 1976.
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- (4) J. F. Fletcher, and W. L. Dotson (compilers), HERMES - Digital Computer Code for Estimating Regional Radiological Effects from the Nuclear Power Industry. USAEC Report HEDL-TME-71-169, Hanford Engineering Development Laboratory, Richland, WA, 1971.
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- (6) G. R. Hoenes, and J. K. Soldat, Age-Specific Radiation Doses Commitment Factors for a One Year Chronic Intake. NUREG-0172, a report for the U.S. Nuclear Regulatory Commission, to be published as a supplement to Regulatory Guide 1.109.

that all of a given radionuclide is in the center of a spherical organ with an appropriate effective radius. Radiation dose factors have units of mrem in the first year per pCi taken into the body during that year either via ingestion or inhalation. Radiation dose commitment factors have units of mrem/50-yr per pCi taken into the body in the first year.

The radiation dose rate factors for external exposure in air were derived using the basic assumption that the contaminated medium is large enough to be considered an "infinite volume" relative to the range of the emitted radiations. With this conservative assumption the energy emitted per gram of medium is equivalent to the energy absorbed per gram of medium. Conversion from MeV per disintegration per gram to rem is made by correcting for the difference in energy absorption between air and body tissue, for the quality factor of the radiation under consideration, and for the physical geometry of each specific exposure situation.

The radiation dose from submersion in air is an external dose either to the skin, or to both the skin and total body, depending on the penetrating power of the radiation emitted. Only beta and gamma radiation, which could penetrate 7 mg/cm^2 of tissue, were considered in calculating skin dose. Gamma radiation dose at a 5-cm depth in tissue was used for calculating external dose to the total body (and for internal organs). These radiation dose factors have units of mrem/hr per pCi/m^3 of air.

Contaminated ground resulting from deposition of airborne radionuclides was represented by a large, uniform, thin plane source of external exposure. The factors for converting surface contamination in pCi/m^2 to gamma dose, one meter above a uniformly contaminated plane have been described.^(1,5) Dose rate factors for exposure to contaminated soil have units of mrem/hr per pCi/m^2 of surface.

For the external exposure pathways associated with air submersion and ground contamination, the 50-year dose commitment is equal to the first year dose.

E.1.1 Resuspension Factors

Estimates of the airborne radionuclide concentrations from resuspended surface or ground contamination are necessary to calculate the maximum annual doses. These, in turn, are used to calculate the acceptable contamination levels for public use of the decommissioned reference PWR as given in Section 8. The resuspension factor, S_f , is related to the airborne radionuclide concentration, $\bar{X}$, by Equation E.2.⁽⁷⁾

$$S_f = \frac{\bar{X}}{S_A} \quad (E.2)$$

Where:

S_f = the resuspension factor, m^{-1}

$\bar{X}$ = the average airborne concentration, radioactivity/ m^3

S_A = the surface deposition, radioactivity/ m^2

Maximum annual doses in the PWR facility were calculated based on airborne radionuclide concentrations derived from a constant resuspension factor of $5 \times 10^{-6} m^{-1}$. This value is one-tenth of the value suggested in an IAEA study for surfaces with removable contamination.⁽⁷⁾ A lower value than the IAEA value is assumed here because the unrestricted release of facilities should not be contemplated with readily removable contamination remaining on accessible surfaces. Actual facility resuspension factors may be considerably lower than this value.

The resuspension factor model used to estimate the airborne radionuclide concentrations from environmental contamination is given in Equation E.3.⁽⁸⁾

⁽⁷⁾ Monitoring of Radioactive Contamination on Surfaces. International Atomic Energy Agency, Technical Report Series No. 120, Vienna, p. 4, 1970.

⁽⁸⁾ L. R. Anspaugh, J. H. Shinn and P. L. Phelps, Resuspension and Redistribution of Plutonium in Soils. UCRL-76419, pp. 14-18, January 1975.

$$S_f = 10^{-4} \exp(-\lambda \sqrt{t}) + 10^{-9} \quad (E.3)$$

Where:

S_f = defined in Equation E.2

10^{-4} = resuspension factor at time $t=0$, m^{-1}

$\lambda = 0.15$ = an effective decay constant controlling
the availability of material for resuspension,
 $day^{-1/2}$

t = time after deposition, days

The second term in Equation E.3, 10^{-9} , has units of m^{-1} and is added based on the assumption that no further measurable decrease in the resuspension factor occurs after the value measured at the longest period for which data are available.⁽⁸⁾ The time dependence of the site resuspension factor is illustrated in Figure E.1-1.

Since unrestricted use of the site will allow activities such as unlimited farming after decommissioning, the amount of surface contamination available for resuspension must be defined. For the calculation of the maximum annual doses, it is assumed that the site surface is plowed after decommissioning to a depth of 15 cm, and the surface contamination is uniformly mixed into that volume of soil, thus reducing the amount of material available for resuspension.

E.1.2 Food Pathway Equations

The radiation dose, R_{vr} in mrem, to a person consuming vegetation grown on the contaminated site is given in Equation E.4.

$$R_{vr} = \sum_{i=1}^n C_{iv} U_v D_{ir} \quad (E.4)$$

Similarly, the radiation dose, R_{ar} in mrem, to a person consuming a particular contaminated animal product is given in Equation E.5.

FIGURE E.1-1. Time Dependence of Surface Resuspension Factor for the Site

$$R_{ar} = \sum_{i=1}^n C_{ia} U_a D_{ir} \quad (E.5)$$

Where:

C_{iv} , C_{ia} = the concentration of radionuclide i in the vegetable or animal products, pCi/kg or pCi/l

U_v , U_a = annual consumption of contaminated vegetable or animal products, kg

D_{ir} = a radiation dose or dose commitment factor discussed in Section E.1.

Exposure from routine releases is assumed to result in a one year chronic (continued) ingestion at a uniform rate. Specific values of U_v and U_a (consumption parameters) are taken from the values listed in WASH-1258⁽⁹⁾ for both the maximum individual and populations.

The model used for estimating the transfer of radionuclides from air to plants through leaves and soil to farm products was originally derived by Soldat.⁽⁵⁾ A more detailed treatment of this topic is found in Reference 2.

⁽⁵⁾ J. K. Soldat, Modeling of Environmental Pathways and Radiation Doses from Nuclear Facilities. USAEC Report BNWL-SA-3939, Pacific Northwest Laboratory, Richland, WA, 1971.

⁽⁹⁾ U.S. AEC, Final Environmental Statement Concerning Proposed Rule-Making Action: Numerical Guides for Design Objectives and Limiting Conditions for Radioactive Material in Light Water-Cooled Nuclear Power Reactor Effluents. WASH-1258, Directorate of Regulatory Standards, Volume 1 of 3, Tables 6B-4 and 6B-5, pp. 6B-34 and 6B-35, July 1973.

⁽²⁾ D. A. Baker, G. R. Hoenes and J. K. Soldat, "FOOD - An Interactive Code to Calculate Internal Radiation Doses from Contaminated Food Products." in Environmental Modeling and Simulation. Proceedings of a conference held in Cincinnati, OH, April 20-22, 1976, pp. 204-208, EPA, Washington, DC, 1976.

E.1.2.1 Deposition on Food Products

Equation E.6 is used to describe the deposition of airborne particulate radionuclides directly on food products:

$$d_i = 86,400 \bar{x}_i V_{di} \quad (E.6)$$

Where:

d_i = deposition rate or flux of radionuclides i , pCi/(m²-day).

86,400 = dimensional conversion factor, seconds/day

V_{di} = deposition velocity of radionuclide i , m/second

$\bar{x}_i$ = annual average air concentration of radionuclide i , pCi/m³, from the resuspension factor discussed in Section E.1.1.

Specific values of V_{di} , the deposition velocity of the various radionuclides, are contained in Reference 10.

E.1.2.2 Concentration in Vegetation

The concentration of radioactive material in vegetation resulting from deposition onto plant foliage and uptake from the soil of prior depositions on the ground is given in Equation E.7.

$$C_{iv} = d_i \left[\frac{r T_v (1 - e^{-\lambda_i t_e})}{Y_v \lambda_i} + \frac{B_{vi} (1 - e^{-\lambda_i t_b})}{P \lambda_i} \right] e^{-\lambda_i t_h} \quad (E.7)$$

Where:

C_{iv} = concentration of radionuclide i in edible portion of plant, pCi/kg.

r = fraction of deposition retained on plant (dimensionless), taken to be 0.25

(10) D. A. Baker, User Guide for Computer Program Food. USERDA Report BNWL-2209, Pacific Northwest Laboratories, Richland, WA, February 1977.

T_v = factor for translocation of externally deposited radionuclides to edible parts of plants (dimensionless). For simplicity it is taken to be independent of radionuclide and is assumed to be 1 for leafy vegetables and fresh forage, and 0.1 for all other produce, including grain. (Reference 4 lists values of this parameter, which vary with nuclide.)

λ_i = radiological decay constant for radionuclide i , days^{-1}

$\lambda_{Ei} = \lambda_i + \lambda_w$ = the effective removal constant for radionuclide i , days^{-1}

λ_w = weathering removal constant for vegetation taken to be $0.693/14$, days^{-1}

Y_v = plant yield, $\text{kg (wet weight)}/\text{m}^2$

B_{vi} = concentration factor for plant uptake of nuclide i from soil, $\text{pCi/kg (wet weight) per pCi/kg (dry soil)}$

t_b = time for buildup of radionuclides in the soil, days, assumed to be 40 years of normal PWR operation

t_e = time of above ground exposure of food crop to contamination during growing season, days

t_h = holdup time between harvest and food consumption, days

P = soil "surface density", $\text{kg (dry soil)}/\text{m}^2$, assuming that the contaminated ground is plowed to a depth of 15 cm for all food product consumption dose calculations. P has a value of $224 \text{ kg}/\text{m}^2$. For the maximum annual doses, the site is assumed to be plowed immediately after decommissioning.

The first term inside the brackets of Equation E.7 relates to the concentration resulting from direct deposition on foliage during the growing season. The second term relates to the uptake from the soil and reflects the deposition throughout the 40-year PWR operating life. Initial plowing is assumed to take place immediately after decommissioning. Specific values for the parameters used in Equation E.7 are found in References 2, 9, and 10.

E.1.2.3 Concentration in Animal Products

The radionuclide concentration in an animal product such as meat, milk or eggs is dependent on the amount of contaminated forage or feed eaten by the animal. This concentration is described by Equation E.8.

$$C_{ia} = S_{ia} \left[C_{iF} Q_F \right] \quad (E.8)$$

Where:

C_{ia} = concentration of radionuclide in animal product, pCi/kg or pCi/l

S_{ia} = transfer coefficient of radionuclide i from daily intake of animal to edible portion of animal product, pCi/l (milk) per pCi/day or pCi/kg (animal product) per pCi/day

C_{iF} = concentration of nuclide i in feed or forage, pCi/kg, calculated from Equation E.7 above

Q_F = consumption rate of contaminated feed or forage by animal, kg/day.

Specific values for the parameters used in Equation E.8 are found in References 2, 9, and 10. Plant concentration factors were originally taken from URCL-50163, Part IV⁽¹¹⁾ as explained in Reference 4. For parameters where data were lacking, comparisons were made with biological data from chemically similar elements.

E.1.3 Air Submersion Radiation Dose

The formula used to calculate doses from air submersion is shown in Equation E.9.

$$R_{pr} = U_p \sum_{i=1}^n \bar{x}_i D'_{ipr} \quad (E.9)$$

where:

R_{pr} = the external dose rate from n nuclides via pathway p to organ r of a person, mrem/yr

U_p = 8766 hr/yr for air submersion, and

D'_{ipr} = a dose rate factor discussed in Section E.1

$\bar{\chi}_i$ = the annual average airborne concentration of isotope i , pCi/m³; from the product of the airborne release quantity times the reported meteorological dispersion parameter discussed in Section 7 and Appendix B for the first year dose and the fifty year dose commitment, or from a resuspension calculation discussed in Section E.1.1 for the maximum annual dose calculation.

Equation E.9 is used to calculate the yearly external dose to an individual. The population dose in man-rem/yr is determined by multiplying the dose from Equation E.9 by the population located within the sector of the land annulus of concern around the site. Values of the dose are assumed to be constant at all locations within each sector.

E.1.4 Inhalation Radiation Dose

Inhalation doses, R_{ipr} in mrem, are calculated using Equation E.10.

$$R_{ipr} = \sum_{i=1}^n 3.2 \times 10^7 D_{ipr} \bar{\chi}_i B \quad (E.10)$$

Where:

3.2×10^7 = the number of seconds in a year

$\bar{\chi}_i$ = the annual average airborne concentration of isotope i , pCi/m³

B = ventilation rate of individual exposed; for chronic release,

$B = 2.31 \times 10^{-4}$ m³/second; for acute release 0-8 hr.,

$B = 3.47 \times 10^{-4}$ m³/second

D_{ipr} = radiation dose or dose rate factor discussed in Section E.1.

The inhalation dose model in equation E.10 is consistent with the model presented by the ICRP in Publication 2.⁽³⁾

E.1.5 External Radiation Dose

Maximum annual doses resulting from exposure to surface contamination, both in the facility and on the site, were calculated using Equation E.1 and external dose rate factors discussed in Section E.1. These dose rate factors were calculated at a point located one meter above the contaminated surfaces. Doses to workers during decommissioning operations were calculated using measured dose rate information as discussed in Section 7 and Appendix C.

E.1.6 Maximum Annual Radiation Dose

Maximum annual radiation doses to the reference organs of an individual are calculated using the pathway models and parameters discussed previously. The annual dose from radiation exposures originating from sources external to the human body tends to decrease as time after shutdown increases due to radioactive decay of the residual radioactivity. Annual doses to organs of reference resulting from chronically ingested or inhaled radionuclides, however, tend to increase with increasing time after shutdown until a maximum value is reached. The annual dose then tends to decrease with increasing time due to both radioactive decay of the residual radioactivity and biological elimination of radionuclides deposited in the organ of reference. For continuous exposure to a radioactively decaying source, the year in which this maximum annual dose occurs depends on the chemical and physical characteristics of the radionuclides, the organs of reference and the environmental pathway.

Maximum annual doses are determined for people working in the decommissioned reference facility by assuming continuous weekly exposure (i.e., 40 hours) to the residual radioactivity. Annual doses from external exposure and inhalation are calculated for each year out to fifty years after planned release of the facility for non-nuclear use. Ingestion of residual activity by people working in the decommissioned facility released for unrestricted use was deemed to be an unrealistic exposure pathway. Maximum annual doses for people using the decommissioned reference site are determined by assuming; 1) continuous residence on the site, i.e., 168 hours per week and 2) consumption of farm products grown on the site.

The maximum annual dose calculation uses existing models and computer codes that are designed to calculate the dose accumulated from all pathways. These codes were modified to calculate dose rate (annual dose) as described in the following paragraphs. Calculation of the annual dose in any year requires a detailed accounting of the dose resulting from previous years intake. The annual dose for the first year, to an organ of interest, is simply the summation of the radiation dose from all exposure pathways. For the second year, the annual dose, A_2 , is the summation of the radiation dose from all exposure pathways during the second year and the dose during the second year from the first year's intake. The annual dose for the second year after exposure begins can be expressed by the following relationship:

$$A_2 = R_2 + (R_{1,2} - R_{1,1}) \quad (E.11)$$

where:

A_2 = the annual dose during the second year to the organ of interest, mrem.

R_2 = the second year dose from all exposure pathways to the organ of interest from intake during the second year, mrem.

$R_{1,2}$ = the dose for two years from all exposure pathways to the organ of reference from intake during the first year, mrem.

$R_{1,1}$ = the dose for the first year from all exposure pathways to the organ of reference from intake during the first year, mrem.

In effect, the term in parenthesis accounts for the dose delivered to the organ of reference in the second year from radionuclides deposited in the organ of reference by the first year's intake. In general, the parenthetical terms can be expressed by the form $R_{i,j}$ where i is a previous year's intake and j is the number of years for which the dose from that intake is calculated.

The annual dose for the third year of exposure, A_3 , has a form similar to that for the second year, with the addition of a term to account for the dose delivered in the third year from the second year's intake. The annual dose in the third year of exposure is given by:

$$A_3 = R_3 + (R_{1,3} - R_{1,2}) + (R_{2,3} - R_{2,2}) \quad (E.12)$$

Where:

A_3 = the annual dose during the third year to the organ of interest, mrem

R_3 = the third year dose from all exposure pathways to the organ of interest from intake during the third year, mrem

and $R_{1,3}$, $R_{2,3}$, $R_{1,2}$, $R_{2,2}$ are expressed by the general form $R_{i,j}$ defined previously.

The general expression for calculating the maximum annual dose to an organ of interest from all exposure pathways is shown in Equation E.13.

$$A_j = R_j + \sum_{i=1}^{j-1} R_{i,j} - R_{i,j-1} \quad (E.13)$$

Where:

A_j = the annual dose during year j to the organ of interest, mrem

R_j = the dose during the j^{th} year from all exposure pathways to the organ of interest from intake during the j^{th} year, mrem

And:

$R_{i,j}$, $R_{i,j-1}$ are as previously defined.

Generally, the summation expression accounts for the dose delivered to the organ of reference during the j^{th} year from radionuclides deposited in the organ by uptake from previous years.

The annual dose for any year, A_j , to the organ of interest is calculated for each value of j from $j = 1$ to $j = 50$ and the maximum annual dose is determined by inspection. The radionuclide inventories used are adjusted for

radioactive decay during the calculation, and the site resuspension factor decreases with time as discussed in Section E.1.1.

E.2 CALCULATED RADIATION DOSES AT THE REFERENCE PWR

The following section contains the radionuclide inventories, presented in Section 7, and the maximum annual radiation doses that were calculated from those inventories for Section 8 of this report. These doses are used in Section 8 to calculate the acceptable residual contamination levels for public use of the decommissioned reference PWR and its site.

E.2.1 Facility and Site Radionuclide Inventories

The radionuclide inventories, which are used to calculate the maximum annual doses, are listed here to help illustrate their relationship to calculated doses. Table E.2-1 is a list of the surface contamination levels in the reference PWR facility normalized to one μCi per square meter at selected decay times. These radionuclide inventories are used as being representative of the isotopic composition of the facility surface contamination. Tables E.2-2 and E.2-3 are lists of the radionuclides estimated to be deposited on the site from annual atmospheric releases^(11,12) over 40-years of normal PWR operation.

E.2.2 Calculated Maximum Annual Doses

Maximum annual doses to the maximum exposed individual, as discussed in Section E.1.6, were calculated for the reference inventories listed in Table E.2-1, E.2-2 and E.2-3. The resulting doses are used in Section 8 to calculate the acceptable residual contamination levels for public use of the decommissioned reference PWR and site. Radiation doses, calculated using the surface contamination levels in the reference PWR facility, Table E.2-1, are listed in Tables E.2-4, E.2-5, and E.2-6 for shutdown, 10 years, and 30 years or more after PWR shutdown. Dose pathways of concern in the PWR facility are inhalation, direct exposure from surface contamination, and air

(11) U.S. NRC, Final Generic Environmental Statement on the Use of Recycled Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors. NUREG-0002, Vol. 3, August 1976.

(12) U.S. NRC, Radioactive Materials Released from Nuclear Power Plants (1975). NUREG-0218, March 1977.

submersion. Inhalation and air submersion doses are calculated using the facility resuspension factor of $5 \times 10^{-6} \text{ m}^{-1}$ defined in Section E.1-1.

Maximum annual doses from the contaminated site were calculated using the inventories listed in Tables E.2-2 and E.2-3. Dose pathways of concern for the site are ingestion of food products, inhalation, direct exposure from ground contamination and air submersion. The resuspension model discussed in Section E.1.1 was used to determine the airborne activity resuspended as a function of time. The average deposition period or "age" of the material deposited on the site is assumed to be 10 years at the PWR final shutdown. Table E.2-7 through E.2-16 contain tabulations of the calculated maximum annual doses for the radionuclide inventories deposited on the site. The calculations are at plant shutdown and for decay periods of 10, 30, 50, and 100 years after shutdown.

The maximum annual doses reported in Table E.2-4 through E.2-16 contain listings of the radionuclides that contribute 1% or more to the total dose of the organs of reference.

TABLE E.2-1. Calculated Isotopic Composition of Radioactive Surface Contamination in the Reference PWR

Radionuclide	Fractional Surface Contamination at Decay Times of:				
	Shutdown	10 Years	30 Years	50 Years	100 Years
^{51}Cr	6.9×10^{-4}	—(a)	—	—	—
^{54}Mn	1.4×10^{-3}	—	—	—	—
^{55}Fe	2.2×10^{-2}	2.8×10^{-3}	—	—	—
^{59}Fe	8.7×10^{-4}	—	—	—	—
^{58}Co	7.5×10^{-3}	—	—	—	—
^{60}Co	7.5×10^{-2}	3.2×10^{-2}	3.8×10^{-3}	4.2×10^{-4}	—
^{89}Sr	1.2×10^{-3}	—	—	—	—
^{90}Sr	6.9×10^{-4}	8.8×10^{-4}	9.1×10^{-4}	8.8×10^{-4}	—
^{90}Y	6.9×10^{-4}	8.8×10^{-4}	9.1×10^{-4}	8.8×10^{-4}	—
^{95}Zr	2.5×10^{-4}	—	—	—	—
^{95}Nb	2.5×10^{-4}	—	—	—	—
$^{129\text{m}}\text{Te}$	3.1×10^{-4}	—	—	—	—
^{131}I	1.4×10^{-2}	—	—	—	—
^{134}Cs	1.2×10^{-1}	6.6×10^{-3}	—	—	—
^{136}Cs	1.1×10^{-3}	—	—	—	—
^{137}Cs	7.5×10^{-1}	9.6×10^{-1}	9.9×10^{-1}	9.9×10^{-1}	1.0
TOTAL	1.0	1.0	1.0	1.0	1.0

(a) A dash indicated values less than 10^{-4} .

NOTE: Inventories at each time period are normalized to one μCi per square meter.

TABLE E.2-2. Estimated Accumulated Radioactivity of Radionuclides Deposited on the PWR Site from Annual Atmospheric Releases Reported in GESMO(11)

Deposited Radioactivity ^(a) ($\mu\text{Ci}/\text{m}^2$) at Selected Decay Times of:					
Radionuclides	Shutdown	10 Years	30 Years	50 Years	100 Years
^{54}Mn	4.2×10^{-4}	9.1×10^{-8}	4.3×10^{-15}	-- (b)	--
^{58}Co	1.5×10^{-4}	--	--	--	--
^{60}Co	5.3×10^{-3}	1.4×10^{-3}	9.8×10^{-5}	6.8×10^{-6}	8.7×10^{-9}
^{59}Fe	6.3×10^{-6}	--	--	--	--
^{89}Sr	7.5×10^{-6}	--	--	--	--
^{90}Sr	9.0×10^{-4}	7.0×10^{-4}	4.2×10^{-4}	2.5×10^{-4}	7.1×10^{-5}
^{90}Y	9.0×10^{-4}	7.0×10^{-4}	4.2×10^{-4}	2.5×10^{-4}	7.1×10^{-5}
^{131}I	2.8×10^{-5}	--	--	--	--
^{133}I	2.8×10^{-6}	--	--	--	--
^{134}Cs	5.2×10^{-4}	1.8×10^{-5}	2.1×10^{-5}	2.4×10^{-11}	--
^{137}Cs	9.3×10^{-3}	7.4×10^{-3}	4.7×10^{-3}	2.9×10^{-3}	9.3×10^{-4}

(11) U.S. NRC, Final Generic Environmental Statement on the use of Recycled Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors. NUREG-0002, Vol. 3, August 1976.

(a) Based on a 40-year plant life and normalized to the Reference PWR power rating of 1175 MWe and plant capacity factor of 0.75.

(b) A dash indicates values less than $1 \times 10^{-15} \mu\text{Ci}/\text{m}^2$.

TABLE E.2-3. Estimated Accumulated Radioactivity of Radionuclides Deposited on the PWR Site from 1975 Reported Atmospheric Releases (12)

Radionuclide	Deposited Radioactivity ^(a) ($\mu\text{Ci}/\text{m}^2$) at Selected Decay Times of:				
	Shutdown	10 Years	30 Years	50 Years	100 Years
⁵¹ Cr	1.1×10^{-5}	--(b)	--	--	--
⁵⁴ Mn	2.6×10^{-5}	5.6×10^{-9}	--	--	--
⁵⁷ Co	1.2×10^{-6}	1.0×10^{-10}	--	--	--
⁵⁸ Co	5.6×10^{-5}	--	--	--	--
⁶⁰ Co	8.0×10^{-4}	2.1×10^{-4}	1.5×10^{-5}	1.0×10^{-6}	1.3×10^{-9}
⁵⁹ Fe	5.1×10^{-6}	--	--	--	--
⁸⁹ Sr	1.6×10^{-6}	--	--	--	--
⁹⁰ Sr	1.6×10^{-4}	1.3×10^{-4}	7.7×10^{-5}	4.6×10^{-5}	1.3×10^{-5}
⁹⁰ Y	1.6×10^{-4}	1.3×10^{-4}	7.7×10^{-5}	4.6×10^{-5}	1.3×10^{-5}
⁹⁵ Zr	9.1×10^{-6}	--	--	--	--
⁹⁵ Nb	9.1×10^{-6}	--	--	--	--
¹⁰³ Ru	8.2×10^{-7}	--	--	--	--
^{110m} Au	2.0×10^{-6}	1.2×10^{-10}	--	--	--
¹²⁴ Sb	1.1×10^{-6}	--	--	--	--
¹²⁵ Sb	1.6×10^{-6}	1.3×10^{-7}	7.6×10^{-10}	4.6×10^{-12}	--
¹³¹ I	1.7×10^{-4}	--	--	--	--
¹³³ I	2.6×10^{-6}	--	--	--	--
¹³⁴ Cs	4.6×10^{-5}	1.6×10^{-6}	1.8×10^{-9}	2.1×10^{-12}	--
¹³⁶ Cs	2.8×10^{-7}	--	--	--	--
¹³⁷ Cs	8.5×10^{-4}	6.8×10^{-4}	4.3×10^{-4}	2.7×10^{-4}	8.6×10^{-5}
¹⁴⁰ Ba	2.3×10^{-6}	--	--	--	--
¹⁴⁰ La	2.3×10^{-6}	--	--	--	--
¹⁴¹ Ce	2.8×10^{-8}	--	--	--	--
¹⁴⁴ Ce	1.4×10^{-6}	1.6×10^{-10}	--	--	--

(12) U.S. NRC, Radioactive Materials Released from Nuclear Power Plants (1975). NUREG-0218, March 1977.

(a) Based on a 40-year plant life and normalized to the Reference PWR power rating of 1175 MWe and plant capacity factor of 0.75.

(b) A dash indicated values less than $1 \times 10^{-15} \mu\text{Ci}/\text{m}^2$.

TABLE E.2-4. Calculated Maximum Annual Radiation Dose to Organs of the Maximum Exposed Individual Working in the Released Reference Decommissioning PWR Facility (Facility Decommissioned at Shutdown) Based on Normalized Activity Levels of $1 \mu\text{Ci}/\text{m}^2$

Organ of Reference	Radionuclide (a)	Maximum Annual Dose, mrem per $\mu\text{Ci}/\text{m}^2$ via			
		Inhalation	Floor (b)	Air Submersion	All Pathways
Thyroid at 1 year (c)	^{60}Co	-- (e)	7.6×10^{-1}	1.5×10^{-3}	7.7×10^{-1}
	$^{131}\text{I} + \text{D} (d)$	2.5×10^{-1}	2.4×10^{-2}	4.3×10^{-5}	2.8×10^{-1}
	^{134}Cs	--	8.6×10^{-1}	1.6×10^{-3}	8.7×10^{-1}
	$^{137}\text{Cs} + \text{D}$	--	1.9	3.3×10^{-3}	1.9
	TOTAL	2.5×10^{-1}	3.6	6.7×10^{-3}	3.9
Lung at 2 years	^{60}Co	1.1	6.7×10^{-1}	1.3×10^{-3}	1.8
	^{134}Cs	2.7×10^{-2}	6.2×10^{-1}	1.1×10^{-3}	6.5×10^{-1}
	$^{137}\text{Cs} + \text{D}$	1.5×10^{-1}	1.8	3.4×10^{-3}	2.0
	TOTAL	1.3	3.1	5.9×10^{-3}	4.4
Body at 1 year	^{60}Co	9.8×10^{-4}	7.6×10^{-1}	1.5×10^{-3}	7.7×10^{-1}
	^{134}Cs	8.7×10^{-2}	8.6×10^{-1}	1.6×10^{-3}	9.5×10^{-1}
	$^{137}\text{Cs} + \text{D}$	3.0×10^{-1}	1.9	3.5×10^{-3}	2.2
	TOTAL	3.9×10^{-1}	3.6	6.7×10^{-3}	4.0
Bone at 1 year	^{60}Co	--	7.6×10^{-1}	1.5×10^{-3}	7.7×10^{-1}
	^{134}Cs	4.1×10^{-2}	8.6×10^{-1}	1.6×10^{-3}	9.1×10^{-1}
	$^{137}\text{Cs} + \text{D}$	3.0×10^{-1}	1.9	3.5×10^{-3}	2.2
	TOTAL	3.5×10^{-1}	3.6	6.7×10^{-3}	3.9

(a) The dose from all radionuclides in the inventory is calculated and included in the total, but only those which contribute more than one percent of the total from all pathways are listed

(b) External dose from floor contamination

(c) The time at which the dose rate peaks after exposure begins for the facility decommissioned at shutdown

(d) + D means plus daughter radionuclides

(e) A dash indicates a dose less than 10^{-13} mrem/year

TABLE E.2-5. Calculated Maximum Annual Radiation Dose to Organs of the Maximum Exposed Individual Working in the Released Reference Decommissioned PWR Facility (Facility Decommissioned 10 Years After Shutdown) Based on Normalized Activity Levels of $1 \mu\text{Ci}/\text{m}^2$

Organ of Reference	Radionuclide (a)	Maximum Annual Dose, mrem per $\mu\text{Ci}/\text{m}^2$ via:			
		Inhalation	Floor (b)	Air Submersion	All Pathways
Thyroid at 1 year (c)	^{60}Co	-- (e)	3.3×10^{-1}	6.4×10^{-4}	3.3×10^{-1}
	^{134}Cs	--	4.8×10^{-2}	8.6×10^{-5}	4.8×10^{-2}
	$^{137}\text{Cs} + \text{D} (d)$	--	2.4	4.5×10^{-3}	2.4
	TOTAL	--	2.8	5.2×10^{-3}	2.8
Lung at 2 years	^{60}Co	4.7×10^{-1}	2.9×10^{-1}	5.6×10^{-4}	7.5×10^{-1}
	^{134}Cs	1.5×10^{-3}	3.4×10^{-2}	6.1×10^{-5}	3.5×10^{-2}
	$^{137}\text{Cs} + \text{D}$	1.9×10^{-1}	2.4	4.4×10^{-3}	2.6
	TOTAL	6.8×10^{-1}	2.7	5.0×10^{-3}	3.4
Body at 2 years	^{60}Co	1.2×10^{-3}	2.9×10^{-1}	5.6×10^{-4}	2.9×10^{-1}
	^{134}Cs	5.9×10^{-3}	3.4×10^{-2}	6.1×10^{-5}	4.0×10^{-2}
	$^{137}\text{Cs} + \text{D}$	6.1×10^{-1}	2.4	4.4×10^{-3}	3.0
	TOTAL	6.1×10^{-1}	2.7	5.0×10^{-3}	3.3
Bone at 2 years	^{60}Co	--	2.9×10^{-1}	5.6×10^{-4}	2.9×10^{-1}
	^{134}Cs	3.0×10^{-3}	3.4×10^{-2}	6.1×10^{-5}	3.7×10^{-2}
	$^{137}\text{Cs} + \text{D}$	6.6×10^{-1}	2.4	4.4×10^{-3}	3.0
	TOTAL	6.7×10^{-1}	2.7	5.0×10^{-3}	3.4

(a) The dose from all radionuclides in the inventory is calculated and included in the total, but only those which contribute more than one percent of the total from all Pathways are listed.

(b) External dose from floor contamination.

(c) The time at which the dose rate peaks after exposure begins for the facility decommissioned 10 years after shutdown.

(d) + D means plus daughter radionuclides.

(e) A dash indicates a dose less than 10^{-13} mrem/year.

TABLE E.2-6. Calculated Maximum Annual Radiation Dose to Organs of the Maximum Exposed Individual Working in the Released Reference Decommissioned PWR Facility (Facility Decommissioned 30, 50 or 100 Years After Shutdown) Based on Normalized Activity Levels of $1 \mu\text{Ci}/\text{m}^2$

Organ of Reference	Radionuclide (a)	Maximum Annual Dose, mrem per $\mu\text{Ci}/\text{m}^2$ via:			
		Inhalation	Floor (b)	Air Submersion	All Pathways
Thyroid at 1 year (c)	^{60}Co	-- (e)	3.9×10^{-2}	7.6×10^{-5}	3.9×10^{-2}
	$^{137}\text{Cs} + \text{D}$ (d)	--	2.5	4.7×10^{-3}	2.5
	TOTAL	--	2.5	4.7×10^{-3}	2.5
Lung at 2 years	^{60}Co	5.6×10^{-2}	3.4×10^{-2}	6.7×10^{-5}	9.0×10^{-2}
	$^{137}\text{Cs} + \text{D}$	1.9×10^{-1}	2.4	4.5×10^{-3}	2.6
	TOTAL	2.7×10^{-1}	2.5	4.6×10^{-3}	2.7
Body at 2 years	^{60}Co	1.4×10^{-4}	3.4×10^{-2}	6.7×10^{-5}	3.4×10^{-2}
	^{137}Cs	6.2×10^{-1}	2.4	4.5×10^{-3}	3.1
	TOTAL	6.3×10^{-1}	2.5	4.6×10^{-3}	3.1
Bone at 2 years	^{60}Co	--	3.4×10^{-2}	6.7×10^{-5}	3.4×10^{-2}
	$^{137}\text{Cs} + \text{D}$	6.8×10^{-1}	2.4	4.5×10^{-3}	3.1
	TOTAL	6.9×10^{-1}	2.5	4.6	3.1

(a) The dose from all radionuclides in the inventory is calculated and included in the total, but only those which contribute more than one percent of the total from all Pathways are listed.

(b) External dose from floor contamination.

(c) The time at which the dose rate peaks after exposure begins for the facility decommissioned 30, 50, or 100 years after shutdown.

(d) + D means plus daughter radionuclides.

(e) A dash indicates a dose less than 10^{-13} mrem/year.

TABLE E.2-7. Calculated Maximum Annual Radiation Dose to Organs of the Maximum Exposed Individual Residing on the Reference Decommissioned PWR Site (Site Decommissioned at Shutdown) Based on The GESMO Annual Release (NUREG-0002)

Organ of Reference	Radionuclide (a)	Maximum Annual Dose, mrem Via				
		Ingestion	Inhalation	Ground (b)	Air Submersion	All Pathways
Thyroid at 1 year (c)	⁵⁴ Mn	-- (e)	--	2.8×10^{-3}	2.2×10^{-9}	2.8×10^{-3}
	⁶⁰ Co	--	--	1.0×10^{-1}	7.9×10^{-8}	1.0×10^{-1}
	¹³¹ I + D (d)	2.2×10^{-3}	2.5×10^{-7}	9.0×10^{-5}	6.4×10^{-11}	2.2×10^{-3}
	¹³⁴ Cs	--	--	7.1×10^{-3}	5.0×10^{-9}	7.1×10^{-3}
	¹³⁷ Cs + D	--	--	4.5×10^{-2}	3.2×10^{-8}	4.5×10^{-2}
	TOTAL	2.2×10^{-3}	2.6×10^{-7}	1.6×10^{-1}	1.2×10^{-7}	1.6×10^{-1}
Lung at 1 year	⁵⁴ Mn	--	3.1×10^{-7}	2.8×10^{-3}	2.2×10^{-9}	2.8×10^{-3}
	⁶⁰ Co	--	1.5×10^{-5}	1.0×10^{-1}	7.9×10^{-8}	1.0×10^{-1}
	¹³⁴ Cs	3.8×10^{-5}	2.3×10^{-8}	7.1×10^{-3}	5.0×10^{-9}	7.1×10^{-3}
	¹³⁷ Cs + D	4.9×10^{-4}	2.9×10^{-7}	4.5×10^{-2}	3.2×10^{-8}	4.5×10^{-2}
	TOTAL	5.2×10^{-4}	1.9×10^{-5}	1.6×10^{-1}	1.2×10^{-7}	1.6×10^{-1}
Body at 1 year	⁵⁴ Mn	3.1×10^{-5}	1.3×10^{-9}	2.8×10^{-3}	2.2×10^{-9}	2.8×10^{-3}
	⁶⁰ Co	6.8×10^{-4}	3.4×10^{-8}	1.0×10^{-1}	7.9×10^{-8}	1.0×10^{-1}
	⁹⁰ Sr + D	2.5×10^{-2}	4.6×10^{-8}	--	1.6×10^{-12}	2.5×10^{-2}
	¹³⁴ Cs	3.1×10^{-4}	1.9×10^{-7}	7.1×10^{-3}	5.0×10^{-9}	7.4×10^{-3}
	¹³⁷ Cs + D	3.1×10^{-3}	1.8×10^{-6}	4.5×10^{-2}	3.2×10^{-8}	4.8×10^{-2}
	TOTAL	2.9×10^{-2}	2.1×10^{-6}	1.6×10^{-1}	1.2×10^{-7}	1.9×10^{-1}
Bone at 16 years	⁹⁰ Sr + D	1.2	2.6×10^{-6}	--	--	1.2
	¹³⁷ Cs + D	1.6×10^{-3}	7.7×10^{-6}	1.1×10^{-2}	6.8×10^{-10}	1.3×10^{-2}
	TOTAL	1.2	2.7×10^{-6}	1.1×10^{-2}	6.9×10^{-10}	1.2

(a) The dose from all radionuclides in the inventory is calculated and included in the total, but only those which contribute more than one percent of the total from all pathways are listed

(b) External radiation from ground contamination plowed to a depth of 15 cm

(c) The time at which the dose rate peaks after exposure begins for the site decommissioned at shutdown

(d) + D means plus daughter radionuclides

(e) A dash indicates a dose less than 10^{-13} mrem/year

TABLE E.2-8. Calculated Maximum Annual Radiation Dose to Organs of the Maximum Exposed Individual Residing on the Reference Decommissioned PWR Site (Site Decommissioned 10 Years After Shutdown) Based on the GESMO Annual Release (NUREG-0002)

Organ of Reference	Radionuclide ^(a)	Maximum Annual Dose, mrem Via:				
		Ingestion	Inhalation	Ground ^(b)	Air Submersion	All Pathways
Thyroid at 1 Year ^(c)	⁶⁰ Co	-- ^(e)	--	2.7×10^{-2}	2.1×10^{-9}	2.7×10^{-2}
	¹³⁷ Cs + D ^(d)	--	--	3.5×10^{-2}	2.6×10^{-9}	3.5×10^{-2}
	TOTAL	--	--	6.3×10^{-2}	4.7×10^{-9}	6.3×10^{-2}
Lung at 1 year	⁶⁰ Co	--	3.9×10^{-7}	2.7×10^{-2}	2.1×10^{-9}	2.7×10^{-2}
	¹³⁷ Cs + D	3.9×10^{-4}	2.3×10^{-8}	3.5×10^{-2}	2.6×10^{-9}	3.6×10^{-2}
	TOTAL	3.9×10^{-4}	7.3×10^{-7}	6.3×10^{-2}	4.7×10^{-9}	6.3×10^{-2}
Body at 15 years	⁹⁰ Sr + D	2.3×10^{-1}	7.9×10^{-8}	--	--	2.3×10^{-1}
	¹³⁷ Cs + D	1.2×10^{-3}	5.6×10^{-8}	9.7×10^{-3}	5.6×10^{-10}	1.1×10^{-2}
	TOTAL	2.3×10^{-1}	1.3×10^{-7}	9.8×10^{-3}	5.7×10^{-10}	2.4×10^{-1}
Bone at 16 years	⁹⁰ Sr + D	9.1×10^{-1}	7.9×10^{-7}	--	--	9.1×10^{-1}
	¹³⁷ Cs + D	1.3×10^{-3}	5.7×10^{-8}	8.9×10^{-3}	5.1×10^{-10}	1.0×10^{-2}
	TOTAL	9.1×10^{-1}	8.5×10^{-7}	8.9×10^{-3}	5.2×10^{-10}	9.2×10^{-1}

(a) The dose from all radionuclides in the inventory is calculated and included in the total, but only those which contribute more than one percent of the total from all pathways are listed.

(b) External radiation from ground contamination plowed to a depth of 15 cm.

(c) The time at which the dose rate peaks after exposure begins for the site decommissioned 10 years after shutdown.

(d) + D means plus daughter radionuclides.

(e) A dash indicates a dose less than 10^{-13} mrem/year.

TABLE E.2-9. Calculated Maximum Annual Radiation Dose to Organs of the Maximum Exposed Individual Residing on the Reference Decommissioned PWR Site (Site Decommissioned 30 Years After Shutdown) Based on the GESMO Annual Release (NUREG-0002)

Organ of Reference	Radionuclide ^(a)	Maximum Annual Dose, mrem Via:				
		Ingestion	Inhalation	Ground ^(b)	Air Submersion	All Pathways
Thyroid at 1 year ^(c)	⁶⁰ Co	-- ^(e)	--	1.9×10^{-3}	1.2×10^{-10}	1.9×10^{-3}
	¹³⁷ Cs + D ^(d)	--	--	2.2×10^{-2}	1.3×10^{-9}	2.2×10^{-2}
	TOTAL	--	--	2.4×10^{-2}	1.4×10^{-9}	2.4×10^{-2}
Lung at 1 year	⁶⁰ Co	--	2.1×10^{-8}	1.9×10^{-3}	1.2×10^{-10}	1.9×10^{-3}
	¹³⁷ Cs + D	2.4×10^{-4}	1.2×10^{-8}	2.2×10^{-2}	1.3×10^{-9}	2.3×10^{-2}
	TOTAL	2.4×10^{-4}	1.8×10^{-7}	2.4×10^{-2}	1.4×10^{-9}	2.4×10^{-2}
Body at 15 years	⁹⁰ Sr + D	1.4×10^{-1}	4.4×10^{-8}	--	--	1.4×10^{-1}
	¹³⁷ Cs + D	7.8×10^{-4}	3.5×10^{-8}	6.1×10^{-3}	3.5×10^{-10}	6.9×10^{-3}
	TOTAL	1.4×10^{-1}	7.9×10^{-8}	6.1×10^{-3}	3.5×10^{-10}	1.5×10^{-1}
Bone at 16 years	⁹⁰ Sr + D	5.5×10^{-1}	4.5×10^{-7}	--	--	5.5×10^{-1}
	¹³⁷ Cs + D	8.0×10^{-4}	3.6×10^{-8}	5.6×10^{-3}	3.2×10^{-10}	6.4×10^{-3}
	TOTAL	5.5×10^{-1}	4.9×10^{-7}	5.6×10^{-3}	3.2×10^{-10}	5.5×10^{-1}

(a) The dose from all Radionuclides in the inventory is calculated and included in the total, but only those which contribute more than one percent of the total from all pathways are listed.

(b) External Radiation from Ground Contamination plowed to a depth of 15 cm.

(c) The time at which the dose rate peaks after exposure begins for the site decommissioned 30 years after shutdown.

(d) + D means plus daughter radionuclides.

(e) A dash indicates a dose less than 10^{-13} mrem/year.

TABLE E.2-10. Calculated Maximum Annual Dose to Organs of the Maximum Exposed Individual Residing on the Reference Decommissioned PWR Site (Site Decommissioned 50 years After Shutdown) Based on the GESMO Annual Release (NUREG-0002)

Organ of Reference	Radionuclide ^(a)	Maximum Annual Dose, mrem Via:				
		Ingestion	Inhalation	Ground ^(b)	Air Submersion	All Pathways
Thyroid at 1 year ^(c)	¹³⁷ Cs + D ^(d)	-- ^(e)	--	1.4×10^{-2}	8.1×10^{-10}	1.4×10^{-2}
	TOTAL	--	--	1.4×10^{-2}	8.2×10^{-10}	1.4×10^{-2}
Lung at 1 year	¹³⁷ Cs + D	1.5×10^{-4}	7.4×10^{-9}	1.4×10^{-2}	8.1×10^{-10}	1.4×10^{-2}
	TOTAL	1.5×10^{-4}	9.8×10^{-8}	1.4×10^{-2}	8.2×10^{-10}	1.4×10^{-2}
Body at 15 years	⁹⁰ Sr + D	8.4×10^{-2}	2.6×10^{-8}	--	--	8.4×10^{-2}
	¹³⁷ Cs + D	4.9×10^{-4}	2.2×10^{-8}	3.9×10^{-3}	2.2×10^{-10}	4.4×10^{-3}
	TOTAL	8.5×10^{-2}	4.8×10^{-8}	3.9×10^{-3}	2.2×10^{-10}	8.9×10^{-2}
Bone at 16 years	⁹⁰ Sr + D	3.3×10^{-1}	2.7×10^{-7}	--	--	3.3×10^{-1}
	¹³⁷ Cs + D	5.1×10^{-4}	2.3×10^{-8}	3.5×10^{-3}	2.0×10^{-10}	4.0×10^{-3}
	TOTAL	3.3×10^{-1}	3.0×10^{-7}	3.5×10^{-3}	2.0×10^{-10}	3.3×10^{-1}

(a) The dose from all radionuclides in the inventory is calculated and included in the Total, but only those which contribute more than one percent of the total from all pathways are listed.

(b) External Radiation from Ground Contamination plowed to a depth of 15 cm.

(c) The time at which the dose rate peaks after exposure begins for the site decommissioned 50 years after shutdown.

(d) + D means plus daughter radionuclides.

(e) A dash indicates a dose less than 10^{-13} mrem/year.

TABLE E.2-11. Calculated Maximum Annual Radiation Dose to Organs of the Maximum Exposed Individual Residing on the Reference Decommissioned PWR Site (Site Decommissioned 100 Years After Shutdown) Based on the GESMO Annual Release (NUREG-0002)

Organ of Reference	Radionuclide (a)	Maximum Annual Dose, mrem Via:				
		Ingestion	Inhalation	Ground(b)	Air Submersion	All Pathways
Thyroid at 1 year (c)	$^{137}\text{Cs} + \text{D}(\text{d})$	--(e)	--	4.5×10^{-3}	2.6×10^{-10}	4.5×10^{-3}
	TOTAL	--	--	4.5×10^{-3}	2.6×10^{-10}	4.5×10^{-3}
Lung at 1 year	$^{137}\text{Cs} + \text{D}$	4.9×10^{-5}	2.3×10^{-9}	4.5×10^{-3}	2.6×10^{-10}	4.5×10^{-3}
	TOTAL	4.9×10^{-5}	2.7×10^{-8}	4.5×10^{-3}	2.6×10^{-10}	4.5×10^{-3}
Body at 14 years	$^{90}\text{Sr} + \text{D}$	2.4×10^{-2}	7.4×10^{-9}	--	--	2.4×10^{-2}
	$^{137}\text{Cs} + \text{D}$	1.7×10^{-9}	7.7×10^{-9}	1.3×10^{-3}	7.8×10^{-11}	1.5×10^{-3}
	TOTAL	2.4×10^{-2}	1.5×10^{-8}	1.3×10^{-3}	7.8×10^{-11}	2.5×10^{-2}
Bone at 16 years	$^{90}\text{Sr} + \text{D}$	9.3×10^{-2}	7.7×10^{-8}	--	--	9.3×10^{-2}
	$^{137}\text{Cs} + \text{D}$	1.6×10^{-4}	7.2×10^{-9}	1.1×10^{-3}	6.5×10^{-11}	1.3×10^{-3}
	TOTAL	9.3×10^{-2}	8.4×10^{-8}	1.1×10^{-3}	6.5×10^{-11}	9.4×10^{-2}

(a) The dose from all radionuclides in the inventory is calculated and included in the total, but only those which contribute more than one percent of the total from all pathways are listed.

(b) External radiation from ground contamination plowed to a depth of 15 cm.

(c) The time at which the dose rate peaks after exposure begins for the site decommissioned 100 years after shutdown.

(d) + D means plus daughter radionuclides.

(e) A dash indicates a dose less than 10^{-13} mrem/year.

TABLE E.2-12. Calculated Maximum Annual Radiation Dose to Reference Organs of the Maximum Exposed Individual Residing on the Reference Decommissioned PWR Site (Site Decommissioned at Shutdown) Based on 1975 Reported PWR Releases (NUREG-0218)

Organ of Reference	Radionuclide ^(a)	Maximum Annual Dose, mrem Via:				
		Ingestion	Inhalation	Ground ^(b)	Air Submersion	All Pathways
Thyroid at 1 year ^(c)	⁵⁸ Co	-- ^(e)	--	4.5×10^{-4}	3.4×10^{-10}	4.5×10^{-4}
	⁶⁰ Co	--	--	1.5×10^{-2}	1.2×10^{-8}	1.5×10^{-2}
	¹³¹ I + D ^(d)	1.3×10^{-2}	1.5×10^{-6}	5.4×10^{-4}	3.4×10^{-10}	1.4×10^{-2}
	¹³⁴ Cs	--	--	6.2×10^{-4}	4.4×10^{-10}	6.2×10^{-4}
	¹³⁷ Cs + D	--	--	4.1×10^{-3}	3.0×10^{-9}	4.1×10^{-3}
	TOTAL	1.3×10^{-2}	1.5×10^{-6}	2.2×10^{-2}	1.6×10^{-8}	3.5×10^{-2}
Lung at 1 year	⁵⁸ Co	--	3.3×10^{-8}	4.5×10^{-4}	3.4×10^{-10}	4.5×10^{-4}
	⁶⁰ Co	--	2.2×10^{-6}	1.5×10^{-2}	1.2×10^{-8}	1.5×10^{-2}
	¹³¹ I + D	--	--	5.4×10^{-4}	3.9×10^{-10}	5.4×10^{-4}
	¹³⁴ Cs	3.3×10^{-6}	2.0×10^{-9}	6.2×10^{-4}	4.4×10^{-10}	6.3×10^{-4}
	¹³⁷ Cs + D	4.5×10^{-5}	2.7×10^{-8}	4.1×10^{-3}	3.0×10^{-9}	4.1×10^{-3}
	TOTAL	4.8×10^{-5}	3.0×10^{-6}	2.2×10^{-2}	1.6×10^{-8}	2.2×10^{-2}
Body at 15 years	⁹⁰ Sr + D	5.5×10^{-2}	6.9×10^{-8}	--	--	5.5×10^{-2}
	¹³⁷ Cs + D	1.4×10^{-4}	7.0×10^{-9}	1.1×10^{-3}	7.0×10^{-11}	1.3×10^{-3}
	TOTAL	5.5×10^{-2}	7.6×10^{-8}	1.1×10^{-3}	7.1×10^{-11}	5.6×10^{-2}
Bone at 16 years	⁹⁰ Sr + D	2.1×10^{-1}	4.8×10^{-7}	--	--	2.1×10^{-1}
	TOTAL	2.1×10^{-1}	4.9×10^{-7}	1.0×10^{-3}	6.3×10^{-11}	2.1×10^{-1}

^(a) The dose from all radionuclides in the inventory is calculated and included in the total, but only those which contribute more than one percent of the total from all pathways are listed.

^(b) External radiation from ground contamination plowed to a depth of 15 cm.

^(c) The time at which the dose rate peaks after exposure begins for the site decommissioned at shutdown.

^(d) + D means plus daughter radionuclides.

^(e) A dash indicates a dose less than 10^{-13} mrem/year.

TABLE E.2-13. Calculated Maximum Annual Radiation Dose to Reference Organs of the Maximum Exposed Individual Residing on the Reference Decommissioned PWR Site (Site Decommissioned 10 Years After Shutdown) Based on 1975 Reported PWR Releases (NUREG-0218)

Organ of Reference	Radionuclide ^(a)	Maximum Annual Dose, mrem Via:				
		Ingestion	Inhalation	Ground ^(b)	Air Submersion	All Pathways
Thyroid at 1 year ^(c)	⁶⁰ Co	-- ^(e)	--	4.1×10^{-3}	3.2×10^{-10}	4.1×10^{-3}
	¹³⁷ Cs + D ^(d)	--	--	3.2×10^{-3}	2.4×10^{-10}	3.2×10^{-3}
	TOTAL	7.5×10^{-12}	--	7.4×10^{-3}	5.6×10^{-10}	7.4×10^{-3}
Lung at 1 year	⁶⁰ Co	--	5.9×10^{-8}	4.1×10^{-3}	3.2×10^{-10}	4.1×10^{-3}
	¹³⁷ Cs + D	3.5×10^{-5}	2.1×10^{-9}	3.2×10^{-3}	2.4×10^{-10}	3.3×10^{-3}
	TOTAL	3.5×10^{-5}	1.2×10^{-7}	7.4×10^{-3}	5.6×10^{-10}	7.4×10^{-3}
Body at 15 years	⁹⁰ Sr + D	4.3×10^{-2}	1.4×10^{-8}	--	--	4.3×10^{-2}
	¹³⁷ Cs + D	1.1×10^{-4}	5.1×10^{-9}	8.9×10^{-4}	5.2×10^{-11}	1.0×10^{-3}
	TOTAL	4.3×10^{-2}	2.0×10^{-8}	8.9×10^{-4}	5.2×10^{-11}	4.4×10^{-2}
Bone at 16 years	⁹⁰ Sr + D	1.7×10^{-1}	1.5×10^{-7}	--	--	1.7×10^{-1}
	TOTAL	1.7×10^{-1}	1.5×10^{-7}	8.2×10^{-4}	4.7×10^{-11}	1.7×10^{-1}

(a) The dose from all radionuclides in the inventory is calculated and included in the total, but only those which contribute more than one percent of the total from all pathways are listed.

(b) External radiation from ground contamination plowed to a depth of 15 cm.

(c) The time at which the dose rate peaks after exposure begins for the site decommissioned 10 years after shutdown.

(d) + D means plus daughter radionuclides.

(e) A dash indicates a dose less than 10^{-13} mrem/year.

TABLE E.2-14. Calculated Maximum Annual Radiation Dose to Organs of the Maximum Exposed Individual Residing on the Reference Decommissioned PWR Site (Site Decommissioned 30 Years After Shutdown) Based on 1975 Reported PWR Releases (NUREG-0218)

Organ of Reference	Radionuclide ^(a)	Maximum Annual Dose, mrem Via:				
		Ingestion	Inhalation	Ground ^(b)	Air Submersion	All Pathways
Thyroid at 1 year ^(c)	⁶⁰ Co	-- ^(e)	--	2.8×10^{-4}	1.7×10^{-11}	2.8×10^{-4}
	¹³⁷ Cs + D ^(d)	--	--	2.0×10^{-3}	1.2×10^{-10}	2.0×10^{-3}
	TOTAL	--	--	2.3×10^{-3}	1.4×10^{-10}	2.3×10^{-3}
Lung at 1 year	⁶⁰ Co	--	3.2×10^{-9}	2.8×10^{-4}	1.7×10^{-11}	2.8×10^{-4}
	¹³⁷ Cs + D	2.2×10^{-5}	1.1×10^{-9}	2.0×10^{-3}	1.2×10^{-10}	2.1×10^{-3}
	TOTAL	2.2×10^{-5}	3.1×10^{-8}	2.3×10^{-3}	1.4×10^{-10}	2.4×10^{-3}
Body at 15 years	⁹⁰ Sr + D	2.6×10^{-2}	8.0×10^{-9}	--	--	2.6×10^{-2}
	¹³⁷ Cs + D	7.1×10^{-5}	3.2×10^{-9}	5.6×10^{-4}	3.2×10^{-11}	6.3×10^{-4}
	TOTAL	2.6×10^{-2}	1.1×10^{-8}	5.6×10^{-4}	3.3×10^{-11}	2.6×10^{-2}
Bone at 16 years	⁹⁰ Sr + D	1.0×10^{-1}	8.3×10^{-8}	--	--	1.0×10^{-1}
	TOTAL	1.0×10^{-1}	8.6×10^{-8}	5.1×10^{-4}	3.0×10^{-11}	1.0×10^{-1}

- (a) The dose from all radionuclides in the inventory is calculated and included in the total, but only those which contribute more than one percent of the total from all pathways are listed.
- (b) External radiation from ground contamination plowed to a depth of 15 cm.
- (c) The time at which the dose rate peaks after exposure begins for the site decommissioned 30 years after shutdown.
- (d) + D means plus daughter radionuclides.
- (e) A dash indicates a dose less than 10^{-13} mrem/year.

TABLE E.2-15. Calculated Maximum Annual Radiation Dose to Reference Organs of the Maximum Exposed Individual Residing on the Reference Decommissioned PWR Site (Site Decommissioned 50 Years After Shutdown) Based on 1975 Reported PWR Releases (NUREG-0218)

Organ of Reference	Radionuclide ^(a)	Maximum Annual Dose, mrem/yr:				
		Ingestion	Inhalation	Ground ^(b)	Air Submersion	All Pathways
Thyroid at 1 year ^(c)	⁶⁰ Co	-- ^(e)	--	2.0×10^{-5}	1.2×10^{-12}	2.0×10^{-5}
	¹³⁷ Cs + D ^(d)	--	--	1.3×10^{-3}	7.5×10^{-11}	1.3×10^{-3}
	TOTAL	--	--	1.3×10^{-3}	7.6×10^{-11}	1.3×10^{-3}
Lung at 1 year	⁶⁰ Co	--	2.2×10^{-10}	2.0×10^{-5}	1.2×10^{-12}	2.0×10^{-5}
	¹³⁷ Cs + D	1.4×10^{-5}	6.7×10^{-10}	1.3×10^{-3}	7.5×10^{-11}	1.3×10^{-3}
	TOTAL	1.4×10^{-5}	1.7×10^{-8}	1.3×10^{-3}	7.6×10^{-11}	1.3×10^{-3}
Body at 15 years	⁹⁰ Sr + D	1.5×10^{-2}	4.8×10^{-9}	--	--	1.5×10^{-2}
	¹³⁷ Cs + D	4.5×10^{-5}	2.0×10^{-9}	3.6×10^{-4}	2.0×10^{-11}	4.0×10^{-4}
	TOTAL	1.5×10^{-2}	6.8×10^{-9}	3.6×10^{-4}	2.0×10^{-11}	1.6×10^{-2}
Bone at 16 years	⁹⁰ Sr + D	6.0×10^{-2}	5.0×10^{-8}	--	--	6.0×10^{-2}
	TOTAL	6.0×10^{-2}	5.2×10^{-8}	3.2×10^{-4}	1.9×10^{-11}	6.1×10^{-2}

(a) The dose from all radionuclides in the inventory is calculated and included in the total, but only those which contribute more than one percent of the total from all pathways are listed.

(b) External radiation from ground contamination plowed to a depth of 15 cm.

(c) The time at which the dose rate peaks after exposure begins for the site decommissioned 50 years after shutdown.

(d) + D means plus daughter radionuclides.

(e) A dash indicates a dose less than 10^{-13} mrem/year.

TABLE E.2-16. Calculated Maximum Annual Radiation Dose to Organs of the Maximum Exposed Individual Residing on the Reference Decommissioned PWR Site (Site Decommissioned 100 Years After Shutdown) Based on 1975 Reported PWR Releases (NUREG-0218)

Organ of Reference	Radionuclide ^(a)	Maximum Annual Dose, mrem Via:				
		Ingestion	Inhalation	Ground ^(b)	Air Submersion	All Pathways
Thyroid at 1 year ^(c)	¹³⁷ Cs + D ^(d)	-- ^(e)	--	4.1×10^{-4}	2.4×10^{-11}	4.1×10^{-4}
	TOTAL	--	--	4.1×10^{-4}	2.4×10^{-11}	4.1×10^{-4}
Lung at 1 year	¹³⁷ Cs + D	4.5×10^{-6}	2.1×10^{-10}	4.1×10^{-4}	2.4×10^{-11}	4.1×10^{-4}
	TOTAL	4.5×10^{-6}	4.8×10^{-9}	4.1×10^{-4}	2.4×10^{-11}	4.1×10^{-4}
Body at 15 years	⁹⁰ Sr + D	4.4×10^{-3}	1.4×10^{-9}	--	--	4.4×10^{-3}
	¹³⁷ Cs + D	1.4×10^{-5}	6.4×10^{-10}	1.1×10^{-4}	6.5×10^{-12}	1.3×10^{-4}
	TOTAL	4.4×10^{-3}	2.0×10^{-9}	1.1×10^{-4}	6.5×10^{-12}	4.5×10^{-3}
Bone at 16 years	⁹⁰ Sr + D	1.7×10^{-2}	1.4×10^{-8}	--	--	1.7×10^{-2}
	TOTAL	1.7×10^{-2}	1.5×10^{-8}	1.0×10^{-4}	5.9×10^{-12}	1.7×10^{-2}

- (a) The dose from all radionuclides in the inventory is calculated and included in the total, but only those which contribute more than one percent of the total from all pathways are listed.
- (b) External radiation from ground contamination plowed to a depth of 15 cm.
- (c) The time at which the dose rate peaks after exposure begins for the site decommissioned 100 years after shutdown.
- (d) + D means plus daughter radionuclides.
- (e) A dash indicates a dose less than 10^{-13} mrem/year.

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APPENDIX F

GENERIC DECOMMISSIONING ACTIVITIES

A number of activities are common to all modes of decommissioning. These activities include decontamination, packaging and shipping of material for disposal, special equipment, and quality assurance procedures. Each of these activities is considered in a subsequent section.

F.1 DECONTAMINATION

Removal of radioactive material from surfaces can be accomplished in two basic ways: by the use of chemicals that remove surface films containing the radioactive material, and by mechanical cleaning and/or removal of the surface itself.

Chemical decontamination will be performed on all systems (pumps, piping, tanks, etc.) which contain significant residual radioactive contamination. Physical decontamination will be performed on building surfaces as required to achieve radioactivity levels that are less than unrestricted release limits. The methods to be employed for physical decontamination include washing with solutions and/or physical removal of surface layers. Suggested procedures are presented for each of these decontamination methods in the following sections.

F.1.1 Chemical Decontamination Procedure

The principal systems to be chemically decontaminated are the Reactor Coolant System (RCS) and the Chemical and Volume Control System (CVCS). During the planning and preparation stage, procedures and results from previous decontamination efforts will be reviewed to obtain maximum benefit from previous experience. Current as-built drawings will be reviewed to identify system deadlegs and to facilitate planning of system modifications required to achieve decontamination. Existing procedures will be reviewed for applicability to the present effort and modified as necessary. A detailed step-by-step procedure will be developed for the decontamination campaign, with check lists for valve settings, etc. to assure proper operation of the systems.

Several solutions are possible to the problem of reactor coolant system decontamination. The advantages and disadvantages of three possible methods are presented in Table F.1-1.

TABLE F.1-1. Comparison of Several Chemical Decontamination Methods

Method	Advantage	Disadvantage
Alkaline permanganate citric/oxalic acid (citrox)	Demonstrated effective solutions, non-damaging to components.	Required repeated cycles to obtain good decontamination. Generates large volumes of chemicals and rinse water to be concentrated and solidified ($\sim 2650 \text{ m}^3$) Requires installation of reactor vessel piping jumper device(a) to reduce system volume.
30% HNO_3 , 2% tartaric acid--20% NaOH , 3 HF--20% HNO_3	Demonstrated effective solutions. Possibly only one cycle of three steps, each followed by a flush.	Generates large volumes of ($\sim 1700 \text{ m}^3$) chemicals and rinse water to be concentrated and solidified. Requires handling relatively hazardous chemicals. Potentially damaging to components. Requires jumper device(a) to minimize volumes.
Ethylenediaminetetraacetic acid (EDTA) in a 1 to 5 solution	One cycle and one rinse ($\sim 760 \text{ m}^3$) Decontaminate reactor vessel and internals as well as external piping, pumps and steam generators. Do not need the reactor vessel piping jumper(a) to reduce system volume. Contaminants can be removed from the recirculating solution via a side-stream through a cation resin bed during decontamination if desired. Can add the chemical directly to the borated primary coolant water.	Recirculation time is lengthy, probably several days minimum. Potentially damaging to components. May create radioactive ion exchange beds to dispose of. Additional R&D needed to determine optimum concentrations and recirculation conditions.

(a) See Section F.3-7 for description of reactor vessel piping jumper device

The decontamination method selected for this study (Third method, Table F.1-1) utilizes the chelating agent ethylenediaminetetraacetic acid (EDTA) in combination with citric and/or oxalic acid, in a weak (1 to 5%) solution at controlled pH. Up to 18 Mg (40,000 lb) of chemicals will be required for the chemical decontamination program. While extended recirculation times are expected to be required, probably several days to a week, only one system volume of the decontamination solution is required. Final disposal of the decontamination solution will be accomplished by conventional evaporation and solidification methods. Since no intermediate water flushes are needed, there is no large incentive to minimize the system volume and the reactor vessel interior can be decontaminated with the vessel internal hardware in place. This approach effectively doubles the system volume compared with using the reactor vessel piping jumper device (see Section F.3-7). However, it actually minimizes the quantities of decontamination solution and/or rinse water that must be concentrated and solidified, and permits beneficial use of the borated water in the reactor coolant system and in storage. Most other methods require several cycles of chemicals and flushing.

All possible branches of the CVCS will be operated normally during the decontamination period, with the solution circulating through pumps, heat exchangers, piping, and tanks, and returning to the RCS loop for reheat and cleanup.

Initial Conditions. With the reactor completely defueled and the pressure vessel head reinstalled, the RCS and the CVCS will be isolated from the Spent Fuel Pool System.

A survey of radiation levels at many locations around the system will be made to obtain base reference data to compare with similar measurements to be made following the chemical decontamination effort.

Prior to adding the decontamination chemicals to the borated reactor coolant system water, shielded directional radiation detectors will be placed at key locations to provide real-time monitoring of the radiation dose rates and of the spectrum of the gamma rays making up the radiation field.

Safety Requirements. The essential systems and services specified in Section 9.1.5 will be in place and operable.

A Portable Filtered Ventilation Enclosure as described in Section F.3.5 will be used at locations where flame cutting or welding operations are carried on. Approved supplied breathing air systems will be available at these locations for use if needed.

Operations. The reactor coolant system will be filled to normal levels with the decontamination solution, and pressurized to 281 Mg/m^2 (400 psig) before starting the RCS pumps. Additional solution will be put into the CVCS, with feed and bleed from the RCS loop, to decontaminate as much of the CVCS as possible. The solution will be heated by the operation of the RCS pumps. Recirculation will continue through the RC and CVC Systems until the shielded directional detectors, which were installed to monitor key locations, indicate no further reductions in residual activity with increasing time. Samples of decontamination solution will be drawn from selected sampling points during circulation as a further check on the effectiveness of the decontamination process. The solutions will be pumped into the CVC holdup tanks to be held for final disposal. The RCS will be refilled with fresh coolant water, pressurized to 281 Mg/m^2 (>400 psig), and the water circulated for a final flushing. A radiation survey will be made at the previously measured locations to determine the effectiveness of the decontamination effort. With the exception of the reactor pressure vessel, the systems will be drained and the rinse water stored for final evaporation and solidification. Decontamination factors ranging from 10 to 1000 depending on specific location are expected, with a conservative value of 10 assumed for analyses related to Safety and Occupational Exposure.

Final Decontamination Station. The final step in the chemical decontamination process will take place after the system components have been disassembled. All nonactivated stainless steel material will be cut into appropriate sizes and transported to the electropolishing station (described in Section F.3.6) that is installed in the new fuel storage area for final cleanup.

Experience with the electropolishing systems has been outstanding, cleaning metal devices with complex shapes from 10 R/hr to background levels in 10 to 15 min. The system utilizes a high-current, low-voltage DC electricity source, a tank containing an appropriate electrolyte (typically phosphoric acid and/or sulfuric acid) and a set of electrodes to make contact with the piece to be cleaned. Current flow through the piece and the electrolyte to the tank wall, removes surface material from the piece, smoothing and cleaning, until the surface contaminants are removed. Sufficient air sweep over the tanks must be maintained to remove the hydrogen gas that is evolved due to electrolysis during the cleaning process. The piece is then rinsed in clean water, checked for any residual contamination and, when clean, placed in the salvage-scrap material storage location.

It is anticipated that essentially all stainless piping, valves, pump bowls and impellers, tank walls and cavity liners can be cleaned to release levels using the electropolishing technique. Most probably, items such as heat exchangers cannot be successfully decontaminated internally. In those cases, all openings into the device will be sealed and the exterior surface of the device will be decontaminated by washing in the cask wash pit prior to shipment for disposal.

F.1.2 Surface Cleaning Procedures

Removal of contamination from sealed surfaces such as painted walls and floors, tank exteriors, etc., can be accomplished using a variety of techniques. For small quantities of loose contamination on floors, simple sweeping compounds have often been effective. For more tenacious contaminants, various cleansing compounds have been utilized. Several proprietary decontamination solutions are available. Ordinary household detergents are quite effective but produce sizable quantities of waste water to process. Aerosol-type foaming cleansers have proven effective and eliminate the problem of large volumes of waste water, but their use produces sizable quantities of contaminated wiping material to dispose of.

Variable pressure, high velocity water jets can be used quite effectively for some types of surface decontamination work. The devices can be operated by one man, at pressures up to $3,164 \text{ Mg/m}^2$ (4500 psig), using hand-held jet lance. On those contaminants for which the water jet is effective, the jet is a very rapid decontamination method. Its use does produce significant volumes of contaminated water which must be processed for disposal.

If none of the above techniques are successful in removing surface contamination, the most likely solution to the problem is to remove the surface itself.

F.1.3 Structural Materials Removal Techniques

There will be concrete surfaces throughout the facility that will be contaminated to a depth of a few centimeters and which cannot be cleaned to release levels by surface washing techniques. In addition, the concrete in the biological shield surrounding the reactor pressure vessel will be radioactive, containing activation products from neutron bombardment. In both instances, the surface materials must be removed and packaged for disposal. However, rather dissimilar methods will be used to remove the material in the two cases.

Several criteria must be considered when selecting a decontamination method for a particular location. The method should facilitate control of airborne contamination and minimize any spread of contamination. The method should minimize the potential for personnel exposure, and the size and weight of removed materials must be controlled to facilitate packaging and shipping for disposal.

The various methods available for removal of concrete and steel are listed in Table F.1-2 together with the advantages and disadvantages of each. Of particular interest are the explosives techniques for use on the biological shield, and the rock-splitting techniques for localized removal of concrete surfaces.

TABLE F.1-2. Concrete and Steel Removal Methods

Removal Method	Advantages	Disadvantages
Flame Cutting	<ul style="list-style-type: none"> • Concrete, steel liner and rebar may be cut in one operation • Fast cutting speed • Adaptable for remote operation • Adaptable for pivoted circumferential cutting • Adaptable for cutting on irregular surfaces • No vibration/shock • Low noise level • Controlled rate of material removal 	<ul style="list-style-type: none"> • Generates copious quantities of toxic gases and smoke • Consumes large quantity of gas • Smoke could potentially spread gross contamination throughout contamination control envelope unless adequately controlled • Requires a through starter-hole made by a core drill to provide an outlet gas-flow path
Blasting	<ul style="list-style-type: none"> • Fast material removal rate where adjacent material damage is not of concern • Adaptable for cutting irregular surfaces • Adaptable to remote operation • Adaptable to the control of size and weight of material removed • Selective removal of radioactive contamination is possible 	<ul style="list-style-type: none"> • May cause vibrations and shocks (Vibrational control possible by limiting maximum charge per delay) • Requires oxyacetylene torch to cut rebar • High noise level • Generates moderate quantities of dust which must be controlled
Thermic Lance Cutting	<ul style="list-style-type: none"> • Cuts both concrete and steel without difficulty • Fast hole punching speed • Ideally suited for cutting irregular surfaces • Remote operation is possible for up to approximately 3 meters • No vibration/shock • Low noise level • Controlled rate of material removal 	<ul style="list-style-type: none"> • Generates moderate quantities of toxic gases and smoke • Requires need for rock jack to break away out concrete sections • Consumes large quantity of bottled oxygen • Smoke could potentially spread gross contamination throughout contamination control envelope unless adequately controlled
Core Boring and Rock-Splitting	<ul style="list-style-type: none"> • Proven technique used extensively in conventional mining operations • Low noise level • Low vibration/shock level • No toxic gases generated • Water cooling minimizes airborne particulate material • Moderate-to-slow cutting speeds • Controlled rate of material removal 	<ul style="list-style-type: none"> • Cutting through steel rebar or structural steel slows cutting speed and tears diamonds off core drill • Requires additional operation of oxyacetylene torch for rebar • Cooling water required for drilling may have to be filtered and recycled • Not ideally suited for operation on irregular surfaces or in cramped quarters

Two techniques of lesser interest, primarily because of the copious quantities of toxic gases and/or smoke produced, are flame cutting and use of the thermic lance. Flame cutting utilizes commercially available 3 m long by 19 mm dia, thin-walled conduit burning bars. Each bar contains up to seven dissimilar metals in the form of strips or strands and is encased in an outer tube into which oxygen is forced during the burning process. Once ignited, the entire hand-held device is consumed as it burns. After 4 to 5 minutes and with about 0.5 m left, the oxygen supply is shut off and the exothermic reaction stops. The final 0.5 m of the first bar can be attached to a second bar and the burning process repeated thus minimizing loss of bar material.

The thermic lance can also be hand-held and uses powders of similar material to the strands of the burning bar. The powders are forced through a tube or nozzle with oxygen and ignited to form a high temperature cutting medium.

Contaminated Surface Removal. The modified rock splitting tool described in Section F.3.3 was selected as the principal device used in this study for removal of contaminated concrete surfaces. Use of this device permits localized removal of concrete surfaces at depths of 50 to 75 mm with no explosions and very little dust. The principal source of dust will be the drilling of the hole in the concrete into which the splitting tool is inserted. Use of one of the portable filtered ventilation enclosures (Section F.3.5) will minimize scatter of contaminated dust. When decontaminating wall surfaces, work should start at the top of the contaminated area to minimize potential recontamination of the clean surface by falling dust or debris. The material removed will be packaged for disposal.

Electric or pneumatic hammers may also be used, particularly on floor surfaces. These devices have the disadvantage of creating significant quantities of dust and small fragments that might be difficult to contain during the operation, and might readily recontaminate the areas being cleaned.

Activated Concrete Removal. The use of explosives has been demonstrated⁽¹⁾ to be a very effective way to remove activated concrete from the biological shield. The configuration of the shield lends itself to the control of dust and airborne contamination in that the interior of the shield can be enclosed within ventilation confinement envelopes quite easily. Placement of blasting mats ~3 m x 4 m, weighing ~1600 kg (~10 ft x 12 ft, weighing ~3500 lb) over the region being blasted will prevent projectiles of debris from penetrating the confinement barriers. Fine fog sprays of water will help to settle the dust created by the blasts. Typical use of the fog spray has been from one minute before to about 15 minutes after blasting. While the blasting sequences are designed to minimize air pressure surges, the ventilation enclosures must be designed to accommodate those pressure surges that are created. Similarly, attention must be given to the building ventilation confinement system to prevent surge damage to filter systems, with monitoring of the system to assure its continued integrity.

The first step in explosive decontamination of concrete is to drill holes to hold the explosives. In some areas these holes may be drilled parallel to and about 127 mm (5 in.) from the surface of the concrete to be removed, outside of the reinforcing steel. The entire length of the surface to be removed must be drilled. Explosives are inserted in the holes and backfilled with sand if necessary to produce the desired blast depth. In other areas, 127-mm (5-in.) deep holes may be drilled perpendicular to the surface to be removed. The holes can be on 305-mm (1-ft) centers over the entire surface. The holes are filled with mortar after the explosives are inserted. The explosives used may be of several varieties as shown in Table F.1-3.

Four to five strands of 150-grain blasting cord twisted together and placed in a tube inserted in the hole has proven effective, as well as partial sticks of dynamite. Liquid explosives may be used where it is difficult to insert solids. The size of the area to be blasted at one time is controlled so that in the event of an accident there is an acceptably low potential for

⁽¹⁾ Final Program Report, AEC-Elk River Reactor, COO-651-93, United Power Association, Elk River, MN, September 1974.

TABLE F.1-3. Types of Explosives for Use in Concrete Removal

1. PETN (Pentaerythritol tetranitrate)

This explosive is used in the form of detonating cord. It is primarily used during the blasting of boreholes up to 5 m (16 ft) in depth. In addition, it has proven to be effective for surface spalling which requires the removal of very small burdens of about 200 mm (8 in.) from exposed surfaces.

2. 85% High Velocity Gelatin Dynamite

This explosive is primarily used in shallow holes ranging in depth from 0.45 m (18 in.) to 1.5 m (5 ft) and as a partial loading in some holes when a concentration of energy is needed in a particular area. Breakage is excellent, however, the explosive does produce a crushed zone around the borehole, thus generating a larger amount of dust than the PETN.

3. Cast TNT (High Detonation Pressure Primers)

This explosive is used where a high degree of fragmentation with less heaving effect than the conventional dynamite is desired.

4. Binary Energy System (Liquid Explosives)

This explosive is mixed onsite and when not mixed, it is not classed as an explosive in most states. It is used in place of Cast TNT during the spalling of surface contamination and to make shaped charges which can be used to punch holes in high density concrete.

5. Water Gel Explosives

A water gel explosive containing a large amount of aluminum is used as a partial replacement for the 85% dynamite. It has exhibited good shattering characteristics and produced larger rubble than any of the other explosives. Its primary use is in areas of little or no reinforcing rods where larger size rubble is desired.

6. Others

High strength ammonia dynamite and plastics have been investigated via small scale blasts and the effects indicated that they did not produce the desired results.

damage with the explosives involved. There are never more explosives in the building than are actually being used. A blast is carried out with the individual charges sequenced so that the shock wave generated by the blast is minimized. Blasting mats are placed over the blast area to minimize the scattering of debris. A water fog spray is used during and after the blast to minimize dust generation. The blasted area is then inspected for the desired results and surveyed to ensure that all contamination has been removed and all explosives have been detonated. The area may then be protected by painting or with heavy plastic or canvas to prevent recontamination during subsequent blasts.

Before an area is blasted, the ventilation system must be protected from damage due to overpressurization. In some areas this may be accomplished by closing existing dampers during the blast. Normal ventilation may be used in this area except during the actual blast. It may be possible in some areas to install a cover over ventilation openings and remove them after a blast. Other areas will require the use of an auxiliary ventilation system that draws from the blast area and exhausts through a series of roughing filters into the normal building ventilation system.

F.2 PACKAGING AND SHIPMENT OF RADIOACTIVE MATERIALS

Several categories of radioactive material are present in a PWR facility that must be properly packaged and shipped to an authorized disposal site. These materials can be classified into combustibles or noncombustibles, activated, or contaminated, liquid or solid. The bulk of the material, which must be packaged for disposal, will be solid, noncombustible, and either activated or contaminated, and will include such things as the activated reactor vessel and internals, the activated concrete from the biological shield, contaminated concrete from walls and floors, and contaminated piping and equipment. The contact radiation dose rates from these materials will vary from a few mr/hr to thousands of R/hr and will require different types of packaging and shielding depending on the radiation levels.

F.2.1 Packaging

Combustible solids that must be packaged include expended ion exchange resins, filter cartridges from liquid streams, HEPA filters from air streams and

and miscellaneous cleaning materials such as rags, mops, protective clothing, and plastic sheeting and bags. The radioactive content of the expended resins may range from 0.1 to 1600 Ci/m³, while the radioactivity in the filter cartridges may range from 0.04 to 1785 Ci/m³, with typical values in the vicinity of 85 Ci/m³. Since these materials were used on a continuing basis during reactor operation, standard handling procedures and standard packaging and shipping procedures will be in place at the facility and will be followed. Some typical shipping containers and casks are illustrated in Figure F.2-1.

FIGURE F.2-1. Some Typical Shielded Shipping Containers

Similarly, radioactive liquids were generated during reactor operation and standard methods will be in place for handling and disposing of the liquid wastes which are generated during the dismantling operations. Current practice is to concentrate the liquids by evaporation, perhaps even by calcination, to a product that can be fixed in a solid such as ureaformaldehyde or cement, in 55-gal (0.21 m³) drums or larger containers, prior to disposal.

All solid-waste handling, packaging and storage facilities are located in controlled access areas with controlled ventilation. The facilities are

provided with shielding to reduce the radiation exposure to personnel in the adjacent work areas. A typical solid waste system process arrangement is shown in Figure F.2-2. It is designed to package all solid wastes in suitable containers for offsite shipment.

FIGURE F.2-2. Typical Solid Waste System Process Arrangement

It will be necessary to construct shielded disposable containers in order to ship and bury such items as segments of the reactor vessel and the vessel internals, as well as portions of the concrete from the biological shield. Based on experience at the Elk River Reactor dismantling, shielded disposable containers cost approximately \$10,000 each. Some segments will not require shielding but are larger and/or heavier than can be accommodated with wooden containers. In these cases, steel containers will be fabricated specifically for these segments at a cost of about \$6,000 each.

The bulk of the contaminated material and slightly activated material will be packaged in a standard container, illustrated in Figure F.2-3, approximately 1.2 m x 1.2 m x 2.4 m (4 ft x 4 ft x 8 ft), made from plywood, reinforced and sealed with fiberglass, to assure confinement of loose contamination. When necessary, containers of special sizes and shapes will be fabricated. Containers of this type are estimated to cost approximately \$400 each.

FIGURE F.2-3. Typical Wooden Shipping Container

Certain equipment items, such as heat exchangers, pumps, and some valves, will be packaged by sealing the piping connections and using the device as its own outer container. The steam generators will be cut into sections and each piece handled as its own container, as illustrated in Figure F.2-4.

F.2.2 Shipment of Radioactive Waste

All disposal shipments are assumed to be made by exclusive-use truck over the interstate highway system. The hauler must have the appropriate license from the Nuclear Regulatory Commission (NRC) and permits from the Department of Transportation (DOT) before he can handle the radioactive waste material.

All shipments of radioactive material must be made in compliance with federal, state and local regulations. Federal transportation regulations of the Department of Transportation and the Nuclear Regulatory Commission establish container regulations, dose rate limits and handling procedures that ensure the safety of the public and transportation workers during shipment of radioactive materials.⁽²⁾ Federal regulations applicable to the highway transport of radioactive materials are:

- Title 49 Code of Federal Regulations Parts 170-199 (49 CFR 170-199)--Department of Transportation regulations governing the transport of hazardous materials.
- 10 CFR 71--Nuclear Regulatory Commission regulations governing the packaging and shipment of radioactive materials.

Additionally, state agencies regulate vehicle sizes and weights and in some cases transportation routes and times of travel.

Dose rates, for highway shipments in exclusive-use vehicles, must not exceed the following values (DOT, 49 CFR 173.393):

⁽²⁾ General Reference

F. B. Conlon and G. L. Pettigrew, Summary of Federal Regulations for Packaging and Transportation of Radioactive Materials, (BRH/DMRE 71-1) U.S. Department of Health, Education and Welfare Public Health Service, Bureau of Radiological Health, Rockville, MD, February 1971.

(a) ASSUMES RCS AND RELATED SYSTEMS HAVE BEEN CHEMICALLY DECONTAMINATED.

(b) WACHS TRAV-L-CUTTER. E.H. WACHS COMPANY, WHEELING, ILLINOIS 60090.
WACHS TOOLS FOR PIPE AND VALVES BROCHURE (SEPT. 17, 1976)

FIGURE F.2-4. Steam Generator Sectioning and Packaging

- 1) 1000 millirem per hour at 0.91 m (3 ft) from the external surface of the package;
- 2) 200 millirem per hour at any point on the external surface of the vehicle.
- 3) 10 millirem per hour at 1.83 m (6 ft) from the external surface of the vehicle; and
- 4) 2 millirem per hour in any normally occupied position in the vehicle.

These dose rate limits are illustrated in Figure F.2-5 for truck transport.⁽³⁾ All of these criteria must be met on a given shipment, with an exclusive-use vehicle, properly placarded. Current disposal site restrictions limit the quantity of radioactivity per container to $\leq 50,000$ Ci. No limits have been set for the dose rate at the surface of the container, but burial costs increase with increasing surface dose rate. It may be appropriate to place the highly activated material in a federal high-level waste repository.

FIGURE F.2-5. Radiation Dose Limits for Exclusive-Use Vehicles

⁽³⁾ Radioactive Materials Transportation Manual, Nuclear Transport Division, Tri-State Motor Transit Co., Joplin, MO, 1974.

All wastes will be transported to a federal or state licensed commercial low-level waste burial ground. Facilities are available in Nevada, Washington, Illinois, and South Carolina. The average shipping distance is assumed to be 500 miles (one way) for all shipments.

Several commercial transportation companies are available with the special capabilities and trained drivers for hauling radioactive wastes.

A formal accident control and recovery plan will be developed before the first radioactive shipment. It will provide for rapid and orderly utilization of utility, carrier, state and municipal emergency personnel, and NRC radiological assistance teams as required in the unlikely event that any abnormal condition or accident occurs. Procedures for control of contamination and exposure to the public, salvage/recovery, as well as of bodily injury and property damage will be included in the plan.

F.3 SPECIAL EQUIPMENT REQUIREMENTS

A number of activities are involved in meeting the need for special tools and equipment. These activities include: the development of functional and operational requirements for devices needed in the decommissioning work; the preparation of detailed designs for noncommercial devices and modifications to commercially-available equipment; procurement or fabrication of the identified equipment items; and the establishment of a mockup facility for use in testing and development of operational parameters for remotely-operated equipment.

A listing of special tools and equipment together with their functions is given in Table F.3-1. Some of this equipment may be available as part of the facility's normal operating equipment complement. Descriptions of some of the more unique devices are presented in the sections that follow.

F.3.1 Underwater Manipulator

The disassembly and sectioning of the reactor vessel and its internal structures will require extensive use of underwater tools. In particular, tools for cutting massive structures, such as the core barrel and support plate will have to be operable and maneuverable under 9 to 15 m of water, in intense radiation fields. It is assumed that the existing refueling

TABLE F.3-1. Special Tools and Equipment for Dismantlement

Item	No Required	Estimated Unit Cost (\$000)	Functions
Underwater manipulator ^(a)	1	300	Movement and positioning of underwater cutting devices used in sectioning and reactor pressure vessel and vessel internals
Underwater plasma cutting torch ^(a)	2	20	Used for sectioning vessel and internals (Ability to cut 230 mm, SS clad steel vessel needs to demonstrated) Used for cutting stainless steel piping, fuel racks, etc Used for general cutting of piping, support structures, etc Used for sectioning vessel internals, possible reactor vessel steam generators, tanks and pool liners
Underwater oxyacetylene torch	2	5	
Portable plasma cutting torch	4	20	
Arc saw ^(b)	1	100	
Guillotine pipe saws ^(c)	9	4	Portable power pipe saws for removal of piping
Closed circuits, high resolution television systems, underwater and in air	2	(plant equip)	For observation and control of remote or underwater operations in high radiation fields
Shielded vehicle with manipulators ^(d) Exchangeable tools, scoop loader, etc	1	50	For remote handling of highly radioactive materials, packaging of activated concrete rubble, etc
Underwater lights and viewing windows	as req	5 total	For illuminating and observing underwater operations
Submersible pumps with disposable filter cartridges	5	1 5	For rapid cleanup of pool water during underwater cutting operations, collection of chips and particles
Assorted underwater tools such as impact wrenches, grapples, bolt cutters, etc	as req	25 total	For underwater disassembly, handling and packaging of highly activated materials
High velocity water jets ^(e)	—	—	For surface decontamination of piping, tanks and equipment
Hydraulic concrete surface spalling device	4	5	For removal of contaminated concrete surfaces See subsequent description in Section F 3 3
Concrete drills Electric/pneumatic hammers	4	0 5	For drilling holes in concrete as required for surface spalling and volume blasting
Portable filtered ventilation enclosures	4	1 5	For collecting and filtering air, smoke, fumes, etc , from cutting operations on contaminated materials See subsequent description in Section F 3 5
Supplied airbubble suit	200	0 05	Provide personnel with maximum respiratory and surface protection against contamination
Safety nets ^(f)	5	25 total	For use under personnel working on elevated systems and structures
Blasting mats	10	0 5	For use over areas being blasted for removal of concrete, prevention of flying missiles, etc
Shipping casks and liners	as req	0 1/day	For transport and burial of radioactive materials at a disposal site See Section F 2 for details
Electropolishing decontamination	1	100	For final decontamination of valuable materials See subsequent description in Section F 3 6
Mobile radwaste processing unit ^(g)	1	10/month	For additional radwaste processing capacity during dismantling, and use after inplant radwaste system has been dismantled
Primary piping jumper	—	—	For bypassing the reactor vessel volume during chemical decontamination, if a multiple cycle decontamination program is followed, to reduce the volume of solutions and rinse water to be processed See subsequent description in Section F 3 7

^(a)See C00-651-93 Final Program Report on dismantling of Elk River Reactor, September 1974

^(b)Schienger, M P (Schienger, Inc , San Rafael, California) and G A Beitel (Atlantic Richfield Hanford Co , Richland, Washington), Topical Report ARC SAW TESTING, February 18, 1976

^(c)Brochures from E H Wachs, Co , Wheeling, Illinois

^(d)Bainbridge, G R Decommission of Nuclear Facilities, A Review of Status, IAEA, October 1973

^(e)Carver, J Barton, New Uses for Cleaning with Water, POWER, May 1976

^(f)Brochures from SINCO, Safety and Industrial Net Division, East Hampton, Connecticut

^(g)Nucleonics Week, June 24, 1976

bridge crane with its fuel element handling boom can be adapted to this task through the attachment of various devices to the boom. For example, wheeled supporting devices for holding cutting torches, or an arc saw, could readily be attached to the boom for cutting the vessel and its components into packageable pieces. Similarly, pneumatic or electric-driven grappling tools could be attached for grasping and moving cut sections, etc.

The particular devices needed depend on the details of dismantling plan, and on the state-of-the-art in cutting equipment at the time. Each attachment device would be specifically designed and engineered for the planned application, and would require extensive predismantlement testing, to assure proper performance. It is anticipated that the total development program for these devices could cost as much as \$300,000.

F.3.2 Underwater Cutting Devices

It is anticipated that the principal devices used in underwater cutting of activated and/or contaminated structures will be oxyacetylene torches, plasma arc torches, and the arc saw. The underwater oxyacetylene torch is a relatively common device and will not be discussed here.

Plasma Arc Torch. Plasma-arc cutting employs an extremely high temperature, high velocity constricted arc between the electrode in the plasma torch and the piece to be cut. The arc is constructed by passing it through a nozzle with a small diameter orifice. The arc is thereby localized so that its energy is concentrated upon a small area on the base plate where its intense heat melts the metal. The gas, which is preheated by the arc, expands and is accelerated as it is forced through the constricting orifice. The metal which is melted is then continuously removed by the jet-like action of the gas stream to form a kerf. The combined heat and force of the arc stream produce a high-quality, saw-like cut. Where inert gases are used, the cutting process depends upon thermal action alone. When cutting materials such as mild steel and cast iron, increased cutting speeds can be achieved using oxygen-bearing cutting gases. In this case, the chemical energy obtained by the combination of oxygen with the base material is added to the arc heat to permit higher cutting speeds. The process can be used to cut any metal. The plasma-arc torch is shown schematically in Figure F.3-1.

The circuit is basically the same as used for gas tungsten-arc welding with the exception that an electrical lead is taken from the ground and passed through a resistor (R) to the nozzle of the torch. A high frequency spark is used to complete the circuit between the electrode (E) and the nozzle, thereby producing a pilot arc. This pilot arc is then used to initiate the main arc from the electrode to the work piece.⁽⁴⁾

FIGURE F.3-1. Plasma Arc Torch Schematic Diagram

There are two types of electrode holders and automatic cutting controls. One type is designed for cutting where the torch is manually guided. The other type is for mechanized cutting where the torch is mounted on an automatic travel mechanism. Water cooling is used to prolong the operating life of the torch electrode and nozzle. Torch surfaces are insulated as completely as possible to prevent the operator from touching an electrically "hot" metallic surface. Gas pressure regulators and flowmeters are required for controlling the flow of cutting gas. A contactor, usually built into the power

⁽⁴⁾"Welding, Cutting, and Related Processes." Welding Handbook, 6th Edition, Section Three, Part A, American Welding Society, 1970.

supply, is necessary for turning power on and off. For mechanized operation, a travel carriage of the straight-line, circle-cutting or shape-cutting type having the desired speed range is required.

The cutting controls contain relays and solenoid valves to control the flow of cutting gas, to operate the high-frequency power supply required for arc starting and to open and close the cutting power contactor. The controls also provide a flow switch to indicate an adequate flow of cooling water. The machine control provides a means of starting and stopping the carriage on which the torch is mounted.

Arc-constricting nozzles are available in a variety of diameters and shapes. The shape and diameter of the arc-constricting nozzle to be used depends on the application and the current to be used, with high currents requiring larger diameter orifices. Both single port and multiport nozzles have been used for cutting. The multiport nozzles generally have auxiliary ports arranged in a circle around a central orifice.

Those power supplies used for cutting have open-circuit voltages in the range of 120 to 400 V direct current, single phase. Heavy-duty cutting requires power supplies with 400 open-circuit volts. The high open-circuit voltage is required for severing thick plates and for piercing metal as thick as 50 mm (2 in.).

Mixtures of argon and hydrogen or nitrogen and hydrogen are generally used for cutting stainless steel, aluminum and other nonferrous metals. Nitrogen and oxygen, supplied separately to the torch and mixed in the torch nozzle, are used for cutting carbon steel, cast iron and alloy steels.

In mechanized operations, the torch standoff distance, in air, is set between 6 mm (1/4 in.) and 16 mm (5/8 in.) from the work piece. The cutting current, gas type and rate of flow are set at values recommended by the manufacturer of the equipment. These values are primarily determined by the type and thickness of the material being cut. After the operator presses a start button to initiate the cutting arc, the sequence of operations is regulated by the cutting control.

For manual operation, the operator selects the gas flow rate and cutting current according to the table of conditions supplied by the manufacturer or

as obtained from his own testing and development program. The torch is held over the work piece and the cutting arc is struck by pressing a button on the torch handle. The torch is then manually guided over the work piece. At the end of the cut the arc automatically extinguishes and the control automatically opens the contactor and closes the gas solenoid valves. The operator can extinguish the arc at any point simply by moving the torch away from the workpiece.

The plasma cutting process can be used in air or under water. It is especially adaptable to automation when highly radioactive material is to be cut, e.g., the pressure vessel of the Elk River Reactor in Minnesota.⁽¹⁾ As it is not necessary to start the cut at the edge of the plate, the plasma arc torch is particularly adaptable to the piercing and cutting of holes in large plates and vessels. The process is also well adapted to gouging applications, including pad washing and scarfing.

Some typical conditions for mechanized plasma arc cutting of stainless steel in air are given in Table F.3-2.

TABLE F.3-2. Typical Conditions for Plasma Arc Cutting of Stainless Steel

Thickness		Speed (m/min)	Orifice Diameter (mm)	Current (amps)	Gas Flow	
mm	in.				(ft ³ /hr)	m ³ /hr
13	1/2	2.54	3.17	300	150 N ₂	4.25 N ₂
25	1	1.24	3.97	400	175 N ₂	4.96 N ₂
51	2	0.51	3.97	500	130 A - 70 H ₂	3.7 A - 2 H ₂
76	3	0.41	4.76	500	130 A - 70 H ₂	3.7 A - 2 H ₂
102	4	0.20	4.76	500	130 A - 70 H ₂	3.7 A - 2 H ₂

⁽¹⁾ Final Program Report, AEC-Elk River Reactor, C00-561-93, United Power Association, Elk River, MN, September 1974.

Arc Saw. Initial development and demonstration work for an arc saw cutting device has been accomplished.⁽⁵⁾ The results indicate that use of the device, particularly in a contaminated equipment volume reduction program, looks very promising and it is assumed that additional development work will provide an even more sophisticated unit that will be available commercially in the near future. It is both an economical and an expedient cutting process relative to other methods. A material cut blade wear factor of 20 to 1 was reported for a 0.91 m (36-in.) diameter blade. It is expected that as blade diameters increase, associated costs could possibly be reduced.

The prototype Arc Saw machine was built to rotate a 0.91 m (36-in.) diameter, 6.35 mm (1/4 in.) thick copper or mild steel blade with an 11.2 kW (15 hp) motor. Blade speed was approximately 880 rpm and blade cooling was accomplished by a water spray system installed in the blade guard. However, cutting can be accomplished remotely either in air or under water while permitting automatic positioning and tracking of the saw blade during cutting operations. All equipment, with the exception of the blades, are commercially available items (modifications are necessary, of course). The blades can be made in any well-equipped machine shop. The arc saw is shown schematically in Figure F.3-2.

The modified power source consists of a 7500 A, 80 V open circuit, 50 V operating DC power supply of the constant current, constant voltage type. The controls include a recorder to monitor voltage, amperage, blade travel, servo current and the necessary servo controller to permit automatic positioning and tracking of the saw blade during cutting operations.

Low voltage, high amperage current is applied through the arc saw blade to the grounded work piece, melting the base metal while producing a narrow kerf. Work piece position changes or vibrations, which would normally bind any other type of saw, merely slows the arc saw's cutting speed with a resultant wider kerf being opened to accommodate the shifting motions. This is due to the "electrical leading edge" being transferred from the narrow tip

(5) G. A. Beitel, and Max P. Schlienger, Arc Saw Testing, ARH-LD-147, Atlantic Richfield Hanford Company, Richland, WA and Schlienger, Inc., San Rafael, CA, July 1976.

- 1 BEARING ATTACHED TO, BUT INSULATED FROM ARC BLADE
- 2 BRUSHES IN CONTACT WITH SHAFT WHICH IS CONNECTED DIRECTLY TO ARC SAW BLADE
- 3 COPPER OR STEEL ARC SAW BLADE
- 4 BLADE GUARD WITH WATER SPRAY DEVICE FOR COOLING

FIGURE F.3-2. Arc Saw Schematic Diagram

of the blade temporarily to spots on the sides of the blade coming into contact momentarily with the work piece. The difference of potential is thus shifted to the lateral areas thus producing the wider kerf. As a result, the saw can cut through a variety of materials, pieces, and loose tubes without jamming. Scaling up the saw blades to diameters of 2 to 3 m (9 to 10 ft) for cutting larger diameter HXs, tanks, etc., appears to be feasible.

F.3.3 Tools for Concrete Surface Removal

Many instances will occur throughout the facility where concrete surfaces are contaminated to a depth such that surface washing with decontamination solutions is ineffective, and removal of the surface layers is the only effective decontamination method.

Modified rock-splitting tools have been developed to facilitate this type of decontamination effort. An example of such a device is illustrated in Figure F.3-3. The tool consists of a split collar with a sharp triangular

FIGURE F.3-3. Hydraulic Concrete Spalling Device

ridge around the circumference, mounted on a shaft which has a tapered end. The tapered portion of the shaft is pulled into the split collar by a hydraulic cylinder, forcing the sharp ridges into the concrete, fracturing the concrete locally. The device is fully portable, weighing about 10 kg (20 lb) (without the hydraulic power unit which supplies the moving force to the hydraulic cylinder), and has dimensions of about 0.41 m (16 in.) in length and about 0.005 m² (8 in.²) area.

A 25 mm (1-in.) diameter hole, about 100 mm (4 in.) in depth, is drilled into the concrete surface. The splitting tool is inserted to the bottom of the hole and the hydraulic actuator forces the sharp edges of the tool into the surrounding concrete, causing the concrete to fracture locally and spall away from the surface. The splitting tool and hydraulic actuator is small and lightweight, and can readily be adapted for remote operation.

For rapid removal of large surface areas, the devices can be ganged together with a corresponding set of concrete drills and operated as a unit.

The spacing between holes and the pattern of the holes (square or triangular array) are important parameters in the effectiveness of this technique for removing concrete surfaces. Arrangement of the concrete drills and the splitting tools into a fixed-geometry array would assure a relatively uniform removal pattern.

F.3.4. Linear-Shaped Charges

An economic and expedient method for equipment or piping to be reduced to a physical size that can be further processed or packaged for storage is via remote self-contained explosive charge units; thus, eliminating or at least minimizing exposure of personnel where high radiation levels are involved. The use of shaped charges, i.e., the perforation and cutting of solids by high-velocity and ultra-high-velocity particles under explosive attack, has been used extensively in the last 15 to 25 years.^(6,7) Recently, such methods have been used for the removal of activated components and contaminated systems (fuel pool liners) that are not amenable to conventional removal techniques.⁽⁸⁾

Shaped charges usually consist of a tubular housing, one side forming an inverted "V". This tube is filled with an explosive cyclotrimethylenetrinitramine, commonly called RDX. Figure F.3-4 shows a typical linear shaped charge cross section. The housing is generally copper but other materials such as aluminum and lead can be used. The melting and subsequent fusing of the housing material with the base material being cut and the ragged edges left may make the electropolishing of those edges very difficult; therefore, in situ decontamination either chemically or mechanically prior to explosive cutting is recommended if unnecessary waste of strategic materials is to be minimized.

The charges are usually equipped with detonators that are activated using a primacord lead having a high explosive core of pentaerythritol tetranitrate, commonly referred to as PETN, which is initiated with an electric blasting

(6) DuPont Blaster's Handbook. 15th Edition, 1969.

(7) M. A. Cook, The Science of High Explosives. American Chemical Society Monograph Series, Reinhold Book Corporation, New York, NY 1968.

(8) K. D. Kok, V. D. Linse and S. J. Basham, Jr., "Use of Linear-Shaped Explosive Charges for Reactor Dismantling." Trans. Am. Nucl. Soc. 22:635, 1975.

FIGURE F.3-4. Typical Linear-Shaped Charge Section

cap. The principle of the shaped charge is that as the detonation wave collapses the inverted "V," the liner material becomes a jet of extremely hot metal particles traveling at very high velocity.

Automatic clamp-on features would be the most desirable and are available commercially. Charges so designed would pose fewer problems in placement, handling and detonating. The number of charges that can be detonated is limited only by the blast effect on nearby equipment. Shock wave and fragment damage can be reduced appreciably by having blast curtains, metal or other material in the vicinity of the detonation. The curtains disrupt the shock wave and stop or slow down the fragments so any damage would be minimal.

The feasibility and demonstration of using linear-shaped charges for explosive cutting of such diverse shaped objects as the following has been documented:⁽⁹⁾

- 0.61 m O.D. x 16 mm wall stainless steel pipe
- 100 mm x 100 mm x 10 mm thick angle iron

⁽⁹⁾ J. D. Kaser and J. O. Vining, Size Reduction of Large Contaminated Process Equipment Using Explosive Shaped Charges. BNWL-192, Battelle, Pacific Northwest Laboratories, Richland, WA.

- 25 mm diameter solid stainless steel bar
- 152 mm x 10 mm thick web "H" beam.

Personnel could be instructed in the use and handling of these explosives, or the operation contracted to experts in the field of explosives.

F.3.5. Portable Filtered Ventilation Enclosure

Many of the cutting operations required for disassembly of the facility have the potential for putting contaminated material into the air. To minimize the personnel hazard and the potential for widespread contamination of the work areas, portable filtered ventilation enclosures are assumed to be used to confine and collect particulate material produced by the cutting processes. A typical unit is illustrated in Figure F.3-5. A large squirrel-cage blower is coupled with a high efficiency particulate air (HEPA) filter preceded by a fiberglass roughing filter, all of which are mounted on a wheeled cart. A Flexible duct couples the cart unit to the enclosure unit which collects the materials being emitted at the cutting location. Another fiberglass roughing filter is installed in the outlet of the enclosure. The enclosure may have any form, whatever shape best performs the required function at a particular location. A simple rectangular open-faced box will suffice for most applications.

FIGURE F.3-5. Portable Filtered Ventilation Enclosure

Radiation detection devices are used to monitor the buildup of radioactive material on the filters. A differential pressure guage is installed across the HEPA filter to monitor the pressure drop across the filter which increases as particulates build up on the filter. The filters are changed when either the dose rate from the collected radioactive particles or the differential pressure across the HEPA filter reach predetermined levels.

F.3.6. Electropolishing Decontamination Station

Reduction of alpha contamination levels from in excess of 1,000,000 d/m per 100 cm² to background within 10 minutes has been demonstrated using electropolishing techniques to show the utility and versatility of this decontamination method on a wide variety of alloy compositions and geometries.⁽¹⁰⁾

Electropolishing is an electrochemical process used to produce a smooth, polished surface. A two-fold result occurs: first, the dissolution of the surface material (along with the contaminants) and second, a progressive smoothing of the surface of the material. This smoothing effect prevents recontamination from the electrolytic solution and allows a simple water rinse as the final step in producing a contamination-free surface.

The process shown in Figure F.3-6, offers the following advantages over other decontmination processes:

- it can be used for a wide variety of metals, alloys, component geometries, and types of contamination;
- it substantially reduces personnel exposure as well as expense;
- it is rapid and effective;
- it contributes to a significant reduction in the amount of solid waste requiring expensive controlled storage, transportation and burial; and
- it permits the reclamation, reuse, and/or recycling of valuable materials.

(10) H. W. Arrowsmith, J. L. Hooper, L. A. Charlot and R. P. Allen, Electropolishing as a Decontamination Process: Progress and Application, PNL-SA-6858, Battelle, Pacific Northwest Laboratories, Richland, WA, April 1978.

FIGURE F.3-6. Schematic Diagram of Electropolishing Cell

An electropolishing decontamination station is assumed to be installed in the area formerly occupied by the New Fuel Storage Racks, adjacent to the Spent Fuel Pool as illustrated in Figure F.3-7. One tank contains the electrolyte and appropriate fixtures. Additional tanks would be used for rinse water plus a final hose rinsing. A large constant current, constant voltage DC power supply capable of providing up to 10,000 A at about 10 V is required to provide current densities at the anode surface of 500 to 1500 A/m². A recirculation system is required to provide constant mixing of the electrolyte, on-line filtration of suspended solids, and on-line removal of dissolved metal ions by ion exchange. The ion exchange column would be connected so as to permit back-flushing and regeneration, with the regeneration solutions piped into the Liquid Radioactive Waste System for disposal. Sufficient air sweep over the

electrolyte tank must be maintained to remove the hydrogen gas which is evolved due to electrolysis during the cleaning process. A power-interrupt switch activated upon loss of the air sweep is needed to prevent operation of the electropolisher and the accumulation of explosive quantities of hydrogen gas when the ventilation system is inoperative. Cost of such a system would be in the \$100,000 range.

FIGURE F.3-7. Suggested Arrangement of Electropolishing Station

F.3.7. Reactor Piping Jumper

Decontamination of the reactor coolant system piping, steam generators, pressurizer, etc., is a significant item on the critical path for dismantling the reactor containment building. It is, therefore, desirable to be able to circulate the decontamination solutions and rinses through all four loops of the reactor coolant system simultaneously, and, at the same time, minimize the volumes of solution and rinse water which have to be evaporated and solidified for disposal. The Reactor Piping Jumper, whose conceptual design is presented in this section, will satisfy both requirements.

The basic features of the jumper assembly are illustrated in Figure F.3-8. Only two sealing surfaces are required to enclose the four inlet and the four outlet nozzle areas of the pressure vessel. These surfaces are sealed using inflatable seals around the periphery of the upper and lower plates of the jumper. The annular region of the jumper is sized to handle the flows produced by the primary pumps, while keeping the volume of the annulus to a minimum, adding only about 2 percent to the external system volume while excluding the volume of the pressure vessel from the system. The unit is supported from the vessel flange from which the core barrel assembly formerly hung.

FIGURE F.3-8. Reactor Piping Jumper Installation

The pressure vessel head is installed and the water level in the vessel is raised to cover the jumper assembly. The vessel interior is tied to the pressurizer to eliminate any pressure differential across the seals. The pressure in the inflatable seals is maintained slightly 14 to 21 Mg/m² (20 to 30 psi) above the pressure in the rest of the system. Automatic pressure controllers maintain the pressure differential while the primary system and the vessel interior are being pressurized from 0 to 281 Mg/m² which is required for primary pump operation.

Use of the jumper assembly would be limited to the situation where a decontamination procedure is employed which involves multiple decontamination solutions and flushes of the reactor coolant system, to minimize the volume of liquid to be evaporated. For example, the reactor coolant system, exclusive of the reactor vessel, holds about 174 to 178 m³ (46,000 to 47,000 gal). The reactor vessel after removal of fuel, holds another 174 to 178 m³. Thus, in a three-step decontamination procedure, a total of six system volumes would have to be handled. Using the jumper assembly, a total volume of about 1100 m³ (290,000 gal) would have to be evaporated. Without the jumper, twice that amount would be involved.

When EDTA/citric acid/oxalic acid solution is used for decontamination, only two system volumes are involved. Thus, less than 757 m³ (200,000 gal) would have to be evaporated, and the jumper is probably not cost-effective.

F.4 QUALITY ASSURANCE

To meet the QA regulations imposed upon it, this complex project will require QA planning from the earliest stages. As each detailed procedure is developed, the QA portions should be included.

Current regulations and guides in place are:

- Regulatory Guide 1.58, Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel for the Construction Phase of Nuclear Power Plants^(a)

(a) Although not written specifically for the decommissioning phase, the QA principles and objectives found herein would, however, apply to all activities of decommissioning.

- Regulatory Guide 1.88, Collection and Storage of Nuclear Power Plant Quality Assurance Records
- 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants^(a)
- MIL-Q-9858A (Military QA Standard), for which a handbook (Handbook 50) helps in the interpretation of QA requirements and guides the user
- ANSI N45.2, Quality Assurance Program Requirements for Nuclear Power Plants
- ANSI N45.2.6, Qualifications of Inspection, Examination and Testing Personnel for the Construction Phase of Nuclear Power Plants^(a)

The QA supervisor is assumed to report directly to the Decommissioning Superintendent. Thus, the QA is organizationally independent of the project staff, up to the highest possible level.

F.4.1 Method Review

Whenever a new procedure is written, there would be documented peer reviews. The reviews assure that the planning includes suitable controls for application to physics, radiation, safety, accident, security, thermal, stress, QA/QC, and hydraulic analyses. The controls may include other checks, tests, and formal design reviews as necessary. Other aspects of the planning that should be reviewed are reliability criteria, the choice of test methods, and the choice of processes or materials. Also receiving documented review is the conservatism in margins of safety established for procedures, equipment, or structures.

F.4.2 Procurement Document Control

The project's suppliers, including the specialty of fixed-price contractors and the shipping cask vendors, should be selected from a QA list of Qualified Suppliers. QA inspections of the vendor's plant may be needed to place new firms on the qualified list.

^(a) Although not written specifically for the decommissioning phase, the QA principles and objectives found herein would, however, apply to all activities of decommissioning.

Procurement Document Control is a QA procedure to verify that any QA requirements for specified hardware or services are identified in procurement documents of purchase orders. This includes references to applicable drawings, specifications, standards, codes, regulations, required records, test data, certifications, and qualifications of personnel or procedures. Special handling, shipping, or packaging requirements should also be identified to vendors. This control should be applied to any supplier of the specific devices and equipment vital to the project, such as environmental monitoring equipment, safety and shielding devices, radiation detection equipment, chemicals, and explosives.

F.4.3 Work Instructions, Procedures, and Drawings

Formal documentation of work instructions, procedures, and drawings, and the sequence of action for preparation, review, approval, and control of information is required. The people responsible for the sequence are clearly identified. These procedures verify the proper completion of activities or steps; for example, the disassembly sequences which assure that vital services remain intact. Other applications include the valve-setting checklists, the setting of charges, and the checking of fog, blasting mats, and ventilation seals before detonations. The latest revision of drawings, plans and work instructions are required to be at the work place. Also included are work instructions for vendors to review hardware for quantitative (dimensions, tolerances, or limits) and for qualitative (finish, workmanship) acceptability criteria. Changes in the planning receive equivalent scrutiny and approval as original planning.

F.4.4 Document Control

Document Control verifies that all essential documentation receives review, approval, release, change and distribution control. There will be a procedure to purge the system of obsolete or erroneous information.

F.4.5. Identification and Control of Procured Items

A program of supplier surveillance or source inspection for vendors of critical items must be established. (Also see the Inspection and Test Section.) Records, such as certification of conformance, will be required from vendors

to show they meet the QA requirements. Receiving inspection of purchased materials or items must be performed. Items must be sufficiently identified to allow proper selection for use, completion of processing, or segregation of nonconforming items. The shelf life of chemicals must be clearly stated.

F.4.6. Special Processes

This project will have several tasks involving new or very unusual work procedures; for example, large-scale chemical decontamination work, and the use of explosive charges. Special provisions must be made to identify and control any unusual processes, using reviews, procedures, and audits.

F.4.7. Inspection and Test

All items produced for the project will be properly inspected or verified. Inspections in the vendor's plant may be necessary for some items. Adequate inspection instructions must be prepared, and inspection records should be prepared and retained.

Verification procedures will assure inspection of each essential operation, as well.

F.4.8. Calibration

All measurements require the use of properly calibrated equipment. Therefore, a system of calibration control is used to assure instrument reading validity. The accuracy requirements and tolerances for the project measurements are identified. Examples of equipment with calibration requirements include radiation monitoring devices, portable radiation detection equipment used in dose rate of shipping package surveys; and the devices used to release cleaned or uncontaminated scrap. Instruments monitoring vital services will probably have calibration procedures already in place. For this equipment, the existing program should be maintained.

F.4.9. Shipping and Receiving

Special instructions for handling, packaging, shipping, receiving, storing and marking of components for the project must be prepared. Included are radioactive shipments which require written QA procedures for each package type, specifying all steps in the package inspection, preparation, loading, and closing. Signed check lists shall be used. Seals must

be carefully verified on contaminated units for which no additional outside packaging is used.

F.4.10. Operating Status

Sufficient identification must be applied to prevent the inadvertent operation of systems no longer functional, or the use of discrepant items. Restricted use may be necessary in many cases for reactor systems, such as pumps, which may not have been fully removed from service.

F.4.11 Nonconformance and Corrective Action

All nonconformances are identified, and the records are retained as part of the project records.

F.4.12 Special Quality Records

An index of records and their required retention periods will be maintained. These records will include operating logs, results of reviews, inspection, orientation, training and qualifications of personnel, procedures, drawings, specifications, criteria documents, procurement documents, shipping records, nonconformance and corrective action reports, and audit reports. Extensive records of dose rate surveys will be needed.

F.4.13 Quality Auditing

Regular audits will verify that each of the QA procedures is being followed. Examples include witnessing radioactive shipments, calibration audits, verification that the correct procedures are at all work locations, vendor inspection, receiving and storage inspections, and audits for complete QA procedures and record retention.

F.5 ENVIRONMENTAL SURVEILLANCE PROGRAM

The levels of surveillance needed are different for the dismantlement operations and for the continuing care period of Safe Storage. The scopes of the necessary programs for these cases are outlined in the following sections.

F.5.1 Environmental Surveillance Program During Dismantlement

The objectives of an environmental surveillance program for operating nuclear facilities are generally recognized as valid and that rationale can be applied to facilities undergoing decommissioning as well. The following objectives are taken from ICRP No. 7.⁽¹¹⁾

- Detection of sudden changes and evaluation of long-term trends of concentration in the environment, with the intent to detect failure or lack of adequate control of releases and to initiate appropriate actions.
- Assessment of the actual or potential exposure of man to radioactive materials or radiation present in his environment, or the estimation of the probable upper limits of such exposure.
- Determination of the fate of contaminants released to the environment, especially with the intent of detecting previously unconsidered mechanisms of exposure.
- Demonstration of compliance with applicable regulations and legal requirements concerning releases to the environment.

"For protection of the public, these varying purposes are listed in order of relative importance. However, the time lag and generally more diffuse effects that are inherent in environmental measurements make primary reliance on any environmental measurement as an action signal unwise. With the exception of long-term accumulations of contaminants from source terms too dilute to be measured, all environmental measurements should be considered as supplementary insofar as action guides are concerned, with primary (and possibly secondary) action warnings from facility interior measurements."⁽¹²⁾

⁽¹¹⁾ International Commission on Radiological Protection. Principles of Environmental Monitoring Related to the Handling of Radioactive Materials. ICRP Pub. 7., Pergamon Press, Oxford, 1966.

⁽¹²⁾ J. P. Corley, et al., Environmental Surveillance for Fuel Fabrication Plants, BNWL-1723, Battelle, Pacific Northwest Laboratories, Richland, WA, April 1973.

Basic radiation exposure criteria and radioactive effluent release criteria applicable to population exposure are given in 10 CFR 20, sections 20.1, 20.105, 20.106, and 20.303. For nonradioactive contaminants, consideration must be given to applicable standards such as Water Quality Criteria⁽¹³⁾ and ambient air quality standards listed by the Environmental Protection Agency.⁽¹⁴⁾ In any case, local or state air quality criteria presumable would apply on a site specific basis.

Additionally, the interfaces of the environmental monitoring program between the plant owner, the appropriate state agencies, and the Environmental Protection Agency (EPA) should be stipulated in the application for the possession-only license as may be required by 10 CFR, Part 50, Section 50.59 (d) and 50.82 (a).

A suggested minimum program of environmental radiological surveillance to be conducted outside the plant for the purpose of establishing population dose is shown in Table F.5-1. This minimum program will continue until all fuel and source material is shipped from the site. At that time, approximately one year after reactor shutdown, the program would be reduced in scope.

The analytical detection limits given in Table F.5-1 are based on the practicability of routine radioactivity measurement techniques, and in all cases should be sufficient to quantify radionuclide concentrations that would result in conservatively estimated whole body and individual organ doses in the range of 1 to 25 mrem/year depending upon the values derived for the disposition criteria for the specific PWR-originated nuclides. These nuclides would be identified upon completion of the operational monitoring program. They would then be defined in terms of concentrations in environmental media at levels which, if sustained, would result in doses in excess of the disposition criteria limits. The derivation of these levels would correspond with the site-specific method of assessing offsite doses from radioactive materials in estimated gaseous and liquid effluents (if any), and would be consistent with the recommendations

(13) Federal Water Quality Administration. Guidelines for State Water Quality Legislation.

(14) Code of Federal Regulations. Title 40, Protection of Environment, "National Primary and Secondary Ambient Air Quality Standards," Part 50, Superintendent of Document, GPO, Washington, DC, 1972.

TABLE F.5-1. Recommended Basic Environmental Monitoring Program for the Period of Active Decommissioning Operations

Sample Type	Frequency	Analysis	Analytical Detection Limit ^(a)	Number of Sampling Stations	
				Onsite	Offsite
Terrestrial Samples					
Air particulate	Weekly	Gross Beta	0.002 pCi/m ³	2	4
	Monthly composite	Gross Alpha	0.002 pCi/m ³	2	4
		Gamma Scan ^(b)	0.3 pCi/m ³ /isotope	2	4
Air radioiodine	Weekly ^(f)	¹³¹ I	0.1 pCi/m ³	2	4
Direct radiation	Quarterly	TLD ^(e)	1.25 mrem/quarter increase	8	10
Rainfall	Monthly	Gross Beta	0.5 pCi/l	1	2
		Tritium	1000 pCi/l	1	2
Soil	Semiannually	Gamma Scan ^(c)	25 pCi/l/isotope	1	2
		⁸⁹ Sr, ⁹⁰ Sr	0.01 pCi/g (dry)	3	3
		Gamma Scan	0.1 pCi/g/isotope (dry)	3	3
Vegetation	Semiannually	⁸⁹ Sr, ⁹⁰ Sr	5.0 pCi/kg (wet)	2	3
		Gamma Scan	50 pCi/kg/isotope (wet)	2	3
Animals	Semiannually	⁸⁹ Sr, ⁹⁰ Sr	5 pCi/kg (wet)	2	3
		Gamma Scan	50 pCi/kg/isotope (wet)	2	3
Milk	Monthly	⁸⁹ Sr, ⁹⁰ Sr	1.0 pCi/l	0	5 ^(d)
		¹³¹ I ^(f)	0.5 pCi/l	0	5 ^(d)
		Gamma Scan	50 pCi/l/isotope	0	5 ^(d)
Aquatic Samples					
Surface water	Monthly	Gross Beta	0.5 pCi/l	3	3
		Tritium	1000 pCi/l	3	3
		Gamma Scan ^(c)	25 pCi/l/isotope	3	3
Well water	Quarterly	Gross Beta	0.5 pCi/l		3
		Tritium	1000 pCi/l		3
		Gamma Scan ^(c)	25 pCi/l/isotope		3
Bottom sediment	Semiannually	Gamma Scan	0.1 pCi/g/isotope (dry)	3	3
Vegetation	Semiannually	⁸⁹ Sr, ⁹⁰ Sr	5 pCi/kg (wet)	2	3
		Gamma Scan	100 pCi/kg/isotope (wet)	2	3
Shoreline soil	Semiannually	Gamma Scan	0.1 pCi/g/isotope (dry)	0	3
Fish	Semiannually	⁸⁹ Sr, ⁹⁰ Sr	5 pCi/kg (wet)	2	3
		Gamma Scan	100 pCi/kg/isotope (wet)	2	3

(a) Analytical detection limit is defined here as that concentration that is three standard deviations above the average concentration in a blank sample, and assures accuracy limits of +25%.

(b) To be performed if gross beta exceeds 0.1 pCi/m³.

(c) To be performed if gross beta exceeds 10 pCi/l.

(d) Includes one sample from a local milk processor.

(e) TLD: thermoluminescent dosimeter.

(f) ¹³¹I analyses needed only for first month following shutdown.

of the Federal Radiation Council⁽¹⁵⁾ and the International Commission on Radiological Protection.⁽¹⁶⁾ The environmental radiological monitoring program would integrate fully with any ongoing programs of the state where the PWR is located. Quality assurance would be achieved, in part, by the site-specific state QA program and by participation in the EPA analytical quality assurance program. Sample collections and radiation measurements required to meet the schedule suggested in Table F.5-1 would likely be conducted at locations and in amounts previously established during the operational lifetime of the reference PWR. The program and any changes thereto would be reportable items to the NRC and to other appropriate regulatory agencies, as environmental technical specifications, in accordance with the terms of the possession-only license. Quality control mechanisms should be exercised for all procedures involved.

Environmental monitoring during the 4-year period of dismantlement activities will be carried out by the Health Physics group of the decommissioning organization as a part of their normal duties.

F.5.2 Environmental Surveillance Program For The Continuing Care Period

An abbreviated version of the environmental monitoring program conducted during plant operation will be carried out during the continuing care period. There is no intent to provide a surveillance program adequate for all potential nonroutine or accidental releases, although the proposed program will be useful in evaluating most lapses of control. For the emergency situation involving releases from events such as fire or malicious acts that may require prompt emergency actions to minimize public risk, special surveillance requirements would apply. The purpose of the program is to identify and quantify any releases of radioactivity to the environment.

⁽¹⁵⁾ Federal Radiation Council, Background Material for Development of Protection Standards, Report No.2, 1961.

⁽¹⁶⁾ International Commission on Radiological Protection, Report of Committee II on Permissible Dose for Internal Radiation, Publication No. 2, 1959.

Quality control mechanisms should be exercised for all procedures involved. The surveillance system operation should be checked at least quarterly. Instrument calibrations should be run at least quarterly and following maintenance. Remedial measures will be taken as needed. Sampling quality control may necessitate temporary reinstatement of some aspects of the facility's former procedures, e.g., multiple sampling, sample splitting, and varying sampling locations. Similar analytical quality assurance procedures such as treating audit samples of known concentrations could be expected to be instituted.

Changes in background levels, in environmental accumulations (e.g., fallout from nuclear weapons testing), and especially in land usage and population distribution may, over a period of years, justify modifications in the post-decommissioning surveillance program. Experience will indicate, and the possession-only licensee should be alert for, relationships between local environmental results which would reasonably permit reduction of the program.

The entire program should be reviewed at the following post-decommissioning milestones:

- After all fuel and source material have been shipped from the site--approximately 12 months after plant shutdown.
- Ten years after plant shutdown.
- After 10 half-lives of ^{60}Co decay--approximately 53 years; economic advantages of further decommissioning effort would be ascertained by the owner. Environmental monitoring conceivably could be eliminated at that time.

The suggested minimum program of environmental radiological surveillance to be conducted outside the plant for the purpose of establishing population dose, shown in Table F.5-1, will continue until all fuel and source material is shipped from the site. It can then be reduced in scope. As experience is gained during the continuing care period, further reductions and/or changes could be expected.

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APPENDIX G

IMMEDIATE DISMANTLEMENT ACTIVITIES, MANPOWER, AND COSTS

The information contained in this appendix forms the bases for the cost and occupational radiation exposure estimates given in Sections 9 and 10 of Volume 1. The procedures given in Section G.1 and the facility description given in Section 7 and Appendix A provide the basis for the development of the detailed work schedules, manpower requirements and direct salary costs for immediate dismantlement presented next. Using the detailed work schedules and manpower estimates in Appendix G.2 and the radiation dose rate data from Section 7 and Appendix C, the detailed estimates of accumulated occupational radiation exposure are developed.

The detailed cost estimates for chemical decontamination of the RCS and CVCS, for disposal of radioactive materials, and for environmental surveillance and quality assurance programs are developed, and the details of the cost estimate for demolition of the decontaminated structures are presented.

G.1 DISASSEMBLY PROCEDURES

Most of the disassembly procedures presented herein are generic in nature, and apply to the removal of a pump, or a heat exchanger, or an ion exchange column, etc., without specifying which particular unit of given type is being removed. Specific procedures are also presented for removal and sectioning of major equipment items such as reactor vessel and internals, the steam generators and the pressurizer. The general conditions in effect at the start of dismantlement of any system are given below:

Initial Conditions

1. Chemical decontamination of the system involved has been completed, with the system flushed, drained and dried.
2. Radiation surveys of the system involved have been completed.

Safety Requirements.

1. Essential systems and services (Table 9.1-2) are in operation.
2. Current safety certifications are in place for all lifting equipment.

3. A portable ventilation enclosure with filtered exhaust and/or certified fresh air breathing equipment will be used when required on cutting operations in air.

G.1.1 Removal of Pumps, Valves, and Piping

This procedure covers the removal of pumps, valves, and piping, that have significant potential value as salvage or scrap. In general, connections to pumps and valves will be cut as close to the unit as feasible. Piping sections will be cut into straight sections not exceeding 2.1 m in length, with bends and tees removed to facilitate final internal decontamination.

1. The basic set of Initial Conditions and Safety Requirements are satisfied.
2. All electric and pneumatic power sources and sensing instrumentation on the system are deactivated, disconnected, and removed.
3. Install the ventilation enclosure around the cutting location(s) and establish adequate air flow to prevent spread of contamination.
4. Install supports, lifting devices, etc., as required to keep the unit and piping stationary during cutting operations.
5. Cut the piping connections using plasma torch, arc saw, linear-shaped charges, or other device as may be appropriate.
6. Seal the pipe openings with plastic, wipe down or place the removed unit in a plastic bag, depending on exterior contamination level, and transport to the Final Decontamination Station for final decontamination effort.
7. Following final decontamination, monitor to determine if releasable for unrestricted use as defined in Section 8, place the unit in the salvage-scrap storage area. If not releasable for unrestricted use, package appropriately for disposal, as radioactive material.

G.1.2 Removal of Tanks and Pool Liners

This procedure covers the sectioning and removal of large tanks and pool liners where large areas of relatively thin-walled material are involved and where the material has significant potential value as salvage or scrap.

The cutting devices used may include arc saw, plasma torch, oxyacetylene torch, mechanical saws, and linear-shaped charges, as dictated by the field

situation and the material being cut. In general, the arc saw, the plasma torch, and the linear-shaped charges will be used on stainless steel.

The material will be cut into sheets no greater than 1.2 m x 2.1 m, to facilitate handling for decontamination and ultimate disposition. Mechanical guides will be used to permit semi-remote operation of the cutting devices since complete enclosure of the cutting areas may not be feasible.

1. The basic set of Initial Conditions and Safety Requirements are satisfied.
2. All electrical and pneumatic power sources and sensing instrumentation on the system have been deactivated, disconnected, and removed.
3. Install mechanical guides for semi-remote cutting operations, if needed.
4. Install ventilation enclosure around the traveling cutting device and establish adequate air flow to minimize spread of contamination.
5. Cut into sections and transport to the Final Decontamination Station for final decontamination effort.
6. Following final decontamination, monitor to determine if releasable for unrestricted use as defined in Section 8, place the material in the salvage-scrap storage area. If not releasable, package for disposal as radioactive material.

G.1.3 Removal of Heat Exchangers

This procedure covers removal of all heat exchangers (HX) except the Steam Generators. It is anticipated that decontamination of heat exchangers to release levels may not be possible, due to the large surface areas and many crevices, most of which cannot be successfully checked for contamination. Therefore, the units will be cleaned as well as possible, removed and sealed, and shipped to a disposal site. No external packaging should be required other than sealing all openings.

1. The basic set of Initial Conditions and Safety Requirements are satisfied.
2. All electrical and pneumatic power sources and sensing instrumentation on the HX unit have been deactivated, disconnected and removed.
3. Install pipe supports as needed for cutting operations.

4. Install cutting equipment, with ventilation enclosure if needed, to prevent spread of contamination.
5. Cut free from associated piping, install and seal weld covers over all openings.
6. Transport to the Cask Wash Pit for final external decontamination.
7. Ship to licensed burial site.

G.1.4 Suggested Methods for Removal of Containment Internals

Removal and segmentation of the reactor vessel and internals presents difficult and potentially hazardous tasks. The highly activated structures must be cut into segments suitable for packaging and shipment to a disposal site. Much of the work must be done under water or using remotely-operated devices to minimize personnel exposure. The components that must be segmented are given in Table G.1-1, together with a suggested method of cutting and information on contamination control needs. A summary of cutting data for this task is given in Table G.1-2.

G.2 SCHEDULES, MANPOWER REQUIREMENTS, AND DIRECT SALARY COSTS

Development of the dismantlement schedule and manpower requirements requires a certain amount of iteration to arrive at a reasonably optimum plan. The sequence in which the various systems must be disassembled is determined, and the manpower and time frame for each system is estimated. The job sequences are then arranged to require a relatively constant-sized work force. The resultant schedules, manpower requirements and direct salary costs for the major job sequences are presented in the following sections.

G.2.1 Chemical Decontamination

Chemical decontamination of the RCS, CVCS and inter-tied systems is the first major step following the defueling of the reactor. In the opinion of the author, chemical decontamination of these systems is a necessary step even if the current decommissioning plan calls for placing the facility in Safe Storage for an extended period of time, since completing the decontamination step removes most of the internal radioactive contamination and leaves all options open for changing the decommissioning plan at a later date.

TABLE G.1-1. Postulated Methods for the Removal of Radioactive Material from Containment Building

Component	Prerequisite	General Method of Segmentation	OPERATIONAL CONDITIONS
Upper Support Assembly (Vessel Internals)	Containment control envelope required	Cut into 12 segments using ARC saw or "jet-cutter explosives."	• Underwater • Remotely
Lower Core Barrel	Containment control envelope required	Cut into 64 segments using ARC saw or "jet-cutter explosives."	• Underwater • Remotely
Upper Support Columns (48)	Containment control envelope required	Cut into 96 segments using ARC saw or "jet-cutter explosives."	• Underwater • Remotely
Guide Tubes (61)	Containment control envelope required	Cut into 122 segments using ARC saw or "jet-cutter explosives."	• Underwater • Remotely
Upper Core Barrel	Containment control envelope required	Cut into 10 segments using ARC saw or "jet-cutter explosives."	• Underwater • Remotely
Upper Core Plate	Containment control envelope required	Cut into 9 segments using ARC saw or "jet-cutter explosives."	• Underwater • Remotely
Neutron Pads (4)	Containment control envelope required	Cut into 12 segments using ARC saw or "jet-cutter explosives."	• Underwater • Remotely
Core Support Columns (36)	Containment control envelope required	Cut into 96 segments using ARC saw or "jet-cutter explosives."	• Underwater • Remotely
Lower Core Plate	Containment control envelope required	Cut into 9 segments using ARC saw or "jet-cutter explosives."	• Underwater • Remotely
Wide Baffles (4) (Vessel Internals)	Containment control envelope required	Cut each baffle into 6 segments using ARC saw or explosives.	• Underwater • Remotely
Medium Baffles (8)	Containment control envelope required	Cut each baffle into 3 segments using ARC saw or explosives.	• Underwater • Remotely
Small Baffles (16)	Containment control envelope required	Cut each baffle into 3 segments using ARC saw or explosives.	• Underwater • Remotely
Lower Core Forging	Containment control envelope required	Cut into 25 segments using ARC saw or explosives.	• Underwater • Remotely
Upper Tie Plate	Containment control envelope required	Cut into 9 segments using ARC saw or explosives.	• Underwater • Remotely
Lower Tie Plate	Containment control envelope required	Cut into 4 segments using ARC saw or explosives.	• Underwater • Remotely
Irradiation Baskets	Containment control envelope required	Cut into 8 segments using ARC saw or explosives.	• Underwater • Remotely
Secondary Core Support	Containment control envelope required	Cut into 4 segments using ARC saw or explosives.	• Underwater • Remotely
Instrument Guide Columns (25)	Containment control envelope required	Cut into 25 segments using ARC saw or explosives.	• Underwater • Remotely
Butt Type Columns	Containment control envelope required	Cut into 30 segments using ARC saw or explosives.	• Underwater • Remotely
Steam Generators (4)	Internal decontamination prior to removal. Contamination control envelope required.	Cut each generator into 8 segments by use of oxyacetylene torch. Weld on closure plates--decontaminate exterior surface.	• Contact cutting in air
Pressure Vessel Shell Section	Containment control envelope required	Cut into 71 segments using oxyacetylene torch and remote control manipulator.	• Cut in air • Remotely
Pressure Vessel Closure Head	Remove all vessel internals	Cut into 5 segments using oxyacetylene torch and remote control manipulator. Weld on closure plates--decontaminate exterior surfaces.	• Cut in air • Contact
Pressure Vessel Bottom Head	Remove all vessel internals	Cut into 9 segments using oxyacetylene torch and remote control manipulator.	• Cut in air • Remotely
Reactor Coolant Pump (4)	Internal decontamination required prior to segmentation	Separate the three sections of each pump unit. Weld closure plates onto each opening of the pump section--decontaminate exterior surface.	• Cut in air • Contact
Pressurizer	Internal decontamination required prior to segmentation	Cut into 8 segments using oxyacetylene torch.	• Cut in air • Contact
Regenerative Excess Letdown	Internal decontamination required prior to segmentation	Disconnect all pipe-weld or bolt closure plates over all openings.	• Cut in air • Contact
Containment Sump Pumps (2)	--	Remove all pipe at connection to pumps--load pumps into disposable containers.	• Cut in air • Contact
Reactor Cavity Drain Pump	--	Remove all pipe at connection to pumps--load pumps into disposable containers.	• Cut in air • Contact
Pressurizer Relief Tank	Internal decontamination required	Cut into 8 segments using portable plasma torch.	• Cut in air
Safety Injection System Accum. (4)	Internal decontamination required	Cut into 8 each segments using portable plasma torch.	• Cut in air • Contact
Primary Piping 8d to 356 mm O.D.	Internal decontamination required	Cut into 220 segments using portable plasma torch.	• Cut in air • Contact
Primary Piping 685 to 787 mm I.D.	Internal decontamination required	Cut into 54 segments using ARC saw or explosives	• Cut in air • Contact
Pressurizer Enclosure Concrete	--	Spall 76 mm to 152 mm from all surfaces to remove contamination. Use explosives, rockjacks or water cannon (remove approximately 102 m ³).	• Cut in air • Contact
Steam Generator Enclosure Concrete	--	Spall 76 mm to 152 mm from all surfaces to remove contamination. Use explosives, rockjacks or water cannon (remove approximately 202 m ³).	• Cut in air • Contact
Biological Shield Contaminated Concrete	Containment control envelope required	Remove approximately 367 m ³ by use of explosives	• Drill holes remotely
Reactor Cavity Liner Steel	Decontaminate prior to removal	Cut into segments, for packaging, using portable plasma torch. ~14,500 kg.	• Contact
Floors and Cavity Liner Plates--Steel	Decontaminate prior to removal	Cut into segments, for packaging, using portable plasma torch. ~139,000 kg.	--
Refueling Cavity--Steel	Decontaminate prior to removal	Cut into segments, for packaging, using portable plasma torch. ~17,200 kg.	--
Refueling Cavity--Concrete	--	Spall approximately 115 m ³ from surface.	• Contact
CRD Missile Shield--Steel	--	Cut approximately 11,000 kg of steel liner using portable plasma torch.	• Contact
Base Slab--Steel Liner	--	Cut approximately 54,400 kg of steel liner using portable plasma torch.	• Contact
Containment Spray Head Piping	Requires scaffolding	Cut approximately 90,700 kg of steel liner using portable plasma torch.	• Contact
Electrical and Instrument Equipment	Radiation instructions no longer required.	Disconnect or cut for packaging these components not salvageable.	• Contact
Ladders, Stairways, Floor Grating, Jib Cranes and Miscellaneous Lifting Devices	Usefulness expired	Remove, segment and package for burial those times not salvageable.	• Contact
Special Dismantling Tools and Equipment	Usefulness expired	Decontaminate and salvage or package for burial	Some may require remote handling
Atmospheric Filters and Ventilation Equipment	After only low-level contamination is left in building	Cut and package for burial.	• Contact
Containment Building Access Hatches	After only low-level contamination is left in building	Remove, decontaminate, salvage or bury.	• Contact
Containment Building Polar Crane	No longer needed	Decontaminate in place. Remove after top section of containment is demolished.	• Contact
Decontamination of Internal Surfaces of Containment Building to Unrestricted Release	--	Remove all contaminated surface by use of chemicals, chipping hammers or explosives.	• Contact

TABLE G.1-1. Postulated Methods for the Removal of Radioactive Material from Containment Building

TABLE G.1-2. Summary of Cutting Data for Removal of Contaminated/
Radioactive Containment Internals

Activity	Linear Meters of Cut	Nominal Material Thickness (mm)	Estimated		Postulated Cutting Time(a) (hr)	Postulated Linear Meters Cut Per hr
			Cut Under Water	Cut in Air		
1. Segment Vessel Internals	949	76	X		960	1.
2. Segment Pressure Vessel	240	216		X	692	0.36
3. Segment Steam Generators	306	152		X	545	0.56
4. Segment Reactor Coolant Pumps and Primary Piping	132	51		X	277	0.48
5. Segment Pressurizer-Relief Tank and Safety Inj. Systems	259	102		X	208	1.24
6. Segment HXs and Containment and Cavity Pumps	37	25		X	139	0.25
7. Remove Containment Internal Structures	1524	6		X	300	5.08
8. Remove Containment Spray Piping and Ventilation System	91	6		X	200	0.46

(a) Does not include preparation or equipment removal time.

It is unlikely that a chemical decontamination could be carried out without major renovation after the facility has been in Safe Storage for a few years, due to equipment deterioration. If a decision were made to dismantle after 5 to 10 years of Safe Storage, significant radiation exposures would be encountered if the plant had not been previously decontaminated.

The sequence and schedule of events for performing the chemical decontamination of the RCS, CVCS and inter-tied systems using an EDTA/oxalic/citric acid solution as described in Appendix F.1 is given in Figure G.2-1. The staff labor requirements and the resulting direct salary costs (from Table I.1-1, Appendix I) are also given.

G.2.2 Dismantlement of the Reactor Containment Building

The major effort for the decommissioning staff is the dismantlement of the Reactor Containment building. All of the neutron-activated materials and large amounts of the activated corrosion products are present in the containment building. Because of the high radiation dose rates from the reactor internals, much of the work is done underwater or using shielded, remotely-operated equipment. Based on the procedures outlined in Section G.1.4, the schedule sequence and manpower requirements given in Figure G.2-2 are developed, together with the estimated direct salary costs for performing the work.

G.2.3 Dismantlement of the Fuel Building

Dismantlement of the Fuel Building takes place in steps that depend upon the need for use of the various systems in the building for support of other decommissioning activities. It is assumed that the concrete surfaces beneath the stainless steel spent fuel pool liner are contaminated to a depth of about 50 mm, requiring the outer 75 to 100 mm of surface to be removed. This same assumption is made for all lined areas in all buildings. The inner surfaces of most of the process cell walls are also assumed to require removal. In some instances, when the wall is relatively thin, the entire wall is removed as contaminated material. Thus, the estimates for the volumes of low specific activity concrete that must be removed and shipped to a licensed burial site, may be conservatively large.

The schedule sequence and estimated manpower requirements to dismantle the Fuel Building are given in Figure G.2-3, together with the direct salary costs.

REACTOR
SHUTDOWN

MONTHS AFTER SHUTDOWN

EVENT DESCRIPTION		1	2	3	4	5
CHEMICAL DECONTAMINATION OF RCS AND CVC SYSTEMS STEPS						
1 COMPREHENSIVE PLANT RADIATION SURVEY (INCLUDES RCS AND CVCS IN THE REACTOR BLDG AND THE PLACEMENT OF TEMPORARY RADIATION MONITORS NEAR 'HOT SPOTS' FOR MONITORING EFFECTIVENESS OF CHEMICAL DECONTAMINATION)		<div>NOTE</div> <div>EACH COLUMN OF NUMBERS TO THE RIGHT OF THE CAPTION CREW PER SHIFT REPRESENTS THE NUMBER OF EACH CATEGORY OF DECOMMISSIONING WORKER REQUIRED PER SHIFT TO ACCOMPLISH THE WORK ITEMS SHOWN DIRECTLY ABOVE THE NUMBERS</div>				
2 MIX CHEMICALS (NONRADIATION ZONE WORK)						
3 INJECT CHEMICALS PRESSURIZE (HEAT AS REQUIRED) AND RECIRCULATE CHEMICAL SOLUTION AROUND-THE-CLOCK FOR SEVEN DAYS						
4 DRAIN SOLUTION (INCLUDING ALL 'DEADLEGS') TO CVCS HOLDUP TANKS IN THE FUEL BLDG						
5 START EVAPORATION AND SOLIDIFICATION OF THIS SOLUTION						
6 REFILL RCS AND CVC SYSTEMS WITH BORATED WATER FROM THE PRIMARY STORAGE TANK (LOCATED OUTSIDE THE REACTOR BLDG) AND PRESSURIZE 2 DAYS TO REFILL						
7 CIRCULATE THIS RINSE SOLUTION FOR TWO DAYS AROUND-THE CLOCK						
8 DRAIN RINSE WATER (INCLUDING ALL "DEADLEGS") TO CVCS HOLDUP TANKS IN THE FUEL BLDG						
9 CONTINUE EVAPORATION AROUND-THE CLOCK UNTIL JOB IS COMPLETED						
10 SOLIDIFY AND PACKAGE EVAPORATOR BOTTOMS LIQUIDS						
MANPOWER						
SHIFT ENGINEER	CREW PER SHIFT (a)					
CREW LEADER						
UTILITY OPERATOR						
LABORER						
H P TECH						
CRAFT SUPERVISOR						
CRAFTSMEN						
SHIFTS PER DAY						
TOTAL MAN DAYS/DAY						
SALARY COST						

NOTES

- (a) STEP 5 IS COVERED ON A 2-SHIFTS/DAY 2 MEN/SHIFT (4 MAN DAYS/DAY) STEP 6 IS COVERED ON A 2 SHIFTS/DAY BASIS 4 MEN/SHIFT (8 MAN DAYS/DAY) STEP 7 IS COVERED ON A 3-SHIFTS/DAY BASIS 2 MEN/SHIFT (6 MAN DAYS/DAY)
- (b) A DASH MEANS THAT CATEGORY OF DECOMMISSIONING STAFF ARE NOT INVOLVED IN THAT PARTICULAR JOB ITEM
- (c) THESE MANPOWER REQUIREMENTS ARE COVERED UNDER THE COMPREHENSIVE PLANT RADIATION SURVEY MANPOWER REQUIREMENTS

FIGURE G.2-1. Schedule Sequence, Manpower Utilization, and Salary Cost Estimates for Chemical Decontamination of the Principal Piping Systems

ASSUMPTIONS

- 1 CALENDAR MONTH - 22 WORKING DAYS
- 1 WORK WEEK - 5 CALENDAR DAYS
- 2 EIGHT-HR SHIFTS/WORKING DAY DURING DECOMMISSIONING
- 2 TEN-HR SHIFTS/DAY DURING DEFUELING AND DISABLING
- (a) PERFORMED BY OPERATIONS STAFF, UNDER THE OPERATING LICENSE
- (b) ON DUTY AND AVAILABLE AS REQUIRED. SUPERVISES MANY WORK ITEMS DURING THE TIME PERIODS GIVEN

FIGURE G.2-2. Schedule Sequence, Manpower Utilization, and Salary Cost Estimates for the Dismantlement of the Reactor Containment Building

REACTOR SHUTDOWN			MONTHS AFTER SHUTDOWN																																			
EVENT DESCRIPTION			1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
1. REMOVE CVC SYSTEM																																						
2. REMOVE CONCENTRATE HOLDING TANK SYSTEM																																						
3. REMOVE NEW FUEL STORAGE RACKS; INSTALL ELECTROPOLISHING SYSTEM																																						
4. REMOVE BORIC ACID SYSTEM (TANKS AND PIPING)																																						
5. REMOVE CCW SYSTEM																																						
6. REMOVE SFP RACKS, FUEL TRANSFER SYSTEM																																						
7. REMOVE SFP CLEANUP SYSTEM, SFP LINERS, AND CONTAMINATED CONCRETE																																						
8. FINAL RADIATION SURVEY																																						
DECOMMISSIONING MANPOWER																																						
NUMBER AVAILABLE																																						
SHIFT ENGINEER																																						
CREW LEADER																																						
UTILITY OPERATOR																																						
LABORER																																						
H.P. TECH.																																						
CRAFT SUPERVISOR																																						
CRAFTSMEN																																						
TOTAL																																						
TOTAL MAN DAYS/WORKING DAY																																						
SALARY COST																																						

NOTE:

EACH COLUMN OF NUMBERS TO THE RIGHT OF THE CAPTION CREW PER SHIFT REPRESENTS THE NUMBER OF EACH CATEGORY OF DECOMMISSIONING WORKER REQUIRED PER SHIFT TO ACCOMPLISH THE WORK ITEMS SHOWN DIRECTLY ABOVE THE NUMBERS.

0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
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NOTE:
EACH COLUMN OF NUMBERS TO THE RIGHT OF THE CAPTION CREW PER SHIFT REPRESENTS THE NUMBER OF EACH CATEGORY OF DECOMMISSIONING WORKER REQUIRED PER SHIFT TO ACCOMPLISH THE WORK ITEMS SHOWN DIRECTLY ABOVE THE NUMBERS.

ASSUMPTIONS:
1 CALENDAR MONTH = 22 WORKING DAYS
1 WORK WEEK = 5 CALENDAR DAYS
2 SHIFTS/WORKING DAY DURING DECOMMISSIONING

(a) 2.55 (56) MEANS 2.55 CALENDAR MONTHS
FOR A TOTAL OF 56 WORKING DAYS

FIGURE G.2-3. Schedule Sequence, Manpower Utilization, and Salary Cost Estimates for the Dismantlement of the Fuel Building

G.2.4 Dismantlement of the Auxiliary Building

The bulk of the radioactive waste treatment systems are located in the Auxiliary Building. Since these systems are required to be in service during much of the plant dismantlement effort, final dismantlement of the Auxiliary Building occurs rather late in the overall schedule. The schedule sequence and manpower utilization estimates are given in Figure G.2-4, together with the direct salary costs.

G.2.5 Dismantlement of Other Buildings

It is assumed that only a small portion of the turbine steam system is contaminated and is removed for burial. The bulk of the systems and equipment is assumed to be non-contaminated and is removed for salvage by the demolition contractor. A similar circumstance is assumed for the Condensate-Demineralizer Building, Control Building. The Warehouse and Administration Building are assumed to require only a radiation survey to verify that they are not contaminated. The schedule sequence and estimated manpower utilization for the dismantlement of these buildings is given in Figure G.2-5, together with the direct salary costs.

G.2.6 Summary of Manpower Requirements

The individual manpower utilization estimates, presented as parts of Figures G.2-1 through G.2-5, are collected and summarized in Table G.2-1. These data, together with the radiation exposure rate data from Section 7 and Appendix C are used to develop the detailed estimates of external occupational radiation exposure, presented in the next section.

G.3 ESTIMATES OF EXTERNAL OCCUPATIONAL RADIATION EXPOSURE FOR THE DISMANTLEMENT OF THE REFERENCE PWR

Detailed estimates have been developed for the external occupational radiation exposure that will be accumulated by the decommissioning workers during the dismantlement program. These estimates are based on a task by task analysis of personnel hours and expected radiation dose rates associated with each task, for all decommissioning personnel. Certain basic assumptions were made in developing these estimates. 1) Every effort is made to implement the philosophy of ALARA (As Low As Reasonably Achievable) when performing an

REACTOR SHUTDOWN			MONTHS AFTER SHUTDOWN																																																																																																																																																														
EVENT DESCRIPTION			1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36																																																																																																																											
1. DECONTAMINATION (BOTH CONTACT WORK AND INTERNAL FLUSHES)																																				1.86 (41) ^(a)																																																																																																																													
2. REMOVE SELECTED BUILDING INTERNALS (PLUGS, WALLS, PLATFORMS)																																				0.59 (13)																																																																																																																													
3. IX RESIN AND LIQUID SYSTEMS FILTERS REMOVAL																																							.45 (10)																																																																																																																										
4. REMOVE IX AND FILTER SYSTEM PIPING																																							1.09 (24)																																																																																																																										
5. REMOVE TANKS, PUMPS AND HX'S																																										3 (66)																																																																																																																							
6. REMOVE HVAC, FIRE SPRINKLER, AND MONORAIL SYSTEMS																																													2.14 (47)																																																																																																																				
7. REMOVE ELECTRICAL EQUIPMENT																																																			.68 (15)																																																																																																														
8. FINAL RADIATION SURVEY																																																						1 (22)																																																																																																											
DECOMMISSIONING MANPOWER	NUMBER AVAILABLE																																																																																																																																																																
SHIFT ENGINEER	2	CREW PER SHIFT																																		.17																																				.36																																				.15			.19			.09			.2																																				.17			0					
CREW LEADER	7																																			1																																				1																																				1			1			1			1																																				1			0					
UTILITY OPERATOR	7																																			0			0																																				0																																				0			0			0			0																																				0			0		
LABORER	18																																			1.8																																				1.77																																				.4			1.85			1.83			1.58																																				1			0					
H.P. TECH.	10																																			.97																																				1.04																																				.6			1.13			1.09			1.12																																				0.31			1					
CRAFT SUPERVISOR	2																																			0			0																																				0																																				0			0			0			0																																				0			0		
CRAFTSMEN	16																																			.79																																				2.43																																				.5			3.66			3.72			3.49																																				3.07			0					
TOTAL	62																																			9.46																																				13.21																																				5.3			15.66			15.46			14.78																																				11.1			2					
TOTAL MAN DAYS/WORKING DAY																																				9.46																																				13.21																																				5.3			15.66			15.46			14.78																																				11.1			2					
SALARY COST																																				\$ 40,911																																				\$ 18,179																																				\$ 5,833			\$ 39,212			\$ 105,924			\$ 77,727																																				\$ 17,732			\$ 4,184			TOTAL = \$ 309,702		

NOTE:
EACH COLUMN OF NUMBERS TO THE RIGHT OF THE CAPTION CREW PER SHIFT REPRESENTS THE NUMBER OF EACH CATEGORY OF DECOMMISSIONING WORKER REQUIRED PER SHIFT TO ACCOMPLISH THE WORK ITEMS SHOWN DIRECTLY ABOVE THE NUMBERS.

ASSUMPTIONS:

1 CALENDAR MONTH = 22 WORKING DAYS

1 WORK WEEK = 5 CALENDAR DAYS

2 SHIFTS/WORKING DAY

(a) 1.86 (41) MEANS 1.86 CALENDAR MONTHS FOR A TOTAL OF 41 WORKING DAYS

FIGURE G.2-4. Schedule Sequence, Manpower Utilization, and Salary Cost Estimates for the Dismantlement of the Auxiliary Building

FIGURE G.2-5. Schedule Sequence, Manpower Utilization, and Salary Cost Estimates for the Dismantlement of Other Buildings

TABLE G.2-1. Summary of Manpower Utilization Estimates

REACTOR SHUTDOWN			MONTHS AFTER SHUTDOWN																																			
REACTOR BLDG.			1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36
MANPOWER/SHIFT																																						
NO. AVAILABLE																																						
SHIFT ENGINEER (a)			1 0 1 → 1 → 1 → 1 1 1 1 1 1 → 1 1 1 0																																			
CREW LEADER			2 2 1 → 1 → 1 → 1 1 1 1 1 1 → 1 1 1 0																																			
UTILITY OPERATOR			3 2 4 → 2 → 2 → 3 2 2 2 2 2 → 2 2 2 0																																			
LABORER			7 2 4 → 4 → 5 → 7 7 7 7 7 → 7 7 7 0																																			
HEALTH PHYSICS TECH			4 4 3 → 2 → 3 → 2 3 2 3 2 → 2 2 2 2																																			
CRAFT SUPERVISOR			1 0 1/2 → .5 → 1 → 1 1 1 1 1 .5 → .5 5 0 0																																			
CRAFTSMEN			4 0 4 → 4 → 4 → 4 4 4 4 4 3 → 2 2 0 0																																			
TOTAL			62																																			
TOTAL MAN DAYS/WORKING DAY			44 20 35 → 29 → 34 → 34 38 36 38 36 33 → 31 31 26 4																																			
FUEL BLDG.																																						
MANPOWER/SHIFT																																						
NO. AVAILABLE																																						
SHIFT ENGINEER			0 0 0 0 → 0 0 0 0 0 0																																			
CREW LEADER			1 1 → 1 → 1 1 1 1																																			
UTILITY OPERATOR			1 1 → 1 → 1 1 1 1																																			
LABORER			2 2 → 2 → 2 2 2 2																																			
HEALTH PHYSICS TECH			1.75 1.25 → 1.12 1.25 1.25 1.21																																			
CRAFT SUPERVISOR (a)			0 0 0 0 → 0 0 0 0 0																																			
CRAFTSMEN			6 .96 → 1.22 1.25 2.07 1.25 1.71																																			
TOTAL			62																																			
TOTAL MAN DAYS/WORKING DAY			23.5 12.4 → 12.7 13 14.6 13 13.8																																			
AUXILIARY BLDG.																																						
MANPOWER/SHIFT																																						
NO. AVAILABLE																																						
SHIFT ENGINEER			0.17 → 0.36 .15 .19 0.09 → 0.2 → 0.17																																			
CREW LEADER			1 → 1 1 1 1 1 → 1 → 1																																			
UTILITY OPERATOR			0 → 0 0 0 0 → 0 → 0																																			
LABORER			1.8 → 1.77 0.4 1.85 1.83 → 1.56 → 1																																			
HEALTH PHYSICS TECH			0.97 → 1.04 0.6 1.13 1.09 → 1.12 → 0.31																																			
CRAFT SUPERVISOR			0 → 0 0 0 0 → 0 → 0																																			
CRAFTSMEN			0.79 → 2.43 0.5 3.66 3.72 → 3.49 → 3.07																																			
TOTAL			62																																			
TOTAL MAN DAYS/WORKING DAY			9.46 → 13.21 5.3 15.7 15.5 → 14.8 → 11.1																																			
MISC. BLDGS.																																						
MANPOWER/SHIFT																																						
NO. AVAILABLE																																						
SHIFT ENGINEER			2																																			
CREW LEADER			7																																			
UTILITY OPERATOR			7																																			
LABORER			18																																			
HEALTH PHYSICS TECH			10																																			
CRAFT SUPERVISOR			2																																			
CRAFTSMEN			16																																			
TOTAL			62																																			
TOTAL MAN DAYS/WORKING DAY			2 2 4 4 4 2 1 3 2 2 2 1 1																																			
(DOES NOT INCLUDE CONTRACTOR STAFF)																																						
SUMMARY (a, c) MAN DAYS/WORKING DAY BY OCCUPATION:																																						
SHIFT ENGINEER (a, d)			2 2 2 0 2 2.34 2.34 2 2 2 2 2 2 2.72 2 2 2 2 2 2 2.3 2.38 2.18 2.18 2.18 2.4 2.4 2 2 2 2 34																																			
CREW LEADER			2 4 6 2 6 6 4 4 2 2 4 2 4 2 4 2 4 2 2 5 5 6 6 6 4 4 3 2 2 2																																			
UTILITY OPERATOR			18 18 6 6 8 10 10 10 10 8 8 6 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 4 0																																			
LABORER			14 8 8 15.6 15.6 12 12 8 8 12 8 8 11.5 9 12 12 10 14 10 14 14.8 17.7 17.7 17.7 17.2 17.2 14 14 14 2																																			
H.P., Q.A., MISC. TECHS			4 4 8 11.5 6 10.5 10.5 8.5 8.25 6 6 6.5 4 4 6 4 6.5 6.5 6 8.42 4 6 6.2 9.3 8.18 8.18 8.24 7.24 6 6 6 6.62 1 1																																			
CRAFT SUPR.			2 0 1 1 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 1 1 1 1 1 1 0 0																																			
CRAFTSMEN			8 12 8 11.6 11.6 10 10.5 8 8 10.5 8 8 12.9 8 12.1 10.5 8 11.4 8 8 9 15.3 15.4 13.4 13.4 13 13 4 4 0 6.34																																			
TOTAL MAN DAYS/WORKING DAY			26 26 44 43.5 35 57 57 47.5 47.8 35 35 42 29 29 42.2 29 43.6 42 34 47.8 34 38 43.3 55.7 55.5 52.5 52.5 46.8 46.8 34 33 28 17.1 1 1																																			

NOTES:

(g) ON DUTY AND AVAILABLE AS REQUIRED; SUPERVISES MANY WORK ITEMS DURING THE TIME PERIODS GIVEN.

(b) DOES NOT INCLUDE CONTRACTOR STAFF.

(c) DECOMMISSIONING ASSUMPTIONS: 1 CALENDAR MONTH = 22 WORKING DAYS; 2 EIGHT-HR SHIFTS/WORKING DAY.

(d) THE ASSISTANT DECOMMISSIONING ENGINEER ALSO SERVES IN THIS CATEGORY AS REQUIRED

(b) THIS WORK IS PERFORMED BY THE OPERATIONS STAFF, UNDER THE OPERATING LICENSE, AS DIFFERENTIATED FROM THE DECOMMISSIONING CREWS WHO PERFORM THE DECOMMISSIONING WORK ITEMS AFTER THE PWR IS DEFUELED AND DISABLED. ASSUMPTION: 2 TEN-HR SHIFTS/DAY, 30 DAYS/MONTH FOR THE TWO MONTHS OF DEFUELING AND DISABLING OPERATIONS WORK; DECOMMISSIONING WORK CREWS WILL ASSIST THE OPERATIONS STAFF AS NEEDED DURING THIS PERIOD OF TIME.

NOTE: EACH COLUMN OF NUMBERS TO THE RIGHT OF THE CAPTION CREW PER SHIFT REPRESENTS THE NUMBER OF EACH CATEGORY OF DECOMMISSIONING WORKER REQUIRED PER SHIFT TO ACCOMPLISH THE WORK ITEMS SHOWN DIRECTLY ABOVE THE NUMBERS.

operation. Thus, extensive use is made of temporary shielding and remote handling techniques and equipment, and emphasis is placed on staying out of radiation fields when not actively participating in the work. 2) The chemical decontamination campaign is reasonably successful, reducing all radiation dose rates from piping systems and components by at least a factor of 10, and perhaps a factor of 50 to 100 in many locations. 3) Careful and current accounting of radiation exposure is maintained to rapidly identify jobs that are causing excessive exposure and permit corrective action to be taken. The accounting system also prevents undue accumulations of radiation exposure to particular personnel by flagging potential problem areas.

The results of these analyses are presented in Table G.3-1. These detailed estimates are summarized in Table 11.3-1.

G.4 ESTIMATED COSTS FOR THE DISMANTLEMENT OF THE REFERENCE PWR

The components of cost for dismantlement are compiled and presented in this section. In some instances, the bases for the cost value are developed in other sections, and those sections are identified for reference. The results of these analyses are summarized and presented in Section 10.

G.4.1 Cost of Chemical Decontamination

The complete chemical decontamination effort includes the addition of the appropriate chemicals to the water in the RCS, and CVCS and inter-tied systems; the circulation of the decontamination solution through the systems for about a week; draining, flushing, and draining of the systems; and disposal of the decontamination and flush solutions. The cost of the last item, disposal, is included in Appendix G.4.2, under Disposal of Radioactive Wastes, since essentially the same volumes of liquid are treated, whether or not a chemical decontamination is carried out.

The initial cost item is chemicals. The unit cost of EDTA/oxalic/citric acid, given in Appendix I.6, is \$1.30 /kg of the 1:1:1 mixture by weight. The required concentration of these chemicals in the solution is from 1 to 5% by weight in 378 m^3 (100,000 gallons) of water. Thus, the cost of chemicals can range from \$23,000 to \$115,000.

TABLE G.3-1. Estimated Occupational Radiation Doses for Immediate Dismantlement

EVENT DESCRIPTION	DECOMMISSIONING PERSONNEL											
	SUPERVISORS			DECOMMISSIONING CREW			CRAFTSMEN			MISCELLANEOUS TECHNICIANS		
	AVERAGE EXPOSURE RATE (R/hr)	TOTAL EXPOSURE HOURS	DOSE MAN-REM	AVERAGE EXPOSURE RATE (R/hr)	TOTAL EXPOSURE HOURS	DOSE MAN-REM	AVERAGE EXPOSURE RATE (R/hr)	TOTAL EXPOSURE HOURS	DOSE MAN-REM	AVERAGE EXPOSURE RATE (R/hr)	TOTAL EXPOSURE HOURS	DOSE MAN-REM
REACTOR BUILDING												
MOVE INTERNALS TO REFUELING CAVITY	0.01	107	1.07	0.01	358	3.58	0.01	96	0.96	0.01	239	2.39
COMPREHENSIVE PREDISMANTLING SURVEY ^(c)	0.03	54	1.62	0.03	128	3.84	0	0	0.03	118	3.54	9
SEGMENT VESSEL INTERNALS AND LOAD CONTAINERS	0.005	2112	10.56	0.005	8448	42.24	0.005	4752	23.76	0.005	3168	15.84
SEGMENT REACTOR PRESSURE VESSEL AND LOAD CONTAINERS	0.005	2152	10.76	0.005	6455	32.28	0.005	4841	24.2	0.005	2152	10.76
SEGMENT STEAM GENERATORS ^(d) AND PREPARE FOR SHIPMENT	0.01	1834	18.34	0.01	5944	59.44	0.01	4347	43.47	0.01	2276	22.76
REMOVE RCS PUMPS AND RCS PIPING	0.01	763	7.63	0.01	3137	31.37	0.01	1908	19.08	0.01	912	9.12
REMOVE PRESSURIZER, RELIEF TANK AND SAFETY INJECTION SYSTEM	0.015	470	7.05	0.015	2113	31.7	0.015	1174	17.61	0.015	563	8.44
REMOVE HX's AND ASSORTED PUMPS	0.01	311	3.11	0.01	1401	14.01	0.01	778	7.78	0.01	389	3.89
REMOVE CONTAMINATED INTERNAL STRUCTURES	0.005	1918	9.59	0.005	8633	43.17	0.005	3643	18.21	0.005	2006	10.03
REMOVE SPRAY PIPING AND VENTILATION SYSTEMS	0.009	530	4.77	0.009	2383	21.45	0.009	397	3.57	0.009	530	4.77
DECONTAMINATE REMAINING INTERNAL SURFACES	< 0.001	833	0.83	< 0.001	3747	3.75	< 0.001	347	0.35	< 0.001	833	0.83
AUXILIARY BUILDING												
REMOVE ELECTRICAL EQUIPMENT	0.0025	122	0.3042	0.0025	104	0.260	0.0024	30	0.072	0.0025	316	0.7819
DECONTAMINATION (CONTACT WORK AND INTERNAL FLUSHES)	0.0086	333	2.8648	0.0085	560	4.7647	0.0105	235	2.4578	0.0083	302	2.5023
REMOVE SELECTED BUILDING INTERNALS AND PACKAGE (PLUGS WALLS PLATFORMS)	0.0024	120	0.2912	0.0022	169	0.3708	0.0017	302	0.5232	0.0028	90	0.2549
IX RESIN AND FILTERS REMOVAL	0.0632	101	6.3866	0.1017	51	5.1867	0.1042	60	6.2533	0.0833	64	5.3334
REMOVE IX AND FILTER SYSTEM PIPING	0.0757	199	15.0584	0.0756	309	23.3739	0.0701	612	42.8822	0.0784	188	14.7457
REMOVE TANKS PUMPS AND HX's REMOVAL OF THIS EQUIPMENT IS BROKEN DOWN INTO 3 RANGES OF EXPOSURE RATES AS FOLLOWS:												
1 to 25 mR/hr	0.0087	285	2.4885	0.0093	461	4.2855	0.0092	929	8.5021	0.0087	281	2.4547
25 to 100 mR/hr	0.0473	103	4.8690	0.0487	170	8.2755	0.0479	355	17.002	0.0474	104	4.9337
> 100 mR/hr	0.2838	51	14.4746	0.3016	70	21.1109	0.2871	181	51.9702	0.2791	50	13.9544
REMOVE HVAC AND FIVE SPRINKLER SYSTEMS AND MONORAILS	0.0032	373	1.1922	0.0030	501	1.5077	0.0033	1072	3.5065	0.0031	352	1.1040
FUEL BUILDING												
REMOVE CVC SYSTEM	0.06	224	13.44	0.06	672	40.32	0.0656	216	14.1696	0.06	280	16.8
REMOVE CONCENTRATE HOLDING TANK SYSTEM	0.0338	32	1.0816	0.0338	96	3.2448	0.032	40	1.28	0.0338	40	1.352
REMOVE NEW FUEL STORAGE RACKS AND INSTALL ELECTROPOLISHING SYSTEM	0.005	16	0.08	0.005	48	0.24	0.005	96	0.48	0.005	28	0.14
REMOVE BORIC ACID SYSTEM, TANKS AND PIPING	0.0473	104	4.9192	0.0473	312	14.7576	0.0675	128	8.64	0.0507	116	5.88
REMOVE CCW SYSTEM	0.0063	96	0.6048	0.0063	288	1.8144	0.0067	164	1.0988	0.0063	116	0.7308
REMOVE SPENT FUEL RACKS FUEL TRANSFER SYSTEM AND PERIPHERAL EQUIPMENT	0.02	56	1.12	0.02	168	3.36	0.0169	116	1.9604	0.02	70	1.4
REMOVE SPENT FUEL POOL RECIRC SYSTEM, LIMERS AND CONTAMINATED CONCRETE	0.0127	88	1.1176	0.0127	264	3.3528	0.0186	112	2.0832	0.0127	110	1.397
ANCILLARIES												
COMPREHENSIVE PLANT RADIATION SURVEY ^(c)	(d)									0.005	400	2
RADIATION SURVEYS (WEEKLY) ^(e)												
REACTOR BUILDING										0.005	818	4.09
AUXILIARY BUILDING										0.005	256	1.28
FUEL BUILDING										0.005	185	0.93
CHEMICAL DECONTAMINATION ^(f) (RCS CVCS AND RELATED SYSTEMS)												
VALVING READINGS ETC				0.025	65	1.6						1.6
EVAPORATORS MAINTENANCE AND REPAIR				0.4	28	11.2	0.4	28	11.2	0.4	6	2.4
DECONTAMINATION (ELECTRO POLISHING AND DISPOSAL (PACKAGING AND SHIPPING OF CONTAMINATED EQUIPMENT AND DEBRIS))	0.005	46	0.23	0.005	742	3.71				0.005	139	0.7
PACKAGING (COMBUSTIBLE WASTES):												
REACTOR BUILDING				0.03	318	9.54	0.03	10	0.3			9.84
AUXILIARY BUILDING				0.03	318	9.54	0.03	10	0.3			9.84
FUEL BUILDING				0.03	308	9.24	0.03	10	0.3			9.54
FUEL HANDLING:												
DEFUEL REACTOR ^(g)	0.009	1232	11.09	0.009	5544	49.9				0.009	1232	11.09
SHIP FUEL OFFSITE	0.01	324	3.24	0.01	5184	51.84				0.01	972	9.72
MISCELLANEOUS ^(h)												~170
TOTAL MAN-REM												375

(a) THE NUMBER OF SIGNIFICANT FIGURES CARRIED IS FOR COMPUTATIONAL ACCURACY AND DOES NOT IMPLY PRECISION TO THAT MANY PLACES

(b) ASSUMES RCS CVCS AND RELATED INTERIED SYSTEMS HAVE BEEN CHEMICALLY DECONTAMINATED

(c) THE REACTOR BUILDING COMPREHENSIVE PREDISMANTLING RADIATION SURVEY AND THE COMPREHENSIVE PLANT RADIATION SURVEY OVERLAP THE LAST MONTH OF THE PLANT RADIATION SURVEY THIS IS NECESSARY SO THAT THE PERSONNEL CAN BE MORE EFFECTIVELY UTILIZED AND THE FIRST MONTHS RADIATION SURVEY DATA ON THE OTHER BUILDINGS AT THE STATION CAN BE USED FOR PLANNING PURPOSES WHILE WORK PROGRESSES IN THE REACTOR BUILDING THE REACTOR BUILDING IS THE FOCAL POINT OF THE MOST CONCENTRATED AND IMPORTANT DECOMMISSIONING WORK ITEMS IN THE DECOMMISSIONING OF THE REFERENCE PWR AND CONSEQUENTLY MAXIMUM EFFORT IS CONCENTRATED EARLY IN THIS BUILDING

(d) A DASH MEANS THAT CATEGORY OF DECOMMISSIONING STAFF ARE NOT INVOLVED IN THAT PARTICULAR JOB ITEM

(e) IT IS ANTICIPATED THAT WEEKLY RADIATION SURVEYS IN EACH OF THE THREE MAJOR BUILDINGS (I.e. REACTOR AUXILIARY AND FUEL) MUST BE COMPLETED AS DECOMMISSIONING WORK PROGRESSES FOR PLANNING AND PREPARATION PURPOSES.

(f) THE TOTAL MAN-REM EXPOSURE FOR THIS ITEM IS FOR THE ENTIRE JOB I.e. FILLING AND DRAINING OF THE SYSTEMS EVAPORATION AND SOLIDIFICATION OF THE WASTE SOLUTIONS AND ~25 MAN-REM ALLOCATED TO EVAPORATORS REPAIR AND MAINTENANCE.

(g) ESTIMATED FROM CAROLINA POWER AND LIGHT COMPANY MEMORANDUM DATED JANUARY 6, 1976, FILE NO. 360000 SERIAL NO. 76-013 "RADIATION EXPOSURE BREAKDOWN DURING THE H B ROBINSON UNIT 2 REFUELING OUTAGE IN 1977" C/P, RALEIGH NC 27602.

(h) DERIVED FROM THE REFERENCE GIVEN IN (g). THE ~170 MAN-REM SHOWN COULD VARY UPWARD BY A FACTOR OF TWO OR MORE DEPENDING ON 1) SHUTDOWN RADIATION/CONTAMINATION LEVELS AND 2) THE APPLICATION OF ALABA CONSIDERATIONS IN THE REFERENCE USED BOTH PLANT AND CONTRACTOR MANPOWER WERE UTILIZED AND THEREFORE IT IS VERY DIFFICULT TO ESTABLISH EITHER AVERAGE EXPOSURE RATES OR TYPES OF PERSONNEL UTILIZED DURING THE REFERENCED REFUELING OUTAGE. SUCH AN ANALYSIS IS BEYOND THE SCOPE OF THIS STUDY HOWEVER, THE BASIS USED TO SELECT WHICH MISCELLANEOUS JOB ITEMS SHOULD BE INCLUDED HERE ARE BASED ON THE ASSUMPTION THAT ONCE THE REACTOR IS DEFUELED THE REACTOR VESSEL HEAD WILL BE REPLACED IN PREPARATION FOR THE RCS CVCS, AND RELATED SYSTEMS CHEMICAL DECONTAMINATION WORK.

(i) THE FOUR SIGNIFICANT FIGURES SHOWN ARE THE SUMMATION OF MANY INDIVIDUAL ESTIMATES AND DO NOT IMPLY FOUR PLACE PRECISION IN THE ESTIMATED VALUE

Circulation of the solutions and rinses was assumed for a total of 10 to 16 days. The maximum cost for electricity, at \$15/MWh, is \$105,000.

The direct staff labor cost, developed in Figure G.2-1, is \$64,051. The total costs directly chargeable to chemical decontamination are summarized in Table G.4-1. The larger amount in Table G.4-1 is used in other cost summaries for conservatism.

TABLE G.4-1. Summary of Chemical Decontamination Costs

<u>Cost Item</u>	<u>Estimated Cost (\$)</u>
EDTA/oxalic/citric acid	23,000 to 115,000
Electrical Power	105,000
<u>Staff Labor</u>	<u>64,050</u>
TOTAL	192,050 to 284,050

G.4.2 Cost of Disposal of Radioactive Materials

The cost of disposal for radioactive materials is a major item in the total dismantlement program. These costs include the direct labor costs for removing and packaging of the material and shipping and burial. These costs and their associated burial volumes are summarized in Tables G.4-2 and 10.1-4, with the detailed bases for the estimates presented in the following sections.

G.4.2.1 Disposal of Activated Materials

All of the activated material is contained in the reactor pressure vessel, vessel internal structures, and in the surrounding steel and concrete biological shield. A detailed breakdown has been tabulated of the weights, volumes, number of pieces after sectioning, number of containers, and associated costs for packaging and disposal of the activated materials, and is presented in Table G.4-3. The total radioactivity expected to be present in the vessel and components is approximately 4.8 million curies, and will occupy nearly 1,200 m³ (42,000 ft³) of space at a disposal site. About 216 truckload shipments are estimated to be required to transport the material to the burial site. The total estimated cost for disposal of the activated material by burial in a licensed burial site is \$2,733,500.

TABLE G.4-2. Summary of Burial Volumes and Costs for Disposal of Radioactive Materials

<u>Material Category</u>	<u>Burial Volume (m³)</u>	<u>Total Disposal Cost^(a) (\$ Millions)</u>
Spent Fuel	37 (bare)	2.467 ^(b)
	57 (encapsulated)	
Activated:		
Metal	484	2.498 ^(c)
Concrete	707	0.235
Contaminated:		
Metal and Misc.	5465	1.659
Concrete	10613	3.525
Radioactive Wastes	<u>618</u>	<u>0.693</u>
TOTAL	17924	11.077

- (a) Number of figures shown is for computational accuracy and does not imply precision to the nearest one thousand dollars.
- (b) No costs are included for ultimate disposal, just transport costs.
- (c) Cost differential for deep geologic disposal of highly activated material is + \$1.8 million.

TABLE G.4-3. Burial of Activated Materials from the Containment Building

Component	Weight (a,b) (lb)	Estimated Radioactivity (Ci)	No. of Pieces	No. of (c) Containers	Container (d) Cost (\$)	Cask (e) Rental (\$)	No. of Shipments	Transportation (f) Costs (\$)	Handling (g) Costs (\$)	Burial Volume (h,i) (ft ³)	Burial Costs (g) (\$)	Liner (g) Surcharge (\$)	Curie (g) Surcharge (\$)	Total Cost/Component (j) (\$)
Pressure Vessel Cylindrical Wall	594,800	19,170	70	38	380,000	19,000	38	70,680	11,400	3,800	10,070	22,250	11,789	505,189
Vessel Head	195,000	<10	20	20	200,000	10,000	20	37,200	6,000	2,000	5,300	-----	-----	258,500
Vessel Bottom	85,700	<10	20	20	200,000	10,000	20	37,200	6,000	2,000	5,300	-----	-----	258,500
Upper Core Support Assembly	26,500	<10	12	4	40,000	2,000	4	7,440	1,200	400	1,060	800	-----	52,500
Upper Support Columns	24,500	<100	96	4	40,000	2,000	4	7,440	1,200	400	1,060	1,000	-----	52,700
Upper Core Barrel	6,000	<1,000	10	2	20,000	1,000	2	3,720	600	200	530	1,250	620	27,720
Upper Core Grid Plate	10,200	24,310	9	5	50,000	2,500	5	9,300	2,200	500	1,325	3,125	2,640	71,090
Guide Tubes	33,300	<100	122	6	60,000	3,000	6	11,160	1,200	600	1,590	1,050	-----	78,000
Lower Core Barrel	94,000	651,000	64	32	320,000	16,000	32	59,520	9,600	3,200	8,480	20,000	41,670	475,270
Thermal Shields	23,000	146,100	12	6	60,000	3,000	6	11,160	1,800	600	1,590	3,750	9,015	90,315
Core Shroud	27,100	3,431,000	96	4	40,000	2,000	4	7,440	1,200	400	1,060	2,500	192,165	246,365
Lower Grid Plate	8,700	553,400	9	5	50,000	2,500	5	9,300	2,200	500	1,325	3,125	30,635	99,085
Lower Support Columns	7,400	10,000	96	1	10,000	500	1	1,860	300	100	265	625	785	14,335
Lower Core Forging	80,400	2,500	25	11	110,000	5,500	11	20,460	3,300	1,100	2,915	6,875	3,300	152,350
Misc. Internals	80,000	2,000	80	8	80,000	4,000	8	14,880	2,400	800	2,120	5,000	3,080	111,480
810 Shield Concrete	~1,950,000	<2,000	---	195	78,000	-----	49	91,140	-----	24,960	66,144	-----	-----	235,284
Reactor Cavity Liner	~32,000	<10	---	4	1,600	-----	1	1,860	-----	512	1,357	-----	-----	4,817
TOTALS (j)	3,278,600 lb 1,487,148 kg	<4,843,320	741	365	1,739,600	83,000	216	401,760	50,600	42,072 ft ³ 1191.35 m ³	111,491	51,350	295,699	2,733,500

(a) Weights approximate, calculated from dimensions or obtained from TROJAN FSAR.

(b) 1 lb = 0.45359 kg

(c) Indicative of volume only. Actual pieces are distributed throughout all containers to satisfy the 50,000 Ci/container limit for burial.

(d) Based on Table I.2-1

(e) Based on Table I.3-1

(f) Based on Table I.4-4

(g) Based on Table I.5-1

(h) Volume includes the disposable container

(i) 1 ft³ = 0.028317 m³

(j) The number of figures shown is for computational accuracy and does not imply precision to that many significant figures.

TABLE G.4-3. Burial of Activated Materials from the Containment Building

Cost Differential for Deep Geological Disposal. The information presented in Table G.4-3 is based on the assumption that all of the activated vessel components will be placed in shallow land burial. Because of the large initial level of radioactivity and the very long lives of the ^{59}Ni and ^{94}Nb radionuclides, and the potential biological hazard of ^{14}C , it has been suggested that the activated components should be placed in a deep geologic disposal facility, in the same way that high level wastes, cladding hulls, etc., from fuel reprocessing are proposed to be isolated from the environment. Based on current designs for the waste containers for such a facility, 0.76 m dia x 3.05 m in length (30 in. dia. x 10 ft in length), and the disposal costs (\$7,100/m³) postulated for such materials,⁽¹⁾ an analysis was made to estimate the additional cost of placing the highly activated materials in deep geologic storage. Only the highly activated components such as the upper and lower core grid pieces, the lower core barrel, the core shroud and the thermal neutron shield pads were considered for placement in deep geologic storage. Mildly activated materials like the vessel heads, the pressure vessel wall, the upper core assembly, support columns and guide tubes, and the lower support columns would be placed in shallow land burial, as before.

The total weight of the highly activated components is calculated to be about 73,935 kg (~163,000 lb), with a density of 8,038 kg/m³ (501 lb/ft³) and a theoretical volume of about 11 m³ (392 ft³). Assuming a packing density of 15% of the theoretical density in a shipping canister that has an internal volume of about 1.2 m³ (42 ft³), a total of 63 canisters with a burial volume of 87.5 m³ is required to package these components. The cost of each canister is assumed to be \$5,000, for a total of \$315,000.

Shipment is assumed to take place in an IF-300 cask, on a railcar, over a round-trip distance of 4,830 km (3,000 mi.), with an elapsed time of 18 days. per round-trip. The cask rental is assumed to be \$3,500/day (Table I.3-1) and the transportation charges are assumed to be \$25,100 per cask shipment (section I.4.2). Each cask is assumed to carry 3 canisters per shipment, for a total of 21 shipments. The total estimated cost of shipping the 62 canisters is \$1,850,100.

⁽¹⁾ K. J. Schneider and C. E. Jenkins, Technology Safety and Costs of Decommissioning a Reference Fuel Reprocessing Plant NUREG-0278, U.S. Nuclear Regulatory Commission, pp. 7-129, September 1977.

The cost of placing the canisters into deep geologic disposal, at $\$7,100/\text{m}^3$, is $\$610,600$. The total estimated disposal costs are summarized below:

<u>Canisters (\$)</u>	<u>Cask Rental (\$)</u>	<u>Transportation (\$)</u>	<u>Disposal (\$)</u>	<u>Total (\$)</u>
315,000	1,323,000	527,100	621,250	2,786,350

These costs are defrayed somewhat by not having to package and dispose of this material in a shallow land burial facility. From Table G.4-3, these costs are estimated to be $\$982,100$, of which $\$530,000$ is due to the cost of shielded containers. Thus, the net increase in cost for placing the highly activated reactor components in deep geologic storage rather than shallow land burial is about $\$1.8$ million, or about a 5% increase in the total decommissioning cost.

G.4.2.2 Disposal of Contaminated Material

Materials considered to be contaminated include nearly all piping and equipment present in the Reactor Containment, Auxiliary, Control and Fuel buildings. In addition, significant areas of concrete surfaces in these buildings are also assumed to be contaminated in depth and will require surface removal to a depth of several inches. A breakdown of the number of containers, volumes of material and disposal costs is given in Table G.4-4 and G.4-5 for Containment Internals and Other Building Internals. The volume of contaminated materials requires about $16,100 \text{ m}^3$ ($567,800 \text{ ft}^3$) at a disposal site and requires an estimated 1,147 truckload shipments to transport the material to the burial site. The radioactivity present in this material is not expected to be great, probably less than 1,000 curies, total.

G.4.2.3 Disposal of Radioactive Wastes

The materials considered in this study as radioactive wastes are the following: used ion exchange resins, cartridge filters, evaporator bottoms liquids, and miscellaneous combustible wastes such as plastic sheeting, rags, disposable protective clothing, etc. Also considered is the cost of shipment of the spent fuel. The IX resins, filters, and evaporator bottoms liquids are mixed with or encapsulated in solid materials such as cement or ureaformaldehyde to assure retention of radioactive materials within the shipping containers

TABLE G.4-4. Disposal of Contaminated Materials from the Containment Building

Component	Weight ^(a,b) (lb)	No. of Cut Pieces	No. of Disposable Containers	Container ^(e) Cost (\$)	No. of Shipments	Transportation ^(d) Cost (\$)	Handling ^(e) Cost (\$)	Burial ^(f,g)		Total Cost ⁽ⁱ⁾ Per Component (\$)
								Volume ^(f,g) (ft ³)	Cost ^(e) (\$)	
Reactor Coolant Pumps	752,000	12	g ^(h)	49,600	12	22,320	6,920	4,200	11,130	89,970
Pressurizer	195,000	8	8	48,000	8	14,880	1,550	3,600	9,540	73,970
Regenerative Heat Exchanger	6,600	1	(h)	800	1	1,120	200	400	1,060	4,380
Excess Letdown HX	1,600	1	(h)	800						
Containment Sump Pumps	2,800	2	1	400						
Reactor Cavity Drain Pump	800	1								
Pressurizer Relief Tank	27,200	8	2	12,000	2	2,240	200	1,200	3,180	17,620
Safety Injection Accumulation Tanks	306,000	32	8	48,000	8	14,880	2,660	4,000	10,600	76,140
Steam Generators	2,752,000	32	(h)	3,600	32	59,520	25,920	21,362	56,609	145,649
Reactor Coolant Piping	222,000	274	7	42,000	7	13,020	1,870	3,300	8,745	65,635
Refueling Cavity Liner	38,000 (steel) 608,000 (concrete)	--- ---	64	25,600	16	29,760	-----	8,192	21,709	77,069
Pressurizer Enclosure Concrete	540,000	---	54	21,600	14	26,040	-----	6,912	18,317	65,957
Steam Gen. Enclosure Concrete	1,060,000	---	106	42,400	26	48,360	-----	13,568	35,955	126,715
Missile Shield	38,000	4	4	1,600	1	1,860	-----	512	1,357	4,817
Base Slab Liner and Concrete	1,150,000 (concrete) 120,000 (steel)	--- ---	128	51,200	32	59,520	-----	16,384	43,418	154,138
Containment Spray Piping	200,000	---	20	8,000	5	9,300	-----	2,560	6,784	24,084
Miscellaneous Equipment	30,000	---	10	4,000	2	2,240	-----	1,280	3,392	9,632
Ladders, Gratings, etc.	100,000	---	20	8,000	4	4,480	-----	2,560	6,784	19,264
Internal Surface Decontamination	40,500	---	5	2,000	1	1,860	-----	640	1,696	5,556
TOTALS (i)	8,190,500 lb 3,715,129 kg		445	369,600	171	311,400	39,320	90,670 ft ³ 2,568 m ³	240,276	960,596

(a) Weights are approximate, calculated from dimensions or obtained from TROJAN FSAR.

(b) 1 lb = 0.45359 kg

(c) Based on Table I.2-1, closure plates @ \$200 each

(d) Based on Table I.4-4

(e) Based on Table I.5-1

(f) Volume includes disposable container

(g) 1 ft³ = 0.028317 m³

(h) Some portions are sealed with welded closure plates, no outer container.

(i) The number of figures shown is for computational accuracy and does not imply precision to that many significant figures.

TABLE G.4-5. Disposal of Contaminated Materials from Other Buildings

Component	Weight ^(a,b) (lb)	No of Disposable Containers	Container ^(c) Cost (\$)	No of Shipments	Transportation ^(d) Cost (\$)	Burial		Total Cost ⁽¹⁾ Per Component (\$)
						Volume ^(f,g) (ft ³)	Cost ^(e) (\$)	
Stainless Steel Systems								
Miscellaneous Valves	140,000	24	9,600	3	5,580	3,072	8,141	23,321
Residual Heat Removal	26,300	4	1,600	1	1,860	512	1,357	4,817
Chem. and Volume Control	39,160	13	5,200	2	2,280	1,664	4,410	11,890
Emergency Core Cooling	222,000	101	40,400	6	11,160	12,928	34,259	85,819
Makeup Water	12,400	4	1,600	---	-----	512	1,357	2,957
Liquid Waste	20,555	7	2,800	1	1,860	896	2,374	7,034
Gaseous Waste	4,435	2	800	---	-----	256	678	1,478
Spent Fuel Cooling	8,340	3	1,200	---	-----	384	1,018	2,218
Structural Material	338,000	112	44,800	14	26,040	14,336	37,990	108,830
Tanks	336,000	112	44,800	14	26,040	14,336	37,990	108,830
Heat Exchangers	204,000	(h)	4,000	5	9,300	2,350	6,228	19,528
Evaporators	66,000	22	8,800	3	5,580	2,816	7,462	21,842
Waste Solidification	104,000	32	12,800	4	7,440	4,096	10,854	31,094
Pumps	194,000	64	25,600	8	14,880	8,192	21,709	62,189
HVAC Ductwork	180,000	80	32,000	10	11,200	10,240	27,136	70,336
Filters	33,000	11	4,400	2	2,280	1,408	3,731	10,411
Carbon Steel Systems								
Auxiliary Feedwater	31,660	5	2,000	1	1,860	640	1,696	5,556
Service Water	132,000	44	17,600	6	11,160	5,632	14,925	43,685
Component Cooling	372,000	124	49,600	15	27,900	15,872	42,061	119,561
Service Cooling	55,400	18	7,200	2	3,720	2,304	6,106	17,026
Condensate Storage	85,500	28	11,200	3	5,580	3,584	9,498	26,278
Turbine Generator	888,000	154	61,600	22	40,920	19,712	52,237	154,757
Auxiliary Steam	53,590	18	7,200	2	3,720	2,304	6,106	17,026
Compressed Air	42,920	14	5,600	2	3,720	1,792	4,749	14,069
Glycol Heating	23,000	8	3,200	1	1,860	1,024	2,714	7,774
Steam Gen Blowdown	6,250	2	800	---	-----	256	678	1,478
Extraction Steam	158,000	52	20,800	7	13,020	6,656	17,638	51,458
Miscellaneous Piping	40,800	13	5,200	2	3,720	1,664	4,410	13,330
Contaminated Concrete	26,374,000	2,638	1,055,200	660	1,227,600	337,664	894,810	3,177,610
<hr/>								
TOTALS (1)	30,191,310 lb	3,709	1,487,600	796	1,470,280	477,111 ft ³	1,264,322	4,222,202
	13,694,476 kg					13,510 m ³		

(a) Weights are approximate, calculated from dimensions or obtained from TROJAN FSAR

(b) 1 lb = 0.45359 kg

(c) Based on Table I 2-1, closure plates @ \$200 each

(d) Based on Table I 4-4

(e) Based on Table I 5-1

(f) Volume includes disposable container

(g) 1 ft³ = 0.028317 m³

(h) Some portions are sealed with welded closure plates, no outer container

(i) The number of figures shown is for computational accuracy and does not imply precision to that many significant figures

after disposal. The combustible wastes are reduced in volume as much as possible by compaction prior to packaging.

A breakdown of the number of containers, volumes of materials and disposal costs is given in Table G.4-6 for disposal of wastes. About 180 truckloads will be shipped that will occupy about 618 m^3 ($21,840 \text{ ft}^3$) at the burial site. The development of the costs for these materials is given in the following sections.

Combustible Wastes

Sizeable quantities of contaminated combustible waste are produced during the decontamination and dismantlement program. Uncompacted volumes of 2,275, 2,200, and 2,275 drums (55 gallon, 0.21 m^3) of waste are postulated to be collected from the Containment, Fuel, and Auxiliary buildings, respectively. These volumes are assumed to be compacted by a factor of 5:1 before disposal. Thus, a total of 1,350 drums are required. Of these, 420 are assumed to have radiation readings sufficiently high to require shipment in a shielded cask. The remaining 930 drums are shipped on the same trucks, unshielded. All shipments are assumed to be over legal weight. The costs for disposal of contaminated combustible wastes are presented in Table G.4-7 and are summarized in Table G.4-6.

TABLE G.4-7. Costs for Disposal of Contaminated Combustible Wastes

Source	No. Drums (a)	Container Cost (b) \$	Cask Rental (c) \$	Transportation Cost (d) \$	Burial Cost (e) \$
Containment Bldg.	455	9,100	10,000	37,200	9,043
Fuel Bldg.	440	8,800	10,000	37,200	8,745
Auxiliary Bldg.	455	9,100	10,000	37,200	9,043
Total	1,350	27,000	30,000	111,600	26,831

(a) Waste volumes have been compacted by 5:1 for final disposal; burial volume is 0.21 m^3 (7.5 ft^3) per drum.

(b) Container costs from Table I.2-1.

(c) Assumes 5 days/shipment, rental from Table I.3-1, 60 shipments.

(d) Each shipment consists of 1 cask w/7 drums, plus 15-20 unshielded drums, overweight.

(e) Surface radiation dose rate assumed to be $<0.2 \text{ R/hr}$ for all drums.

TABLE G.4-6. Disposal of Radioactive Wastes from Decommissioning

Material	No of Disposable Containers	Container ^(a,b) Cost	Cask ^(c) Rental	No of Shipments	Transportation ^(d) Cost	Handling ^(e) Cost	Burial		Liner ^(e) Surcharge	Curie ^(e) Surcharge	Total ^(h) Cost
							Volume ^(f,g)	Cost ^(e)			
		(\$)	(\$)		(\$)	(\$)	(ft ³)	(\$)	(\$)	(\$)	(\$)
Combustible Wastes	1,350	27,000	30,000	60	111,600	12,000	10,125	26,831	-----	-----	207,431
Spent Resins	20	18,500	10,000	20	37,200	4,000	2,000	5,300	9,500	7,800	92,300
Spent Filter Cartridges	42	4,200	3,000	6	11,160	1,200	315	835	5,250	1,960	27,605
Evaporator Bottoms Liquids	94	86,950	47,000	94	174,840	18,800	9,400	24,910	12,700	-----	365,200
TOTALS (h)	1,506	136,650	90,000	180	334,800	36,000	21,840 ft ³ 618 m ³	57,876	27,450	9,760	692,536

(a) Container costs from Table I 2-1

(b) Includes solidification cost from Section I 2 2, where applicable

(c) Based on Table I 3-1

(d) Based on Table I 4-4

(e) Based on Table I 5-1

(f) Volume includes disposable container

(g) 1 ft³ = 0.028317 m³

(h) The number of figures shown is for computational accuracy and does not imply precision to that many significant figures

Spent Resins

The total volume of spent ion exchange resins for disposal is assumed to be 30 m^3 ($1,060 \text{ ft}^3$), or about 3 times the volume generated annually during reactor operation. The resins are estimated to contain up to 42,000 Ci of radioactivity. The resins are assumed to be mixed with a solidifying agent and packaged in steel cask liners having a burial volume of 2.84 m^3 (100 ft^3). Twenty containers are required. Each container is shipped in a shielded cask, one cask per truck shipment. Each shipment is assumed to be over legal weight. The average curie content per container is assumed to be 2,000 Ci. The costs for burial of contaminated spent ion exchange resins are presented in Table G.4-6.

Used Particulate Filters

Forty-two cartridge filters for fluid streams are assumed to be packaged and shipped to a burial site. Each filter is placed within a concrete-lined drum (burial volume 0.21 m^3 , 7.5 ft^3), for shipment. The drums are shipped in a shielded cask, 7 drums per cask, one cask per shipment. A total of 5,000 Ci of radioactivity is assumed to be contained in the filters, for an average content of about 120 Ci per filter. Each truck shipment is assumed to be over legal weight, and requires five days, round trip. The costs for disposal of these filters is summarized in Table G.4-6.

Evaporator Bottoms Liquids

The initial plant inventory of water borated to about 0.2% by weight is assumed to be about $3,785 \text{ m}^3$ (1,000,000 gallons). One coolant system volume ($\sim 378 \text{ m}^3$) also contains decontamination chemicals at the 1% by weight concentration. An additional $3,785 \text{ m}^3$ (1,000,000 gallons) of water from rinsing, flushing and washing operations is also assumed to be evaporated. The decontamination solution is assumed to be concentrated to 57 m^3 , the remaining borated water concentrated to 38 m^3 , and the rinse water to 38 m^3 , for a total of 133 m^3 of evaporator bottoms liquids to be solidified. This liquid is assumed to be solidified in 94 containers whose burial volumes are 2.84 m^3 (100 ft^3) each. Twenty-two containers are assumed to have surface radiations dose rates between 1 and 50 R/hr, 45 containers are assumed to have surface radiation rates between 0.2 and 1 R/hr, and the remaining 27 containers to have surface radiation dose rates less than 0.2 R/hr.

The estimated costs for disposal of contaminated evaporator bottoms liquids are presented in Table G.4-8 and are summarized in Table G.4-6.

TABLE G.4-8. Costs for Disposal of Evaporator Bottoms Liquids

Assumed Surface Radiation Dose Rate (R/hr)	No. (f) of Containers	Container ^(a,b) and Solidification Costs	Cask ^(c) Rental	Transportation Cost	Handling ^(e) Cost	Liner ^(e) Surcharge	Burial ^(e) Cost
50	22	20,350	11,000	40,920	4,400	10,450	5,830
1	45	41,625	22,500	83,700	9,000	2,250	11,925
0.2	27	24,975	13,500	50,220	5,400	--	7,155
TOTAL	94	86,950	47,000	174,840	18,800	12,700	24,910

(a) Container costs from Table I.2-1.

(b) Solidification costs from Section I.2.2.

(c) Assumes 5 days/shipment, rental from Table I.3-1.

(d) Assumes overweight shipment, from Table I.4-4.

(e) Costs from Table I.5-1.

(f) One container in cask per shipment.

Spent Reactor Fuel

Since the ultimate disposition of spent reactor fuel is still an unknown, for this study it was assumed that the 193 fuel assemblies are shipped in IF-300 casks, in shipments of 7 assemblies per cask, to a repository located 2424 km (1,500 mi.) from the reactor site, using rail transportation. Thus, 28 shipments, requiring 18 days round trip, are needed. Based on the cask rental given in Table I.3-1 of \$3,500 per day, the total cask rental is \$1,764,000. The rail freight cost, from Appendix I.4.2, is \$702,800. Thus, the estimated cost for removing the fuel from the reactor and transporting the fuel to a repository is \$2,466,800. This cost does not include any charges associated with handling at the repository. The unencapsulated volume of the spent fuel assemblies is approximately 37 m³.

G.4.3 Cost of The Quality Assurance Program

The principal costs of the QA program are the staff labor to execute the program. From Table 10.1-2, the program outlined in Appendix F.4 requires 5.3 man years of a QA supervisor, 5.1 man years of a QA Engineer, and 12.5 man years of QA Technicians. Not explicitly included is the secretarial assistance that would be needed. The total cost of the QA staff labor is estimated to be \$723,000.

G.4.4 Cost of The Environmental Surveillance Program

The environmental surveillance activities, as described in Appendix F.4 would continue at the operational level of effort until the final shipment of radioactive material from the site. At that time, four years after reactor shutdown, the program would be terminated. The first year's decommissioning monitoring costs are given in Table G.4-9. The costs shown assume the work is accomplished by the same subcontractor who provided the environmental monitoring services during the operational life of the facility. Assuming a four year program, the total cost is \$154,000.

TABLE G.4-9. Costs of the Environmental Monitoring Program for the First Year of Decommissioning Activities

Labor	\$33,000
Equipment and supplies	1,000
Utilities and servies	<u>1,000</u>
Subtotal	35,000
Contractors' 10% overhead and profit	<u>3,500</u>
TOTAL	38,500

G.4.5 Cost of Demolition of Structures

To assure a realistic estimate of costs for demolition of the decontaminated and non-contaminated structures at the reference PWR site, a wrecking contractor was engaged to develop these costs, just as he would for bidding on the job. The results of that analysis⁽²⁾ are presented in this section. The costs presented in this section reflect wage scales and operating costs prevailing in the Pacific Northwest in mid-1976. These costs have been increased by 11.3% for final presentation in Table 10.1-13, Section 10.1, to account for escalation between mid-1974 and January, 1978.

The basic conditions for the analysis were the following:

- All structures on the plant site are demolished and removed.
- Building structures are demolished down to the lowest sub-floor level, holes broken or drilled in sub-basement floors for drainage, then filled to within 1 meter of surface level with uncontaminated rubble and backfilled, the last 1 meter with typical local topsoil.
- Demolition contractor would have building structures salvage rights with a fair profit to the contractor for the salvage but with the total estimated bid cost to reflect this potential salvage profit where applicable.
- The excess uncontaminated rubble can be disposed of at suitable landfill within 16 kilometers of the PWR site.
- The entire demolition job goes according to the time schedule shown in Figure 9.1-2.

The information in this section has not been converted to metric units to avoid confusion when comparing these data with the source document. The use of the units (yd³ and \$/yd³) is intermediate to the final result (\$) and doesn't impact the results of the study.

The initial step in any demolition job is to determine a possible procedure to follow to take down safely. The initial procedure and its possible alternatives are then compared and reviewed to arrive at a preferred procedure, bearing in mind that discovery of difficulties or alternative possibilities later could cause the preferred procedure to be discarded in favor of a new one. This preferred procedure provides the indispensable function of outlining the series of tasks that must be sequenced and costed to arrive at

(2) J. M. McFarland, Report on Representative Cost Estimates for Demolition of Structures at a Pressurized Water Reactor Site, PNL-2450, prepared for Battelle, Pacific Northwest Laboratory, by McFarland Wrecking Corporation, Seattle, WA 98108, September 30, 1976.

reasonable time and cost estimates. The cost estimate in 1976 dollars for demolition of uncontaminated structures, is summarized in Table G.4-10, and includes all job costs, supplies, overhead and profit. It does not include extraordinary insurance requirements, bonding, or state sales tax. Included in the analyses are pertinent comments regarding estimating, hauling costs, salvage, equipment requirements and current wage rates as well as tangible judgment factors.

TABLE G.4-10. Summary of Estimated Demolition Costs for the Reference PWR(a) (1976 Dollars)

Cooling Tower	\$2,217,500
Reactor Containment Vessel	1,192,000
Auxiliary Building.	598,200
Fuel Building	339,780
Control Building	384,000
Turbine Building	781,600
Turbine Auxiliary Building	96,500
Condensate Demineralizer Building.	25,510
Administration Building	13,680
Shop and Warehouse.	2,000
Chlorine Building	5,760
Intake Structure (no quantity estimate)	
Barge Loading Facility (no quantity estimate)	
Intake Piping (no quantity estimate)	
Contingency Allowance for No Quantity Items	50,000
Area Landfilling (excludes roadways).	<u>50,000</u>
Total Cost Estimate to Remove PWR Plant.	\$5,756,530

(a) These estimates are based on wage scales and other costs existing in September, 1976. An increase of 11.3% was applied to account for escalation to January, 1978, when these data were presented in Table 10.1-3.

G.4.5.1 Cooling Tower

The postulated procedure for the removal of the Cooling Tower is to first clear out the inside cooling fins. Next, drill the top ring walkway; drill all columns and the base of the hyperbolic section up about 10 ft; drill fracture lines 100 ft up the hyperbolic face on the side that will touch ground first and back 100 degrees on each side, more if further study so indicates. Weaken the top ring with light charges before collapse. Blast the hyperbolic fracture lines about the 60 degree section that will touch ground first and the section of columns and hyperbolic base below those first fracture lines. Drop the structure by simultaneously taking out the columns, column bases and hyperbolic base to the right and left of the initial column removal together with additional hyperbolic fracture lines. As the structure starts over, release it by blasting the back columns and hyperbolic base ring. Clean up with crane and breaking ball and haul away excess rubble. The project assumes leaving the foundation ring and the center slab. Break the reservoir ring above grade with a breaking ball. Assumed above the foundation is 40,500 yd³ of concrete. Following toppling of the structure, the crane and ball cost for breaking the structure for loading is estimated at \$25/yd³. Loading and hauling is estimated at \$10/yd³.

Preparatory breaking by explosives of the heavier portions of the tower plus possible breaking up during collapse could substantially reduce the cost of preparing the material for loading. The figure used presupposed that substantial breaking up of the structure, after it reached the ground, would still be required. For placement of the explosives drill holes, the driller will need details on the placement and spacing of the steel reinforcing.

The removal of the cooling fins assumes a loose in-place column of ~20,000 yd³ which would reduce to ~10,000 yd³ stacked cubic measure. Cost breakdown for the cooling tower removal is as follows:

Crushing and preparation	20,000 yd ³ at \$10/yd ³	\$200,000
Loading and hauling	10,000 yd ³ at \$10/yd ³	<u>100,000</u>
Total cost of cooling fins removal		\$300,000

Blasting top ring	870 yd ³ est. at \$46/yd ³	\$ 40,000
Blasting hyperbolic cut out and lines	6,000 yd ³ est. at \$46/yd ³	276,000
Blasting 88 columns and hyperbolic base ring	4,125 yd ³ at \$32/yd ³	132,000
Secondary blasting (contingency)		<u>52,000</u>
Total Explosives Work		\$500,000

Breakup of reservoir ring and opening slab for percolation is included with secondary blasting and balling and separation of rubble.

Balling and separating the rubble	40,500 yd ³ at \$25/yd ³	\$1,012,500
Loading and hauling	40,500 yd ³ at \$10/yd ³	<u>405,000</u>
Total Cleanup Cost		\$1,417,500
Total Cost of Removing Cooling Tower		\$2,217,500

G.4.5.2 Reactor Containment Vessel

Removal of the Reactor Containment Vessel (RCV) was considered from three approaches:

- 1) drilling and blasting plus crane and ball
- 2) crane and ball, and
- 3) an innovative procedure utilizing internal heating.

In each case, the post-tensioning tendons are removed as an initial step. The explosives procedure, utilizing the vertical post-tensioning tubes, combined with crane and ball, was the preferred conventional procedure. This cost estimate is used in the cost summary, Table G.4-9.

Explosives Removal of RCV

This approach makes use of the post-tensioning ducts for the placement of the explosives. As the tendons are withdrawn, they can be used to pull lengths of 2-in. PVC pipe into the ducts or the explosives could be lowered after opening the ducts in the dome. This 2-in. PVC would be used for placement and containment of the charges. To provide a firm transmittal of the shock from the explosives, the spaces between the pipe and the duct should be grouted. An alternative could be to grout a 2 1/2-in. PVC into the duct,

then drawing a 2-in. PVC pipe pre-loaded with explosives through the larger one when preparing to shoot. The cost of loading these tubes grouting and blasting is estimated at \$4/linear ft of tubing. The economy of using the existing ductwork is apparent together with the difficulty of drilling new openings into the vessel. Initial consideration has been limited to the vertical tendons; but could be also applied to the horizontal ducts at the same rate if more detailed study should indicate that a better breakage pattern would result. Access to the vertical ducts at the dome was not obvious; however, access could be obtained by a preliminary blast of the horizontal duct work at the upper level. After blasting, the RCV would be sufficiently weakened that the crane and ball, together with men and torches in crane-held baskets, could take down the structure. The base would be drilled and blasted, cleaned and burned until the base was penetrated.

The balling, separating, loading and hauling estimates allow for a strong tangled rebar mass with residual concrete bound to and in it. This will slow the repetitive cycles of breaking the concrete, cutting the rebar and cleaning away the loose material. Cost breakdown for the explosives removal of the RCV is as follows:

RCV shell blasting	42,000 lin. ft at \$3/ft	\$ 126,000
RCV shell balling and separating	10,000 yd ³ at \$40/yd ³	400,000
RCV shell loading and hauling	10,000 yd ³ at \$15/yd ³	150,000
RCV internals blasting	3,200 yd ³ at \$25/yd ³	80,000
RCV internals balling and separating	3,200 yd ³ at \$30/yd ³	96,000
RCV internals loading and hauling	3,200 yd ³ at \$15/yd ³	48,000
RCV base blasting	4,000 yd ³ at \$18/yd ³	72,000
RCV base balling and cleaning	4,000 yd ³ at \$40/yd ³	160,000
RCV base loading and hauling	4,000 yd ³ at \$15/yd ³	<u>60,000</u>
TOTAL		\$1,192,000

Crane and Ball Procedure for Removal of RCV

The crane and ball alternative for the RCV demolition is at the limits of the state-of-the-art. Because of the height of the target at the dome level, and because of the crane's swing and "suck-line" (fairlead control line from the base of the boom) limitations, the crane would operate at 5 percent or less efficiency. The shape of the target makes a square hit with the drop action very difficult and limited. The thickness of the concrete, its extra strength and heavy reinforcement might exceed the breaking power of a 10,000 ball. It would take a 200 ton crane (the present limit of the state-of-the-art) to handle that height and reach. The solution would be to leave the dome and concentrate on hitting the RCV closer to its base where the crane has its best geometry for "suck-line" balling; and let the dome fall down to where it can be worked on. Because of the thickness and strength of the structure, a 20,000 lb battering ram would be needed to pair up with the 200 ton crane rigged up with the "suck-line" arrangement and 280 ft of boom. It would cost in the area of \$100/yd³ of concrete to take the structure down, plus the cost of loading and hauling at \$15/yd³.

The base would probably withstand the drop of a 20,000 lb ball; but it should be tried for a day or so. It is expected that drilling and blasting would have to augment the ball work here; and that cutting of rebar to permit repeated drilling would be necessary. The driller should seek the record of the concrete pour openings and try to follow them. Cost of drilling and blasting the base is estimated to be \$18/yd³. Balling after blasting of the base is estimated at \$40/yd with loading and hauling at \$15/yd.

Cost breakdown for the RCV crane and ball procedure is as follows:

Balling and preparation (external and internals above base)	13,200 yd ³ at \$100/yd ³	\$1,320,000
Loading and hauling	13,200 yd ³ at \$15/yd ³	198,000
Base blasting	4,000 yd ³ at \$18/yd ³	72,000
Base balling and separating	4,000 yd ³ at \$40/yd ³	160,000
Base loading and hauling	4,000 yd ³ at \$15/yd ³	<u>60,000</u>
TOTAL		\$1,810,000

Heating Procedure for Removal of the RCV

The following innovative method for removal of the RCV is unique due to the nature of the RCV itself. Analysis of this procedure indicated significant savings were possible; however, it is untried and its feasibility would have to be verified by further investigation. Savings of approximately \$500,000 compared to the explosives/crane and ball method could conceivably be realized. Since it is untried however, it was not used as the basis of any cost estimates but appears in the context of a conceivable alternative method only.

Concrete structures involved in fires are weakened to the extent that as the intense heat penetrates the concrete at temperatures of 1200 to 1500°F, concrete loses its adhesion due to differential expansion of the aggregate.

It can be seen that a substantial wood/coal fire augmented with adequate air intake and exhaust (and possibly augmented with injected oil or gas to keep adding heat over an extended period of time) in a confined area such as the RCV--particularly with the tendon gallery and reactor pit for getting the heat into the ground slab--could convert the concrete to a loose sandstone that could be shaken off the reinforcing skeleton with very little effort and very little mess.

Using the heat dissipation rate of concrete, which depends upon the aggregate composition, the heat input requirements and heating time could be estimated by calculation. The conductivity of the rebar and the post-tensioning ducts should speed the heat distribution. The rebar in the slab with heating from below in the post-tensioning gallery should distribute the heat throughout and make possible the slab's decomposition. The expansion differential of the steel reinforcement compared to the cement aggregate should be such that the steel is left entirely clean and salvageable. Salvage of steel reinforcement should be close to 100% depending upon the degree of success of the method.

A property of steel, that could have application to the inner steel lining is that under heat it is much weaker than wood. Whereas wood maintains its strength in the uncharred cross-section, steel loses its entire

strength when it reaches a critical temperature. The implication is that the steel lining could be induced by its own nature and weight or by suspended weights or other added forces to buckle and separate from the concrete. After cooling, it could be more easily cut up and removed. Possibly this effect could be achieved by the weight of the loose aggregate if the dome concrete is balled loose onto the steel inner liner while it is still critically hot--it could collapse a good part of the way to the bottom.

The cost for taking apart the RCV should be on the order of \$10 to \$20 per cubic yard of concrete. Costs of loading and hauling should be \$10 to \$15/yd³.

Cost breakdown for the entire structure is as follows:

Balling and preparation	17,234 yd ³ at \$20/yd ³	\$344,680
Loading and hauling	17,234 yd ³ at \$15/yd ³	<u>258,510</u>
TOTAL		\$603,190

G.4.5.3 Other Buildings and Structures

The Auxiliary Building, Fuel Building, Control Building, Turbine Building, Auxiliary Bay and Turbine Building are much heavier than conventional buildings, with some heavy reinforced concrete floors and with heavy central core concrete structures. The Warehouse and Shop and Administration Buildings are conventional and relatively inexpensive to remove.

The mechanical-electrical salvage can be best cut loose by oxyacetylene torch cutting and lifted out as the roofs, sidewalls and floors are removed. Nominal thickness concrete floors can be broken in place and the steel ribbing cut out or sections can be cut and lifted clear of the structure. Heavy floors and walls can be balled most economically if the crane is effective. If not, drilling and blasting is necessary to speed up the operation.

The heavy floors and walls would have an estimated removal cost of \$40 to \$60/yd³. This would be accomplished by crane and ball combined with explosives for footings and heavy concrete too difficult for the crane. Balling of these floors and walls could range from \$10 to \$40/yd³. Drilling and blasting such walls and floors is estimated to cost \$16/yd³ with follow-up balling. For loading and hauling add \$10/yd³.

Auxiliary Building

The procedure is to remove wood framing, metal siding, accessible piping and machinery. Remove the roof, sheathing and steel framework above the heavy concrete. Ball the floors and walls if it appears that the breaking ball will be effective. If pre-weakening with explosives is required, drill and blast by attempting to drill vertically with track drills wherever possible for economy. After weakening with explosives, ball and separate the concrete from the rebar and repeat if necessary preparatory to loading out. Cost of drilling and blasting is estimated at \$16/yd³.

Secondary balling after blasting, exposure for separation and follow-up blasting if necessary is estimated at \$34/yd³. This relatively high rate reflects the heavy, well reinforced construction. Where drilling and blasting are not needed, this rate is sufficient to cover heavy primary balling. Loading and hauling is estimated at \$10/yd³.

Drilling and blasting	9,970 yd ³ at \$16/yd ³	\$159,520
Balling and separating	9,970 yd ³ at \$34/yd ³	338,980
Loading and hauling	9,970 yd ³ at \$10/yd ³	<u>99,700</u>
TOTAL		\$598,200

Fuel Building

Use the same procedure as for the Auxiliary Building and the same cost estimate rates:

Drilling and blasting	5,663 yd ³ at \$16/yd ³	\$ 90,608
Balling and separating	5,663 yd ³ at \$34/yd ³	192,542
Loading and hauling	5,663 yd ³ at \$10/yd ³	<u>56,630</u>
TOTAL		\$339,780

Control Building

This structure has four stories above grade, walls and roof of 2-ft 6-in. thick concrete. Use the same procedure as for the Auxiliary Building and use the same cost estimate.

Drilling blasting	6,400 yd ³ at \$16/yd ³	\$102,400
Balling and separating	6,400 yd ³ at \$34/yd ³	217,600
Loading and hauling	6,400 yd ³ at \$10/yd ³	<u>64,000</u>
TOTAL		\$384,000

Turbine Building Auxiliary Bay

This structure is 40 ft x 206 ft x 70 ft, and is 3 stories high. Remove roof and walls and steel framing at \$1.50/ft² per floor. Ball concrete slabs and walls at \$25/yd³. Load and haul at \$10/yd³.

Roof and sidewalls	24,700 ft ² at \$1.50/ft ²	\$37,000
Concrete	1,700 yd ³ at \$35/yd ³	<u>59,500</u>
TOTAL		\$96,500

Turbine Building

Use the same procedure as for Auxiliary Building.

116 ft x 304 ft = 35,000 ft² area

The turbine mat appears to be more than 3 ft below grade. Therefore, blast it to provide cracks and holes for percolation of water and abandon in place. The estimated cost to remove the building roof and sidewalls is \$4/ft² down to elevation 95 ft, and \$1/ft² for sidewalls from elevation 95 ft to 65 ft. Remove mechanical plus electrical at no net cost by receiving scrap metal credit. Drill and blast concrete walls and pedestal bases vertically with an air track drill at an estimated cost rate of \$16/yd³. Secondary concrete balling and balling of floors is estimated at \$20/yd³. Load and haul for \$10/yd³. Concrete quantities are 9,100 yd³ in pedestals, and 6,600 yd³ in structure.

Roof and sidewalls	35,000 ft ² at \$4/ft ²	\$140,000
Roof and sidewalls	25,000 ft ² at \$1/ft ²	25,000
Concrete	9,100 yd ³ at \$46/yd ³	418,600
Concrete	6,600 yd ³ at \$30/yd ³	<u>198,000</u>
TOTAL		\$781,600

Condensate Demineralizer Building

Remove roof, walls and equipment.

Drill outer walls and interior walls vertically to depth of removal and blast.

Ball floors, blast base floor for percolation.

Leave fill below elevation 40 ft.

Cost estimate assumes \$16/yd³ to drill and blast and \$20/yd³ to ball, separate, load and haul, or \$6/yd³ to separate and bury.

Roof and wall removal at \$1.50/ft² of floor area per floor.

142 ft x 34 ft x 3 at \$1.50 = \$21,720.

Estimated concrete above break line --	\$36/yd ³ x 105 yd ³	\$ 3,780
TOTAL		~\$25,500

Intake Structure

The intake structure, pump bay and pipe structures are estimated to be drilled and blasted for \$14/yd³ and balled, cut, loaded and hauled for an additional \$40/yd³, or \$6/yd³ to ball and bury in place. In the case of buried intake steel pipes, they will have to be drilled, cleaned off, then cut and lifted, and the concrete rubble balled back into the hole. Shaped explosives could be used for opening the base slabs for percolation if needed.

Intake Piping

Removal of concrete to permit removal of steel pipe is estimated at \$14/yd³ to drill and blast plus \$25/yd³ for removal. Add \$10/yd³ for loading and hauling, or \$2/yd³ for backfilling.

Barge Loading Facility

Use the same technique as for the intake structure. Drill and blast at an estimated \$14/yd³. Bulldoze and cover over at \$6/yd³, or load and haul at \$40/yd³.

Administration Building

Demolition and removal is estimated at \$1.50/ft² of each floor area, 95 ft x 48 ft/2 story. However, this type of building should or may be sold, moved and reused. Possibly only the upper floor can be moved.

If the structure is moved, site cleanup and slab and footing removal is estimated at \$20/yd³ of concrete content.

9,120 ft ²	at \$1.50/ft ²	Costs \$13,680
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Shop and Warehouse

This type of building can generally be sold for removal costs, but the ground slab and footings would cost an estimated \$20/yd³ for breaking, loading and hauling for cleanup.

Cleanup Estimate = \$2,000

Chlorine Building

Removal is estimated at \$1.50/ft² building area.

3,840 ft ²	at \$1.50/ft ²	Costs \$5,760
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G.4.5.4 Area Backfill

Earth loaded, hauled, placed and graded is estimated to cost \$5/yd³ bank measure. This equates to about \$4/yd³ truck measure. These unit cost figures may be raised or lowered depending upon the restrictiveness or permissiveness of the fill specifications.

It would be most economical to use concrete rubble for fill in areas more than below grade and to only haul in dirt fill for the top. If possible, the dirt fill should be hauled to the site on the return trip when hauling out the excess concrete rubble.

The areas underlying the structures to be removed are estimated at approximately 10,000 yd², which at 3 ft depth amounts to 10,000 yd³ bank measure.

10,000 yd³

at \$5/yd³

Costs \$50,000

Removal and covering of road areas are not included.

G.4.5.5 Judgment Factors

Estimating. There is no precedent to rely on for a job of this magnitude; therefore, judgments are formed based upon a reasonable understanding of basic physical principles together with experience and observation in applying these principles to the demolition of structures of a wide range of types.

An area of judgment of particular concern in estimating costs is that used for balling and separating the reinforced concrete both before or after it has been weakened by explosives. The unit costs used of \$25 to \$40/yd³ may be somewhat high--particularly when compared with removal costs for unreinforced concrete in easily accessible locations, which is about \$2 to \$5/yd³. The difficulties involved may be overestimated by as much as \$10/yd³. With some 97,000 yd³ involved, this could classify \$970,000 as a contingency cost. Initial blast results would greatly affect this cost item. The reasoning used in this study is that this concrete is all high test concrete with an initial strength of 3,000 psi and better, extra thick, well bonded to extra heavy reinforcing and well aged. These factors tend to increase costs rapidly as single factors; their combination compounds the difficulty. Also, most of the structures have confining and self-reinforcing cross walls that restrict access. The concrete is all tough enough to be classified as "\$100/yd³ concrete"--the kind that is very difficult to demolish. The factors of best judgment, experience, and intuition make up what is believed by the author to be representative of what the actual costs will be.

Hauling Costs. Costs of loading and hauling can be more readily compared to estimated norms. Hauling 30,000 lb in demolition truck trailers is more normal than the 40,000 lb assumed for this study due to the heavy construction of the demolition trailers. The trucks do not load as fast nor dump as fast as dirt and rock hauling trucks. This is due to longer trailers,

higher sides on the trailers and to the uneven sizes and weights of material hauled; in this case, unevenly broken concrete, rebar protruding from concrete, and salvage loads of twisted steel, rebar and sheetmetal. A full load of twisted rebar does not add up to the 40,000 lb, or even 30,000 lb, nor does a load of ductwork. Cost figures in the wrecking industry reflect this condition and are higher than comparable figures from excavation haulers. For a job of this magnitude, equipment designed for the particular job could be obtained and efficiencies improved. Nevertheless, one should expect higher costs for demolition hauling than for new materials or for dirt and rock hauling.

Salvage. Salvage is ordinarily used as a contingency/profit item in the wrecking business due to the inherent risks of structures being tougher than estimated, innovative methods that are untried, human error and variations in employee productivity. In a case where salvage is easily separable from the wrecking, as in the case of machinery and substantial mechanical/electrical salvage, it must be given separate consideration. When structural salvage is readily apparent as a substantial consideration, it is generally offset against the overall cost of demolition through a lessened rate per unit cost rather than itemized as a separate item. This is justified on the basis that the degree of effort to preserve the highest salvage value is directly reflected in higher direct demolition costs. In cost-plus contracts, its value is tallied exactly as received to reduce the cost.

The following method to account for salvage metal appears reasonable and was used in this study: the basic cost of handling and loading scrap (\$18 to \$20 per net ton) is deducted from the published scrap market price, less shipping cost, with the balance credited to the owner. Under this formula, the wrecking contractor is assured of his basic costs, and is rewarded for preserving the highest salvage value, while the owner is assured of a fair allowance for his scrap and benefits fully from rising scrap markets. Using this salvage formula with a \$20/ton cost of removal, together with the fact that the currently depressed steel scrap market still is \$40 to \$45/ton (the West Coast), the salvaged metal would pay for its own removal and yield a credit to the owner of about \$20/ton for all that could be saved.

In calculating the difficulties in removing the various reinforced concrete structures, the cost of separation of the concrete from the steel has been included in the cubic yard price of concrete removal. The recovery of steel reinforcement beyond that point would be charged at the \$20/ton cost of recovery and salvage. Since it pays for itself, the estimates associated with abandoned piping and equipment, rebar and structural steel removal adds no net cost to decommissioning.

Quantity estimates of salvageable rebar should exclude that which can be abandoned below grade, should discount the quantities smaller than 1-in. diameter to about 30% salvage, and larger than 1-in. to about 60%, except that the Cooling Tower and the RCV could yield a higher percentage above their bases. Another possible exception, of course, would be that in the event of successful application of the innovative heat procedure, close to 100% of the RCV rebar could be saved.

Labor Rates. The primary labor rates currently paid in the Seattle area, by the Northwest Chapter of the Association of General Contractors (AGC), on which the cost estimates are based are as follows:

Teamsters - Demolition, 50 yd ³ trucks	(\$11.10 base pay)	\$15.31 hr
Operating Engineers - Operator	(\$11.08 base pay)	\$15.50 hr
Laborers - Basic	(\$8.36 base pay)	\$12.39 hr
- torch premium	(\$8.61 base pay)	\$12.74 hr

These pay scales include base pay, FICA, Unemployment Insurance, Union, State Industrial, Bodily Injury and Property Damage Insurance, and Medical Aid. They do not include supervision, overhead and burden nor profit.

List of Contractor Major Equipment Needed:

200 ton crane	Clamshells
(1) 100 ton conventional crane	Track drills
(1) 65 ton conventional crane	Breaking balls
(2) 45 ton conventional cranes	Compressors
(2) 50 ton hydraulic cranes	Battering ram (w/RCV balling option)
(2) 30 ton hydraulic cranes	Service truck

(2) 977 track loaders	Rigging
(2) 983 track loaders	(2) high lift "giraffes"
(10) Demolition trailers w/ tractors	(2) Selmas man lift platforms
(40) Oxyacetylene cutting outfits	(4) "man cans" (crane held work boxes)

REFERENCES

1. K. J. Schneider and C. E. Jenkins, Technology, Safety and Costs of Decommissioning a Reference Fuel Reprocessing Plant. NUREG-0278, pp. 7-129, U.S. Nuclear Regulatory Commission, September 1977.
2. J. M. McFarland, Report on Representative Cost Estimates for Demolition of Structures at a Pressurized Water Reactor Site. PNL-2450, prepared for Battelle, Pacific Northwest Laboratory by McFarland Wrecking Corporation, Seattle, WA 98108, September 30, 1976.

APPENDIX H

SAFE STORAGE AND DEFERRED DISMANTLEMENT ACTIVITIES, MANPOWER AND COSTS

The information contained in this appendix forms the bases for the cost and occupational radiation exposure estimates given in Sections 10 and 11 respectively of Volume 1. The decontamination procedures given in section F.1 of Appendix F, and the facility description given in Appendix A, provide the bases for the development of the detailed work schedules, manpower requirements and salary costs for Safe Storage activities presented in this appendix. Using these detailed work schedules and manpower estimates, and the radiation dose rate data from Section 7 and Appendix C, the detailed estimates of accumulated occupational radiation exposure are developed.

The detailed cost estimates for chemical decontamination of the RCS and CVCS, for disposal of radioactive materials, and for environmental surveillance and quality assurance programs are developed, and the details of the cost estimates for the period of continuing care are presented.

H.1 SCHEDULES, MANPOWER REQUIREMENTS, AND SALARY COSTS FOR SAFE STORAGE

Development of the schedule and manpower requirements for the preparations for Safe Storage requires a certain amount of iteration to arrive at a reasonably optimum plan. The sequence in which the various systems/buildings must be placed in a safe storage condition is determined, and the manpower needs and time frame for each system/building is estimated. The job sequences are arranged to require a relatively constant-sized work force. The resultant schedules, manpower requirements, and salary costs for the major job sequences are presented in the following sections.

H.1.1 Reactor Building Preparations for Safe Storage

Chemical decontamination of the RCS, CVCS and inter-tied systems is the first major step following the defueling and disabling of the reactor. Chemical decontamination of the RCS, CVCS and inter-tied systems, using an EDTA/oxalic acid/citric acid solution, is described in Appendix F.1 and

is not repeated here. The sequence and schedule of events for preparing the Reactor Building for the continuing care period is given in Figure H.1-1. The staff labor requirements and the resulting salary costs (based on Table I.1-1, Appendix I) are also given.

ASSUMPTIONS

1 CALENDAR MONTH = 22 WORKING DAYS

1 WORK WEEK = 5 CALENDAR DAYS

NORMALLY 2 EIGHT-HOUR SHIFTS/WORKING DAY

(UNLESS NOTED OTHERWISE)

THE SYMBOLS AT THE FRONT OF EACH BAR HAVE THE FOLLOWING MEANING:

D DECOMMISSIONING CREWS

HP HEALTH PHYSICS STAFF

C CONTRACTOR STAFF

^(a) 2 TEN-HOUR SHIFTS/DAY DURING DEFUELING AND DISABLING

^(b) STEP D IS COVERED ON A 2 SHIFTS/DAY 2 MEN/SHIFT (MAN DAYS/DAY). STEP E IS COVERED ON A 2 SHIFTS/DAY BASIS 4 MEN/SHIFT (8 MAN DAYS/DAY).

STEP F IS COVERED ON A 3 SHIFTS/DAY BASIS 2 MEN/DAY BASIS 2 MEN/SHIFT

^(c) ON DUTY AND AVAILABLE AS REQUIRED MAY HELP SUPERVISE MANY WORK ITEMS DURING THE TIME PERIODS GIVEN

^(d) A DASH MEANS THAT CATEGORY OF DECOMMISSIONING STAFF ARE NOT ACTIVELY INVOLVED IN THAT PARTICULAR JOB ITEM

^(e) FOR DEFUELING OPERATIONS THESE STAFF MEMBERS ARE LICENSED REACTOR OPERATORS. ONCE THE REACTOR IS DEFUELED AND DISABLED 12 OF THEM ARE NO LONGER REQUIRED AND THE REMAINING 8 BECOME DECOMMISSIONING STAFF UTILITY OPERATORS AND THE TOTAL DECOMMISSIONING STAFF WORK FORCE IS SUBSEQUENTLY REDUCED TO 37 IN NUMBER

FIGURE H.1-1. Schedule Sequence, Manpower Utilization, and Salary Cost Estimates for Preparing the Reactor Building for Safe Storage

H.1.2 Fuel Building Preparations for Safe Storage

The start of decommissioning of the Fuel Building is dependent on the assumption that two casks (GE IF-300 or equivalent) are available for shipping spent fuel assemblies to a spent fuel repository. Two casks, each capable of transporting seven PWR fuel assemblies by rail, are assumed to have a complete turnaround time of 18 days per shipment, including two days for loading/unloading operations. Since fuel shipment is on the critical path for placing the facility in Safe Storage, the availability of more than two shipping casks could reduce the total time required to complete preparations for the continuing care period. It is further assumed that the fuel shipment operations include a final fuel inventory for accountability purposes. Preparations for safe storage of the Fuel Building begin immediately after the final fuel shipment leaves the site. The schedule sequence and manpower utilization estimates are given in Figure H.1-2, together with the estimated salary costs.

H.1.3 Auxiliary Building Preparations for Safe Storage

The bulk of the radioactive waste treatment systems equipment is located in the Auxiliary Building. Since these systems are required to be in service during much of the plant decommissioning effort, final preparations for the safe storage of the Auxiliary Building occurs rather late in the overall schedule. The schedule sequence and manpower utilization estimates are given in Figure H.1-3, together with the salary costs.

H.1.4 Decommissioning of Other Buildings

Covering of windows and other openings, installation of fire and intrusion alarms, and a final radiation survey are assumed to be all that is required to place the non-contaminated buildings, listed below, into Safe Storage:

- Condensate-Demineralizer Building
- Administration and Warehouse Buildings
- Control Building.

ASSUMPTIONS

1 CALENDAR MONTH - 22 WORKING DAYS

1 WORK WEEK - 5 CALENDAR DAYS

2 SHIFTS /WORKING DAY

^{a)} 0.45 (10) MEANS 0.45 CALENDAR MONTHS, FOR A TOTAL OF 10 WORKING DAYS REQUIRED FOR THE JOB ITEM^{b)} INCLUDES: DRAINING SFP, DECONTAMINATION OF LINER AND RACKS, IMMOBILIZATION OF REMAINING RESIDUAL CONTAMINATION, AND INSTALLATION OF SFP COVER. CONTRACTOR ASSUMED TO PROVIDE H. P. COVERAGE FOR THIS JOB ITEM

THE SYMBOLS AT THE FRONT OF EACH BAR HAVE THE FOLLOWING MEANING

D - DECOMMISSIONING CREWS

HP - HEALTH PHYSICS STAFF

C - CONTRACTOR STAFF

FIGURE H.1-2. Schedule Sequence, Manpower Utilization, and Salary Cost Estimates for Preparing the Fuel Building for Safe Storage

FIGURE H.1-3. Schedule Sequence, Manpower Utilization, and Salary Cost Estimates for Preparing the Auxiliary Building for Safe Storage

It is assumed that some decontamination may be required in the Turbine Building because of the possibility of contaminated leakage from the secondary coolant systems. The schedule sequence and estimated manpower utilization for the decommissioning of these buildings is given in Figure H.1-4, together with estimated salary costs.

ASSUMPTIONS

- 1 CALENDAR MONTH - 22 WORKING DAYS
- 1 WORK WEEK - 5 CALENDAR DAYS
- 1 SHIFT/WORKING DAY UNLESS INDICATED OTHERWISE

THE SYMBOLS AT THE FRONT OF EACH BAR HAVE THE FOLLOWING MEANING

- D - DECOMMISSIONING CREW
- HP - HEALTH PHYSICS STAFF
- C - CONTRACTOR

(a) AFTER DECONTAMINATION (AS REQUIRED) AND DECOMMISSIONING IS COMPLETED ON THESE BUILDINGS, A SECURITY CONTRACTOR INSTALLS, FUNCTIONALLY TESTS, AND ASSUMES SECURITY RESPONSIBILITIES FOR THE BUILDINGS

(b) A DASH MEANS THAT CATEGORY OF DECOMMISSIONING STAFF ARE NOT ACTIVELY INVOLVED IN THAT PARTICULAR JOB ITEM

FIGURE H.1-4. Scheduel Sequence, Manpower Utilization, and Salary Cost Estimates for Preparing Other Buildings for Safe Storage

H.1.5 Summary of Manpower Requirements

The individual manpower utilization estimates, presented as parts of Figures H.1-1 through H.1-4, are collected and summarized in Table H.1-1. These data, together with the radiation exposure rate data from Section 7 and Appendix C are used to develop the detailed estimates of external occupational radiation exposure, presented in the next section.

H.2 ESTIMATES OF EXTERNAL OCCUPATIONAL RADIATION EXPOSURE FOR THE PREPARATIONS FOR SAFE STORAGE OF THE REFERENCE PWR

Detailed estimates have been developed for the external occupational radiation exposure that will be accumulated by the decommissioning workers during the preparations for Safe Storage. These estimates are based on a task by task analysis of personnel hours and expected radiation dose rates associated with each task for all decommissioning personnel. Certain basic assumptions were made in developing these estimates. 1) Efforts are made to implement the philosophy of ALARA (As Low As Reasonably Achievable) when performing an operation. Thus, extensive use is made of temporary shielding and remote handling techniques and equipment, and emphasis is placed on staying out of radiation fields when not actively participating in the work. 2) The chemical decontamination campaign is reasonably successful, reducing all radiation dose rates from piping systems and components by at least a factor of 10, and more probably a factor of 50 to 100 in many locations. 3) Careful and current accounting of radiation exposure is maintained to rapidly identify jobs that are causing high radiation exposure and permit corrective action to be taken. The accounting system also prevents undue accumulations of radiation exposure to particular personnel by flagging potential problem areas.

The results of these analyses are presented in Table H.2-1. These detailed estimates are summarized in Table 11.3-2.

TABLE H.2-1. Estimated Occupational Radiation Doses for Preparations for Safe Storage^(a,b)

EVENT DESCRIPTION	DECOMMISSIONING PERSONNEL													EVENT TOTAL DOSE MAN REM
	SUPERVISORS			DECOMMISSIONING CREW			CRAFTSMEN			MISCELLANEOUS TECHNICIANS				
	AVERAGE EXPOSURE RATE (R/hr)	TOTAL EXPOSURE HOURS	DOSE MAN REM	AVERAGE EXPOSURE RATE (R/hr)	TOTAL EXPOSURE HOURS	DOSE MAN REM	AVERAGE EXPOSURE RATE (R/hr)	TOTAL EXPOSURE HOURS	DOSE MAN REM	AVERAGE EXPOSURE RATE (R/hr)	TOTAL EXPOSURE HOURS	DOSE MAN REM		
REACTOR BUILDING														
COMPREHENSIVE PREDECOMMISSIONING SURVEY ^(c)	0.03	44	1.32	d1							0.03	176	5.28	6.6
DECONTAMINATE REACTOR CAVITY AND REFUELING CAVITY (CONTRACTOR)														4
CHEMICAL DECONTAMINATION ^(e) (RCS, CVS, AND RELATED SYSTEMS)														
VALVING READINGS, ETC.				0.025	65	1.6								1.6
EVAPORATORS MAINTENANCE AND REPAIR				0.4	28	11.2	0.4	28	11.2	0.4	6	2.4	24.8	
PURGE RCS AND CVC SYSTEMS	0.003	69	0.207	0.003	138	0.414	0.003	77	0.231	0.003	69	0.207	1.06	
DECONTAMINATE REMAINING INTERNAL SURFACES AND PLACE REACTOR BUILDING IN SAFE STORAGE	0.003	1124	3.372	0.003	2248	6.744	0.003	1264	3.792	0.003	1124	3.372	17.28	
INTRUSION, RADIATION, AND FIRE ALARMS INSTALLATION (CONTRACTOR)											0.003	88	0.264	4.49 ^(f)
FINAL REACTOR BUILDING RADIATION SURVEY											0.003	352	1.056	1.056 63.886
AUXILIARY BUILDING														
COMPREHENSIVE PREDECOMMISSIONING SURVEY	0.005	80	0.4								0.005	640	3.2	3.6
IX RESIN AND FILTERS REMOVAL	0.063	101	6.36	0.1	51	5.1	0.1	60	6	0.083	64	5.31	~23	
DECONTAMINATION (CONTACT WORK AND INTERNAL FLUSHES)	0.009	105	0.945	0.009	300	2.7	0.01	150	1.5	0.008	150	1.2	6.345	
PLACE AUXILIARY BUILDING IN SAFE STORAGE	0.001	23	0.023	0.001	46	0.046	0.001	69	0.069	0.001	23	0.023	0.161	
INTRUSION, RADIATION, AND FIRE ALARMS INSTALLATION (CONTRACTOR)											0.001	30	0.03	0.679 ^(f)
FINAL AUXILIARY BUILDING RADIATION SURVEY											0.001	792	0.792	0.792 ~34.568
FUEL BUILDING														
COMPREHENSIVE PREDECOMMISSIONING SURVEY	0.005	40	0.4								0.005	160	0.8	1.2
DECONTAMINATION (CONTACT WORK AND INTERNAL FLUSHES)	0.02	45	0.9	0.02	298	5.96	0.02	82	1.64	0.02	56	1.12	9.62	
MISCELLANEOUS WORK	0.001	12	0.012	0.001	71	0.071	0.001	27	0.027	0.001	18	0.018	0.128	
PLACE THE SFP IN SAFE STORAGE (CONTRACTOR) ^(h)													5	
INTRUSION, RADIATION, AND FIRE ALARMS INSTALLATION (CONTRACTOR)											0.001	30	0.03	0.679 ^(f)
FINAL FUEL BUILDING RADIATION SURVEY											0.001	360	0.36	0.36 16.978
ANCILLARIES														
RADIATION SURVEYS (WEEKLY) ^(i,j)														
REACTOR BUILDING											0.005	156	0.78	0.78
AUXILIARY BUILDING											0.005	54	0.27	0.27
FUEL BUILDING											0.005	54	0.27	0.27
PACKAGING (COMBUSTIBLE WASTES)														
REACTOR BUILDING				0.03	104	3.12	0.03	5	0.15					3.27
AUXILIARY BUILDING				0.03	40	1.2	0.03	3	0.09					1.29
FUEL BUILDING				0.03	44	1.32	0.03	3	0.09					1.41
FUEL HANDLING														
DEFUEL REACTOR ^(k)	0.009	1232	11.09	0.009	5544	49.9					0.009	1232	11.09	72.08
SHIP FUEL OFFSITE	0.01	324	3.24	0.01	5184	51.84					0.01	972	9.72	64.8
MISCELLANEOUS ^(k)														~170 ~314.17
TOTAL MAN REM												~427 ^(L)		

a) THE NUMBER OF SIGNIFICANT FIGURES CARRIED IS FOR COMPUTATIONAL ACCURACY AND DOES NOT IMPLY PRECISION TO THAT MANY PLACES

b) ASSUMES RCS, CVS, AND RELATED INTERIED SYSTEMS HAVE BEEN CHEMICALLY DECONTAMINATED

c) THE REACTOR BUILDING COMPREHENSIVE PREDECOMMISSIONING RADIATION SURVEY AND THE COMPREHENSIVE PLANT RADIATION SURVEY OVERLAP DURING THE LAST MONTH OF THE PLANT RADIATION SURVEY. THIS IS NECESSARY SO THAT D H P PERSONNEL CAN BE MORE EFFECTIVELY UTILIZED. AND 2) THE FIRST MONTHS RADIATION SURVEY DATA ON THE OTHER BUILDINGS AT THE STATION CAN BE USED FOR PLANNING PURPOSES WHILE WORK PROGRESSES IN THE REACTOR BUILDING. THE REACTOR BUILDING IS THE FOCAL POINT OF THE MOST CONCENTRATED AND IMPORTANT DECOMMISSIONING WORK ITEMS IN THE PREPARATIONS FOR SAFE STORAGE OF THE REFERENCE PWR AND CONSEQUENTLY MAXIMUM EFFORT IS CONCENTRATED EARLY IN THIS BUILDING.

d) A DASH MEANS THAT CATEGORY OF DECOMMISSIONING STAFF ARE NOT INVOLVED IN THAT PARTICULAR JOB ITEM.

e) THE TOTAL MAN REM EXPOSURE FOR THIS ITEM IS FOR THE ENTIRE JOB I.E. FILLING AND DRAINING OF THE SYSTEMS. EVAPORATION AND SOLIDIFICATION OF THE WASTE SOLUTIONS. AND ~25 MAN REM ALLOCATED TO EVAPORATORS REPAIR AND MAINTENANCE.

f) IT IS ESTIMATED THAT CONTRACTOR STAFF INSTALLING ALARM EQUIPMENT ACCUMULATE ~4.49 MAN REM OF THE 4.49 MAN REM ESTIMATED FOR THIS JOB. THEY WILL BE COVERED ON THE JOB BY DECOMMISSIONING STAFF HEALTH PHYSICS PERSONNEL.

g) IT IS ESTIMATED THAT CONTRACTOR STAFF INSTALLING ALARM EQUIPMENT ACCUMULATE ~0.64 MAN REM OF THIS TOTAL AMOUNT. THEY ARE ASSURED TO BE COVERED ON THE JOB BY DECOMMISSIONING STAFF H P PERSONNEL.

h) INCLUDES DRAINING SFP, DECONTAMINATION OF LINER AND RACKS, IMMOBILIZATION OF REMAINING RESIDUAL CONTAMINATION, AND INSTALLATION OF SFP COVER. CONTRACTOR ASSURED TO PROVIDE HIS OWN H P COVERAGE FOR THIS JOB ITEM.

i) IT IS ANTICIPATED THAT WEEKLY RADIATION SURVEYS IN EACH OF THE THREE MAJOR BUILDINGS, I.E. REACTOR, AUXILIARY, AND FUEL, MUST BE COMPLETED AS DECOMMISSIONING WORK PROGRESSES DURING EACH BUILDING'S RESPECTIVE TIME SCHEDULE FOR PLANNING AND PREPARATION PURPOSES.

j) ESTIMATED FROM CAROLINA POWER AND LIGHT COMPANY YEAR-END DATED JANUARY 6, 1976, FILE NO. 3600X1, SERIAL NO. 76-013, RADIATION EXPOSURE BREAKDOWN DURING THE H B ROBINSON UNIT 2 REFUELING OUTAGE IN 1975. CPL. RALEIGH NO. 27602.

k) DERIVED FROM THE REFERENCE GIVEN IN (j). THE ~170 MAN REM SHOWN COULD VARY UPWARD BY A FACTOR OF TWO OR MORE DEPENDING ON 1) SHUTDOWN RADIATION CONTAMINATION LEVELS, AND 2) THE APPLICATION OF ALARA CONSIDERATIONS IN THE REFERENCE USED. BOTH PLANT AND CONTRACTOR MANPOWER WERE UTILIZED AND THEREFORE, IT IS VERY DIFFICULT TO ESTABLISH EITHER AVERAGE EXPOSURE RATES OR TYPES OF PERSONNEL UTILIZED DURING THE REFERENCED REFUELING OUTAGE. SUCH AN ANALYSIS IS BEYOND THE SCOPE OF THIS STUDY. HOWEVER, THE BASIS USED TO SELECT WHICH MISCELLANEOUS JOB ITEMS SHOULD BE INCLUDED HERE ARE BASED ON THE ASSUMPTION THAT ONCE THE REACTOR IS DEFUELED, THE REACTOR VESSEL HEAD WILL BE REPLACED IN PREPARATION FOR THE RCS, CVS, AND RELATED SYSTEMS CHEMICAL DECONTAMINATION WORK.

L) THE THREE SIGNIFICANT FIGURES SHOWN ARE THE SUMMATION OF MANY INDIVIDUAL ESTIMATES AND DO NOT IMPLY THREE PLACE PRECISION IN THE ESTIMATED VALUE.

TABLE H.2-1. Estimated Occupational Radiation Doses for Preparations for Safe Storage

H.3 ESTIMATED COSTS FOR THE PREPARATIONS FOR SAFE STORAGE OF THE REFERENCE PWR

The components of cost other than salary for the preparations for Safe Storage are compiled and presented in this section. In some instances, the bases for the cost value are developed in other sections, and those sections are identified for reference. The results of these analyses are summarized and presented in Section 10.2.

H.3.1 Cost of Chemical Decontamination

The complete chemical decontamination effort includes the addition of the appropriate chemicals to the water in the RCS, and CVCS and inter-tied systems; the circulation of the decontamination solution through the systems for about a week; draining, flushing, and draining of the systems; and disposal of the decontamination and flush solutions. The cost of the last item, disposal, is included in Appendix H.3.2, under Disposal of Radioactive Wastes, since the same volumes of liquid are treated, whether or not a chemical decontamination is carried out.

The initial cost item is chemicals. The unit cost of EDTA/oxalic acid/citric acid, given in Appendix I.6, is \$1.30/kg of the 1:1:1 mixture by weight. The required concentration of these chemicals in the solution is from 1 to 5% by weight in 378 m^3 (100,000 gallons) of water. Thus, the cost of chemicals can range from \$23,000 to \$115,000.

Circulation of the solutions and rinses was assumed for a total of 10 to 16 days. The maximum cost for electricity, at \$15/MWh, is \$105,000.

The staff labor cost, developed in Figure G.2-1 Appendix G, is estimated to be \$64,050. The total costs directly chargeable to chemical decontamination are summarized in Table H.3-1. The larger amount in Table H.3-1 is used in other cost summaries for conservatism.

TABLE H.3-1. Summary of Chemical Decontamination Costs

<u>Cost Item</u>	<u>Estimated Cost (\$)</u>
EDTA/oxalic acid/citric acid	23,000 to 115,000
Electrical Power	105,000
Staff Labor	64,050
Total	192,050 to 284,050

H.3.2 Disposal of Radioactive Wastes

The radioactive wastes considered are the following: used ion exchange resins, cartridge filters, evaporator bottoms liquids, and miscellaneous combustible wastes such as plastic sheeting, rags, disposable protective clothing, etc. Also considered is the cost of shipment of the spent fuel. The IX resins, filters, and evaporator bottoms liquids are mixed with or encapsulated in solid materials such as cement or ureaformaldehyde to assure retention of radioactive materials within the shipping containers after disposal. The combustible wastes are reduced in volume as much as possible by compaction with existing plant equipment prior to packaging.

A breakdown of the number of containers, volumes of materials and disposal costs is given in Table H.3-2 for disposal of wastes. About 139 truckloads of radioactive waste will be shipped which will occupy about 388 m^3 ($13,710 \text{ ft}^3$) at the disposal site. The development of the costs for these materials is given in the following sections.

Combustible Wastes. Moderate quantities of contaminated combustible wastes will be produced during the defueling, decontamination, and other decommissioning activities. These materials are assumed to be compacted by a factor of 5 to 1 in 208 l (55-gal) drums prior to shipment. Estimates have been made of the number of drums of dry, uncompacted waste produced in the three major buildings: Reactor Building (740; includes ~100 drums for reactor defueling/disabling operations), Fuel Building (308), and Auxiliary Building (280). Compacted waste results in 266 drums, each having a final disposal volume of 0.21 m^3 (7.5 ft^3) per drum, and occupying a total of 55.86 m^3 (1995 ft^3) in the burial site. Of these, 133 are assumed to have

TABLE H.3-2. Disposal of Radioactive Wastes from Preparations for Safe Storage

Material	No. of Disposable Containers	Container ^(a,b) Cost (\$)	Cask ^(c) Rental (\$)	No. of Shipments	Transportation ^(d) Cost (\$)	Handling ^(e) Cost (\$)	Burial ^(f,g) Volume (ft ³)	Burial ^(f,g) Cost (\$)	Liner ^(e) Surcharge (\$)	Curie ^(e) Surcharge (\$)	Total Cost (\$)
Combustible Wastes	266	5,320	9,500	19	35,340	3,800	1,995	5,287	--	--	59,247
Spent Resins	20	18,500	10,000	20	37,200	4,000	2,000	5,300	9,500	7,800	92,300
Spent Filter Cartridges	42	4,200	3,000	6	11,160	1,200	315	835	5,250	1,960	27,605
Evaporator Bottoms Liquids	94	86,950	47,000	94	174,840	18,800	9,400	24,910	12,700	----	365,200
TOTAL	422	114,970	69,500	139	258,540	27,800	13,710 (388 m ³)	36,332	27,450	9,760	544,352

(a) Container costs from Table I 2-1

(b) Includes solidification cost from Section I 2 2, where applicable

(c) Based on Table I 3-1

(d) Based on Table I 4-4

(e) Based on Table I 5-1

(f) Volume includes disposable container

(g) 1 ft³ = 0.028317 m³

radiation readings sufficiently high to require shipment in a shielded cask. The remaining 133 drums are shipped on the same trucks, unshielded. All truck shipments are assumed to be over legal weight. The purchase price of the drums was assumed to be \$20.00 each, and the burial charges were assumed to be \$2.65/ft³, for costs of \$5,320 and \$5,287, respectively. Nineteen shipments by truck would be required to transport the drums 800 km (500 miles) one-way to the burial site, at a cost of \$1,860 per trip, or \$35,340.

Spent Resins. The total volume of expended resins for disposal has been assumed to be 30 m³ (1060 ft³), or about 3 times the volume generated annually during reactor operation.⁽¹⁾ The resins are estimated to contain up to 42,000 Ci of radioactivity. The average curie content per container is assumed to be 2,100 Ci. The resins are assumed to be mixed with a solidifying agent such as cement or ureaformaldehyde (UF) and packaged in steel cask liners having a burial volume of 2.84 m³ (100 ft³) per liner. Each container is shipped in a shielded cask, one cask per shipment; each shipment is over legal weight. Twenty containers will be required, at a cost of \$550 for each container and \$375 for the UF in each container, for a total cost per container of \$925. Cask rental for five days per trip, at \$100 per day and 20 trips at \$1,860 per trip are assumed. Thus, the total cost of containers is \$18,500 and the cost of shipping is \$37,200. The cask handling costs total \$4,000. All containers are assumed to have surface radiation dose levels of 50 R/hr. Therefore, the burial costs will be \$5,300, plus a \$475 per container liner surcharge, for a total of \$22,600.

Spent Filters. Forty-two spent filters from fluid streams are assumed to be packaged and shipped to a disposal site. The filters are assumed to contain a total of 5,000 Ci of radioactivity, for an average curie content of about 120 Ci per filter. Each filter is placed within a concrete-lined drum (burial volume 0.21 m³, 7.5 ft³) for shipment. The drums are shipped in a shielded cask, 7 drums per cask, one cask per shipment, over legal weight. The dose rate at the drum surface is assumed to be less than 2 R/hr. Each drum costs \$100 for a total of \$4,200. Cask rental for five days per trip,

⁽¹⁾ TROJAN Nuclear Plant Environmental Report, Volume I, Section 6.3, Portland General Electric Company, November 1971.

at \$100/day, totals \$3,000, and 6 trips at \$1,860 each totals \$11,160. The cask handling surcharge is \$200 for each of the 6 shipments, for a total of \$1,200. Burial costs are estimated to be about \$835, plus a curie surcharge of \$1,960 and a liner surcharge of \$5,250, for a total of \$8,045.

Evaporator Bottoms Liquids. The initial plant inventory of water borated to about 0.2% by weight is assumed to be about 3,785 m³ (one million gal.). One RCS volume (~378 m³) also contains decontamination chemicals at the 1% by weight concentration. An additional 3,785 m³ (one million gal.) of water from rinsing, flushing and washing operations is also assumed to be evaporated. The decontamination solution is assumed to be concentrated to 57 m³, the remaining borated water concentrated to 38 m³, and the rinse water to 38 m³, for a total of 133 m³ of evaporator bottoms liquids to be solidified. This liquid is assumed to be solidified in 94 containers (2.83 m³, 100 ft³/each container). Twenty-two containers are assumed to have surface reading dose rates between 0.2 and 1 R/hr, and the remaining 27 containers to have surface radiation dose rates less than 0.2 R/hr.

A summary of the estimated costs for disposal of the contaminated evaporator bottoms liquids is given in Table H.3-2. An itemized disposal cost breakdown for the evaporator waste materials based on the assumed container surface radiation dose rates is presented in Table H.3-3.

TABLE H.3-3. Costs for Disposal of Evaporator Bottom Liquids

Assumed Surface Radiation Dose Rate (R/hr)	No (f) of Containers	Container ^(a,b) and Solidification Costs (\$)	Cask ^(c) Rental (\$)	Transportation ^(d) Cost (\$)	Handling ^(e) Cost (\$)	Liner ^(e) Surcharge (\$)	Burial ^(e) Cost (\$)
150	22	20,350	11,000	40,920	4,400	10,450	5,830
1	45	41,625	22,500	83,700	9,000	2,250	11,925
0.2	27	24,975	13,500	50,220	5,400	--	7,155
	94	\$86,950	\$47,000	\$174,840	\$18,800	\$12,700	\$24,910
TOTAL COST		\$365,200					

(a) Container costs from Table I 2-1

(b) Solidification costs from Section I 2 2

(c) Assumes 5 days/shipment, rental from Table I 3-1

(d) Assumes overweight shipment, from Table I 4-4

(e) Costs from Table I 5-1

(f) One container in cask per shipment.

Spent Reactor Fuel

Since the ultimate disposition of spent reactor fuel is still an unknown, for this study it was assumed that the 193 fuel assemblies are shipped in two IF-300 casks, in shipments of 7 assemblies per cask, to a repository located 2,400 m (1,500 mi.) from the reactor site, using rail transportation. Thus, 28 shipments, requiring 18 days per round trip are needed. Based on the cask rental cost given in Table I.3-1 of \$3,500 per day, the total cask rental is \$1,764,000. The rail freight cost, from Appendix I.4.2, is \$702,800. Thus, the cost for removing the fuel from the reactor and transporting the fuel to a repository is \$2,466,800. This cost does not include any charges associated with handling at the repository. The unencapsulated volume of the fuel is about 37 m³.

H.3.3 Potential Cost Postponements by Using On-Site Retrievable Storage (an option)

Use of the Containment Building Refueling Cavity as an onsite retrievable storage location is worthy of consideration. The cavity dimensions are approximately 6.1 m (20 ft) x 11.3 m (37 ft) x 11.3 m (37 ft) or 778.9 m³ (27,380 ft³). Assuming 2.84 m³ (100 ft³) containers are about 1.4 m (5 ft) in dia by 1.9 m (6 ft) high, 24 containers/layer could be stored with up to 6 layers, for a total of 144 containers. The volumes of solidified waste anticipated to be stored together with the savings estimated from use of the onsite retrievable storage are shown in Table H.3-4. It is readily seen that the anticipated volume of all solidified wastes shown in Table H.3-4 can be stored onsite in an easily shielded, secured location.

TABLE H.3-4. Potential Costs Postponed by Using Onsite Retrievable Storage

Number of Containers	Volume		Material	\$ Savings
	m ³	ft ³		
94 @ 2.83 m ³ (100 ft ³)/each	266.02	9,400	Evaporator Bottoms	199,750
20 @ 2.83 m ³ (100 ft ³)/each	56.6	2,000	Resins	42,500
42 @ 0.22 m ³ (7.5 ft ³)/each	9.24	315	Spent Particulate Filter Cartridges	11,995
TOTAL	331.86	11,715		\$254,245

The reactor vessel interior is another potential storage location. Within the core shroud is an area of approximately 33 m^3 (1165 ft^3) that could be utilized for storage of radioactive and contaminated waste materials. Similarly, the Spent Fuel Pool (SFP), with dimensions of 9.14 m (30 ft) x 12.19 m (40 ft) x 12.19 m (40 ft), could be utilized for retrievable onsite storage. Precast concrete slabs could be used for top shields and a sealing surface.

If the Onsite Retrievable Storage option is selected for use, the shipping and burial expenses, which amount to about \$250,000, could be postponed. Selection would be contingent upon the type of wastes involved; i.e., those that do not contain high-level radioactive fission product waste material as defined in 10 CFR 50, Appendix F. Recognizing that the ideal situation is one in which the only permanently committed land is that used for a federal nuclear waste repository, it should also be pointed out that as long as millions of curies are confined in the reference PWR facility, that site is already committed far into the future once Safe Storage is selected as the interim decommissioning mode. Ultimately, this material must be shipped to a burial site, but the delay will reduce radiation levels, shielding costs, and curie surcharges at the burial site.

H.3.4 Cost of the Quality Assurance Program

The principal costs of the QA program are the staff labor to execute the program. From Table 10.2-2, the program outlined in Appendix F.4 requires 2.53 man years of a QA Supervisor/Engineer, and 2.83 man years of QA Technicians. Not explicitly included is the secretarial assistance that would be needed. The total cost of the QA staff labor is estimated to be about \$177,000. Details of the total QA cost estimate are presented in Table H.3-5.

Office space requirements are not included in this estimate. Trailers and suitable vault storage for QA records must be considered if facilities are not available onsite.

TABLE H.3-5. Quality Assurance Cost Breakdown
During the Decommissioning Period

<u>Cost Item</u>	<u>Cost (\$)</u>
Quality Assurance Supervisor/Engineer (2.53 MY)(a)	110,814
QA/QC Audit and QC Inspection (2.83 MY)	65,656
Materials and Equipment	5,000
Travel Expenses	5,000
Subtotal	186,470
25 Contingency	<u>46,618</u>
TOTAL	\$233,088

(a) MY is man year.

H.3.5 Cost of the Environmental Surveillance Program

The environmental surveillance activities, as described in Appendix F.5, would continue at the operational level of effort until the final shipment of fuel from the site. At that time, approximately one year after reactor shut-down, the program would be reduced in scope. The first year's decommissioning monitoring costs are given in Table H.3-6. The costs shown assume the environmental surveillance is accomplished by the same specialty subcontractor who will provide the environmental monitoring services during the continuing care period.

TABLE H.3-6. Costs of the Environmental Monitoring
Program for the First Year of Decom-
missioning Activities

Labor	\$33,000
Equipment and Supplies	1,000
Utilities and Services	<u>1,000</u>
Subtotal	35,000
Contractors' 10% Overhead and Profit	<u>3,500</u>
TOTAL	\$38,500

H.4 ESTIMATED MANPOWER UTILIZATION COSTS AND OCCUPATIONAL RADIATION EXPOSURE FOR THE CONTINUING CARE PERIOD OF SAFE STORAGE

The information contained in this section forms the bases for the summaries of cost and occupational radiation exposure estimates given in Sections 10 and 11 of Volume 1, for the continuing care period.

Preparations for surveillance and maintenance are detailed in Section 9.2 and are not repeated here. Details of security preparations and estimated security program costs during the continuing care period of Safe Storage are presented in this section and are summarized in Sections 9.2 and 10.3 respectively.

H.4.1 Estimated Manpower Requirements and Costs for The Continuing Care Period of Safe Storage

The organization shown in Figure H.4-1 is postulated to take over the surveillance, maintenance, and security for the duration of the continuing care period of Safe Storage. The estimated manpower requirements and annual costs for continuing care of the reference PWR analyzed in this study are summarized in Table H.4-1.

NOTE: EQUIVALENT MAN YEARS/YEAR INDICATED IN PARENTHESES ().

FIGURE H.4-1. Postulated Staff Organization for the Continuing Care Period of Safe Storage

TABLE H.4-1. Estimated Annual Manpower Utilization and Surveillance, Maintenance and Security Costs During the Continuing Care Period of Safe Storage

<u>Labor</u>	<u>Cost (\$)</u>
Surveillance and Maintenance (S&M) Representative (0.25 Man Years)	6,500
Health Physicist (0.1 MY)	3,940
Secretary (0.25 MY)	5,075
Repairman (0.1 MY)	2,710
Security (0.3 MY) (See Table H.4-3, Option C for details)	8,800 ^(a)
Third Party Inspection (0.06 MY) ^(b)	7,500
Environmental Radiological Monitoring Program Personnel (See Table H.4-2. for details) (0.35 MY)	16,434 ^(c)
Quality Assurance Specialist	900
Other Costs:	
Equipment and Supplies	1,000
Annual Allowance for Repairs	5,000
Utilities and Services	5,000
NEL-PIA Insurance	<u>1,000</u>
Subtotal	63,859
25% Contingency	<u>15,965</u>
TOTAL	\$79,824

^(a) Refer to Section H.4-2. for initial costs.

^(b) Third party inspection costs are based on an assumed cost of \$500 per man day.

^(c) Refer to Section H.3.5 for initial costs during the first year after reactor shutdown.

The estimated annual manpower utilization, activities, and costs of the ongoing environmental monitoring program during the continuing care period are given in Table H.4-2.

TABLE H.4-2. Estimated Annual Manpower, Activities, and Costs of an Ongoing Environmental Monitoring Program During the Continuing Care Period of Safe Storage

<u>Position</u>	<u>Activities</u>	<u>Cost (\$)</u>
Health Physicist (0.05 MY) ^(a)	Collect data. Archive storage of samples and data.	1,450
Health Physicist, Supervisor (0.1 MY)	Preparation of reports. Data analysis.	3,940
Radiochemist (0.1 MY)	Laboratory analysis of samples.	3,940
Instrument Specialist (0.1 MY)	Maintain equipment, calibrations.	2,710
Quality Assurance Specialist	Audit Functions.	900
Other Costs:		
Utilities and Services		1,000
Equipment and Supplies		<u>1,000</u>
Subtotal		14,940
Contractors' 10% Overhead and Profit		<u>1,494</u>
Subtotal		16,434
25% Contingency		<u>4,104</u>
TOTAL		\$20,543

^(a)MY is man year.

H.4.2 Estimated Security Costs

The information contained in this section forms the bases for the estimated security costs for the continuing care period of Safe Storage. For this study, it is postulated that at the start of the continuing care period the guard force is disbanded and replaced by private security agency forces and electronic intrusion systems.

Safe Storage security arrangements begin before the reactor is shut down, with the selection of the security agency itself. First, a site survey is conducted by the security agency to determine the scope of surveillance and monitoring required. Factors considered include, but are not limited to:

- The site-specific goals and requirements of the owner
- Knowledge of points of vulnerability (site-specific data)
- Type of hardware and alarm systems (taking into account existing, usable equipment)
- Notification to a central station, a modified central station, or an answering service
- Response times of security agency personnel, local police, fire department, and surveillance and maintenance representative
- Authority and responsibilities of the security agency
- Maintenance, monitoring, and visual patrol needs
- Local laws and ordinances,^(a) or Federal laws and regulations.

Cost comparisons of the use of a single guard working 40 hr per week, a multiple guard force working 24 hr every day, and a private security agency using remote surveillance equipment are given in Table H.4-3 in decreasing order of estimated annual operating costs, after the first year. Consideration was given to the following parameters:

- Renovation and utilization of existing site security and surveillance-related equipment
- Possible relocation of the perimeter fence to include only those buildings housing activated and/or contaminated equipment and areas
- Addition of special equipment as indicated by the security agency's site survey
- Installation of remote readout facilities either onsite or offsite.

(a) For example, a local code may require that fire detection systems and alarms wiring be encased in suitable conduit from the area sensor to the facility receptor.

TABLE H.4-3. Security Program Options, Costs, and Related Considerations^(a)

Option A (Around-the-Clock Guards)		Cost (\$000)	Option B (Day-Shift Guard)		Cost (\$000)	Option C (Electronic Surveillance)		Cost (\$000)
1	Site security survey	2 0 ^(b)	1	Site security survey	2 0 ^(b)	1	Site security survey	2 0 ^(b)
2	Five security personnel, i.e., 1 supervisor and 4 guards (24 hr/day including weekends)	60 0	2	One guard on dayshift (8 hr/day, no weekend coverage)	18 2	2	Reduce circumference of existing security fence	10 0 ^(b)
3	Utilize existing plant equipment, essentially no changes, service and maintenance	10 0	3	Utilize existing plant equipment, essentially no changes, service and maintenance	10 0 ^(c)	3	Utilize portions of existing security equipment, add others	40 0 ^(d)
			4	Install remote readout at offsite location	10 0 ^(b)	4	Install remote readout at offsite location	25 0 ^(b,e)
			5	Private offsite security agency services	5 9	5	Lease of equipment, maintenance, monitoring, response (includes temporary maintenance when required), patrol (unscheduled, any hour), and supervision	8 8
Total (first year)		72 0			46 1			85 8
Total/yr (after first yr)		70 0			34 1			8 8

- Assumes all equipment (installed and/or added) is Underwriter Laboratories (UL) listed
- Affect of option selection on facility insurance is beyond the scope of this study
- All costs are per year unless noted otherwise

(a) Includes onsite detection and notification systems for security (intrusion), fire, and radiation levels

(b) One-time cost only

(c) Cost breakdown: repairman (0.1 man year) \$3,300, annual allowance for repairs \$3,000, equipment and supplies \$3,700

(d) One-time cost only, existing equipment allowances would be a contractual matter open to negotiation between the facility owner and the security agency. This dollar value includes installation of devices and hardware not normally found at the site. Such items include audible alarms, 24-hr video tape recorder, multiplex systems (self troubleshooting), battery backup power where required, and capacitance perimeter fence protection.

(e) Includes installation charges for security agency hardware in addition to Option B, Item 4, which are primarily for the offsite installation costs of 'slave' annunciator drops from existing plant security equipment.

The private security agency operating remote surveillance equipment from their own control center offsite on a 24-hr per day schedule (Option C in Table H.4-3.) appears to be the most cost-effective and performs three vital functions: 1) constant observation, 2) rapid investigation, and 3) notification, as required, to the proper authorities. Reliability of dependable electronic surveillance and notification systems are, therefore, effectively coupled with human judgment.

Option C is essentially an electro-mechanical security system. High security locks, properly installed, provide the first line of defense. They perform a function that the most sophisticated alarm system is not capable of--time delay. If the lock is defeated by an intruder, or even while the attempt is in progress, the alarm system alerts the security agency. During the period from the time of entry to the arrival of the security personnel, other secondary barriers such as concrete walls, welded doors, hatches, plates, etc., prevent the "snatch and run" theft of equipment and minimize the risk of major facility damage. Additionally, the multiple barriers delay unauthorized entry into sealed radiation areas and may prevent intruders from encountering hazards before security personnel can intercept them.

Although higher in initial cost, Option C offers other advantages besides low annual cost. It is estimated that a single guard can make two effective surveillance passes per hour. After each area is observed, the area is, of course, unobserved until the guard returns. Strategically located television cameras utilizing motion detectors that automatically trigger video tape systems for programmed movements can survey the entire PWR site, inside and outside, approximately every 30 sec. This type of service plus leasing of equipment, monitoring, response (including immediate temporary maintenance), visiting unscheduled patrol, and highly trained and skilled supervision is assumed available at the reference PWR site for approximately \$700 per month⁽²⁾. In addition to apprehension of intruders, the video replay portion of the television security system aids in legal convictions in courts of law if the facility owner decides to prosecute.

⁽²⁾ Personal communication: from Mr. A. W. Pound, Chief, Moon Security Service, Inc., Pasco, WA 99301, April 11, 1977.

In summary, security is provided during the continuing care period of Safe Storage by several methods. Locks on the gates in the fence around the decommissioned facility provide the first level of defense. The fence is maintained in good condition throughout the continuing care period. The locks and secured entrance ways of the structures provide the second level of defense. The secured areas within the structures and the remote surveillance equipment and alarm system provides the final level of defense.

It is arranged with the security agency to assure that the Surveillance and Maintenance (S&M) Representative, who is thoroughly knowledgeable about the facility, can be contacted at any time. The S&M Representative is designated by the utility and is responsible for controlling authorized access into and movement within the facility. He is further charged with the responsibilities of appropriate actions and notifications regarding breaches of security, upkeep of plant surveillance and maintenance programs, and administrative reporting of these events as required by state and federal regulations. A postulated administrative reporting process and security notification chain-of-command is shown in Figure H.4-2 delineating responsibilities of reportable events. It is imperative that such a notification and reporting process be kept current throughout the continuing care period.

FIGURE H.4-2. Postulated Security and Administrative Notification and Reporting Process

H.4.3 Estimated Occupational Radiation Exposure During the Continuing Care Period of Safe Storage

From the types and quantities of manpower given in Section H.4.1, and the radiation dose data given in Appendix C, the anticipated cumulative radiation exposures for surveillance, maintenance, and security activities for selected intervals during the continuing care period were computed. Cumulative radiation exposures for personnel for the first year of the continuing care period and for intervals of 10, 30, 50, and 100 years after final reactor shutdown are given in Table H.4-4. The majority of the occupational radiation exposure to personnel is accumulated during the first 30 years after shutdown of the reference PWR. Less than 4 man-rem total radiation exposure are accumulated during the next 70 years.

H.5 COSTS, OCCUPATIONAL RADIATION EXPOSURE, AND OTHER CONSIDERATIONS FOR DEFERRED DISMANTLEMENT

The same basic activities that are performed during Immediate Dismantlement are also performed during Deferred Dismantlement. The reactor vessel components, especially the most highly activated ones, are removed and sectioned semi-remotely while under water, using the same techniques as previously. The reactor vessel is sectioned in air, using semi-remote cutting equipment. Removal of the biological shield concrete that is still radioactive due to activated trace elements such as ^{152}Eu and ^{154}Eu may still be necessary, depending upon the length of the Safe Storage period.

It is assumed that the radioactive contamination of the piping systems, tanks, pools, etc. is primarily ^{60}Co . Thus, for Safe Storage periods of less than fifty years (~ 10 half-lives of ^{60}Co), the material remains radioactively contaminated to levels greater than those that would permit unrestricted use of the material. After 50 years of decay, it is assumed that the radioactive contamination on the bulk of the formerly contaminated material has decayed to levels that are indistinguishable from the natural radioactivity in the environment, and can be either salvaged for scrap value, buried in a land-fill, or left in the structures.

TABLE H.4-4. Estimated Cumulative Occupational Radiation Exposure During the Continuing Care Period of Safe Storage(a)

Type of Employee	Estimated Man Year Labor/Yr(b)	Estimated % Time Spent in Radiation Zone	Avg. Est. Radiation Zone, mR/hr(e)	Est. Man-Rem Exposure For the First Year of the Continuing Care Period
S&M Representative	0.25	<5	<5	<0.13 ^(f)
Surveillance and Maintenance Manpower:				
Equipment Maintenance	0.1	25	<10	<0.5
Third Party Inspections	0.06	40	<5	<0.24
Quality Assurance	0.02	25	<5	<0.05
Radiation and Environmental Surveillance	0.05 ^(c)	50	<10	<0.5
Administration and Clerical	0.25	-0-	-0-	-0-
Security	1.0 ^(d)	<1	<5	<0.44
FIRST YEAR TOTAL	1.73			<1.87

CUMULATIVE EXPOSURE TOTALS FOR:

- The first year of Continuing Care (CC) <1.87
- 10 years after final reactor shutdown (9 years of CC) <9.87
- 30 years after final reactor shutdown (29 years of CC) <13.91
- 50 years after final reactor shutdown (49 years of CC) <14.2
- 100 years after final reactor shutdown (99 years of CC) <14.22

(a) Cumulative exposure is calculated assuming ^{60}Co to be the dominant occupational dose contributor during the first 100 years of the continuing care period.

(b) Derived from Appendix H.4-1.

(c) Includes only the support personnel that are involved in the collection of data in radiation zones.

(d) Based on 8766 hrs/yr; i.e., offsite security force on 24-hr call and response (Option C of Table H.4-3.).

(e) The tabular values shown are estimated for the start of the continuing care period.

(f) Example calculation:

$$(\text{man yr}) \left\{ 250 \frac{\text{working days}}{\text{man yr}} \right\} \left\{ \frac{8 \text{ hrs}}{\text{working day}} \right\} \left\{ \frac{\% \text{ time estimated}}{\text{in the radiation zone}} \right\} \left\{ \frac{\text{Exposure}}{\text{in mR/hr}} \right\} = \text{Estimated Annual Man-Rem Exposure}$$

Activities such as reactor defueling and fuel shipment, chemical decontamination of the fluid systems, and removal of radioactive wastes such as filters, resins, and evaporator bottoms, were accomplished during preparations for Safe Storage. These activities are replaced by extensive training and familiarization of the work force with the facility, the restoration of essential services to areas within the facility where they are needed, and removal of the various plates, locks, welds, and barricades that were installed to secure the facility prior to the Safe Storage period. In view of these considerations, it is assumed that a work force of essentially the same size as was utilized in Immediate Dismantlement is needed for Deferred Dismantlement, and for approximately the same time duration.

H.5.1 Estimated Costs for Deferred Dismantlement

Based on the discussion given above and in Section 9.3, a convenient way to estimate the costs for Deferred Dismantlement after periods of Safe Storage of various lengths is to examine only those cost parameters that are different from Immediate Dismantlement. The manpower costs are assumed to be the same. The cost items that are significantly different include transport of spent fuel, supplies, electrical power, radioactive material disposal, and nuclear insurance.

Estimates of the volumes of the various types of radioactive materials that must be packaged and shipped for burial when dismantlement is performed starting immediately and starting 10, 30, 50 and 100 years after reactor shutdown are given in Table H.5-1. The volume of radioactive material from preparations for Safe Storage is also given. These volumes are summarized from Tables G.4-3, -4, -5 and -6 in Appendix G and from Table H.3-2. The volume of radioactive waste estimated for dismantlement at 10 and 30 years is the difference between the volumes for Immediate Dismantlement and for preparations for Safe Storage. This quantity is assumed to be reduced to 150 m^3 at 50 years and to 100 m^3 at 100 years based on engineering judgment, and consists primarily of combustible wastes.

The volume of contaminated material is assumed to remain constant through 30 years but to have decreased to 100 m^3 by 50 years and thereafter based on engineering judgment.

TABLE H.5-1. Volumes of Radioactive Material Packaged and Buried During the Various Decommissioning Options

Decommissioning Option	Option Starts: (Years After Shutdown)	Burial Volume (m ³)			Option Total (Rounded)	Decommissioning Total (Rounded)
		Radioactive Waste	Contaminated Material	Activated Material		
Immediate Dismantlement	0	618	16,078	1,191	17,900	17,900
Preparations for Safe Storage	0	388	--	--	390	--
Deferred Dismantlement	10	230	16,078	1,191	17,500	17,900
	30	230	16,078	1,191	17,500	17,900
	50	150	100	1,191	1,440	1,830
	100	100	100	1,191	1,390	1,780

The volume of activated material is assumed to remain constant over the 100 year time span. Depending upon the particular radionuclides present in the concrete biological shield, the activated concrete may have decayed to unrestricted use levels in 100 years, but that possibility is not assumed in this study.

The costs of Immediate Dismantlement, as given in Section 10.1, are broken down into expenditures by year in Table H.5-2. The 25% contingency has been incorporated into the tabular values. Using the volumes of radioactive material listed in Table H.5-1, the costs for disposal of those materials during dismantlement that has been deferred for different time periods are also given in Table H.5-2, in terms of a cost reduction. For example, the volume of radioactive wastes for disposal during dismantlement at 10 or 30 years is 388 m³ less than for Immediate Dismantlement, since that quantity of radioactive wastes (filters, resins, etc.) was removed during the preparations for Safe Storage. Therefore, the expenditures listed for waste disposal for the deferred case are reduced appropriately and the amount of the reduction is listed in the table under the deferred case. Similarly, shipment of spent fuel does not take place in the deferred cases, thus a reduction of \$3.084M is shown for the deferred cases. On the other hand, there are the additional costs of continuing care during the two years of planning and preparation that must be added. The different values listed for the Deferred Dismantlement cases are summed algebraically with the yearly total for Immediate Dismantlement to obtain the yearly totals for the deferred cases, as given in the lower portions of Table H.5-2.

TABLE H.5-2. Time Distribution of Dismantlement Expenditures^(a)

Cost Item	Costs in Millions of Dollars ^(b)						Total
	Year of Expenditure Relative to Start of Dismantlement						
	-2	-1	1	2	3	4	
<u>Immediate Dismantlement</u>							
Fuel Shipment			1 652	1 432			3 084
Equipment	0 206	0 822					1 028
Supplies		0 975	0 487	0 325	0 162		1 949
Power			1 806	1 755	0 796	0 018	4 375
Activated Materials			1 762	1 656			3 418
Contaminated Material			2 203	2 592	1 684		6 479
Radioactive Waste			0 541	0 181	0 144		0 866
Staff Labor	0 332	0 896	3 682	3 447	2 372	0 504	11 233
Contractor Services			0 170	0 170	0 170	0 170	0 680
Demolition Services				2 811	3 514	1 687	8 012
Nuclear Insurance			0 375	0 250	0 250	0 125	1 000
TOTAL	0 538	2 693	12 678	14 619	9 092	2 504	42 124
<u>Dismantlement Deferred for 10 or 30 Years</u>							
Continuing Care	+0 080	+0 080					+0 160
Fuel Shipment			-1 652	-1 432			-3 084
Equipment			-0 075				-0 075
Supplies		-0 669	-0 149	-0 074			-0 892
Power			-0 258				-0 258
Radioactive Waste			-0 339	-0 114			-0 453
Nuclear Insurance			-0 250	-0 125	-0 125		-0 500
TOTAL	0 618	2 104	9 955	12 874	8 967	2 504	37 022
<u>Dismantlement Deferred 50 Years</u>							
Continuing Care	+0 080	+0 080					+0 160
Fuel Shipment			-1 652	-1 432			-3 084
Equipment			-0 075				-0 075
Supplies		-0 669	-0 149	-0 074			-0 892
Power			-0 258				-0 258
Radioactive Waste			-0 410	-0 137			-0 547
Contaminated Material			-2 189	-2 576	-1 674		-6 439
Nuclear Insurance			-0 250	-0 125	-0 125		-0 500
TOTAL	0 618	2 104	7 695	10 275	7 293	2 504	30 489
<u>Dismantlement Deferred 100 Years</u>							
Continuing Care	+0 080	+0 080					+0 160
Fuel Shipment			-1 652	-1 432			-3 084
Equipment			-0 075				-0 075
Supplies		-0 669	-0 149	-0 074			-0 892
Power			-0 258				-0 258
Radioactive Waste			-0 453	-0 152			-0 605
Contaminated Material			-2 189	-2 576	-1 674		-6 439
Nuclear Insurance			-0 250	-0 125	-0 125		-0 500
TOTAL	0 618	2 104	7 652	10 260	7 293	2 504	30 431

(a) The values shown for the Deferred Dismantlement cases are the difference in that particular cost item for that year compared to Immediate Dismantlement. The totals for the deferred cases are the algebraic sum of the Immediate Dismantlement total and the differences for that year.

(b) Values include a 25% contingency.

A similar display of the costs of preparation for Safe Storage, broken down by year of expenditure, is given in Table H.5-3, based on Table 10.2-1 in Section 10. The tabular values include a 25% contingency.

TABLE H.5-3. Time Distribution of Expenditures
During Preparations for Safe Storage

Costs in Millions of Dollars ^(a)					
Year of Expenditure Relative to Reactor Shutdown					Continuing Care/Year
-2	-1	1	2	Total	
0.205	1.990	8.821	1.602	12.618	0.080

^(a) includes 25% contingency

The total cost (in constant 1978 dollars) for each of the decommissioning options is shown in Table H.5-4. The values listed under Option Total include the cost of continuing care during Safe Storage, for the number of years shown. The option total given for deferred dismantlement starting 30 years after reactor shutdown includes the cost of continuing care from the end of the preparations for Safe Storage until the start of dismantlement, or about 28 1/2 years, plus the dismantlement costs. The decommissioning total is the sum of the costs for preparations for Safe Storage, the cost of continuing care, and the cost of dismantlement.

TABLE H.5-4. Total Costs for Various Decommissioning Choices

Years After Reactor Shutdown	Decommissioning Costs in Millions of Dollars (Rounded)				Total
	Immediate Dismantlement	Preparations for Safe Storage	Continuing Care	Deferred Dismantlement	
0	42.1				42.1
10		12.6	0.6	37.0	50.2
30		12.6	2.2	37.0	51.8
50		12.6	3.7	30.5 ^(a)	46.8
100		12.6	7.8	30.4 ^(a)	50.8

^(a) Reduced dismantlement costs result from smaller volumes of contaminated material to be packaged, shipped and buried, due to decay of the contaminating radionuclides.

H.5.2 Partial Dismantlement with Safe Storage

A variation of Safe Storage that is of potential interest is one in which the noncontaminated structures outside of the exclusion area (shown in Figure 9.2-1) are demolished during the preparations for Safe Storage. By removing the structures and removing any contaminated soil or paving material that might be present, the radiation levels on the outer portion of the site can be reduced to levels that are sufficiently low to permit unrestricted use of the property. It is anticipated that the removal of a few centimeters of surface soil from a few square meters of ground surface and removal of sections of paving material to be packaged and shipped to a burial site would be essentially all that is needed to obtain the release of the outer area of the reactor site. Thus, about 4.6 km^2 could be released for other uses while retaining the 0.1 km^2 Exclusion Area in Safe Storage.

Comparison of the calculated site surface contamination levels, given in Tables 7.3-13 and 7.3-14 with the levels of radioactivity that would produce an annual dose to the maximally exposed individual member of the public of 1 and 25 mrem, given in Tables 8.2-5 and 8.2-6, shows that the site contamination levels anticipated to be present after 40 years of reactor operation are less than those that produce a 1 mrem annual radiation dose. Therefore, except for isolated "hot spots" that resulted from operational accidents or spills, the outer area of the site could be released for unrestricted use.

Data on the levels of radioactive contamination present on the surfaces of the outer areas of the site are extremely sketchy and have not been collected in a constant and systematic manner. A program to collect such data from many reactor sites in a consistent manner would be very useful in helping to evaluate the validity of the calculations of surface depositions, as presented in Section 7.3.2. An extensive soil surface radiation survey and analysis of numerous soil samples using sensitive gamma ray analysis equipment would be needed to assure that the radiation limits as defined by the methodology given in Section 8 are not exceeded.

The costs of the surveys and analyses, and of the removal and disposal of any identified material that exceeded releasable radiation levels are

expected to be small, probably less than \$50,000, and are insignificant in comparison with the \$13 million costs of the rest of the preparations for Safe Storage.

The major difference in costs would be the cost of demolition of the structures in the outer area, estimated to be about \$2.5 million. This cost would be added to the cost of preparations for Safe Storage, but would be subtracted from the demolition costs for Deferred Dismantlement, so that in terms of constant dollars there would be no net change in the total cost of the decommissioning program.

The activities needed to obtain the release of the outer area of the site would have no impact on the occupational radiation exposure accumulated during the decommissioning program.

H.6 EXTERNAL OCCUPATIONAL RADIATION EXPOSURE DURING DEFERRED DISMANTLEMENT

For this study it is assumed that the amounts of occupational radiation exposure accumulated by the decommissioning workers is controlled largely by the radiation levels of ^{60}Co throughout the plant. Thus, if a given task performed immediately after shutdown caused a radiation exposure of N_0 , that same task performed t years later during Deferred Dismantlement would cause an exposure of $N(t) = N_0 e^{-\lambda t}$, where λ is the decay constant for ^{60}Co in years.

Since one of the key assumptions made for Deferred Dismantlement is that essentially all of the same jobs would be performed in approximately the same way as for Immediate Dismantlement, using the same techniques and equipment, the occupational radiation exposure accumulated during Deferred Dismantlement would be proportional to that which was accumulated during Immediate Dismantlement, reduced by the relative reduction of the radioactivity levels of ^{60}Co over the Safe Storage period. Several of the job items given in Table 11.3-1 for the radiation exposure accumulated during Immediate Dismantlement such as reactor defueling and chemical decontamination should be subtracted from the total exposure since they are not performed during Deferred Dismantlement. Several other items such as unsecuring the facility and reestablishing essential services should be added. However, for purposes of estimating the occupational

exposure for Deferred Dismantlement, a simple reduction of the Immediate Dismantlement exposure in proportion to the decay of ^{60}Co over the Safe Storage period is a reasonable and conservative approach. These estimates are given in Table H.6-1 for dismantlement starting 10, 30, 50 and 100 years after reactor shutdown. After 100 years, essentially all of the remaining radioactivity is contained in the activated reactor vessel components, and the occupational radiation exposure associated with dismantlement is extremely small.

TABLE H.6-1. Estimated External Occupational Radiation Exposure During Deferred Dismantlement^(a)

<u>Decommissioning Mode</u>	<u>Time After Final Reactor Shutdown</u>	<u>Estimated Dose, Man-Rem</u>
Immediate Dismantlement	0	1224
Deferred Dismantlement	10	329
	30	24
	50	2
	100	1

^(a) Man-rem estimates derived from Table 11.3-1 in Section 11.

REFERENCES

1. TROJAN Nuclear Plant Environmental Report, Volume I, Section 6.3, Portland General Electric Company, November 1971.
2. Personal communication: from Mr. A. W. Pound, Chief, Moon Security Service, Inc., Pasco, WA 99301, April 11, 1977.

APPENDIX I

COMPILATION OF UNIT COST FACTORS

To make it easy to apply the information developed during this study to other decommissioning situations, the base data used in making the estimates of costs for the various portions of decommissioning are presented in this appendix. Categories included are: salaries, packaging, cask rental, transport, disposal, special equipment and services. All data are late 1977, early 1978 costs.

I.1 MANPOWER COSTS

Salary data for typical decommissioning staff positions are given in Table I.1-1. The original 1975 data base has been adjusted by a factor of 1.29 to account for escalation between 1975 and 1978, based on the Handy-Whitman Index.⁽¹⁾

I.2 COSTS OF PACKAGING RADIOACTIVE MATERIALS FOR DISPOSAL

The costs of packaging radioactive materials for disposal in a shallow land burial site are composed of the container cost and, in the cases of liquids or ion exchange resins, the cost of a solidifying agent.

I.2.1 Shipping Container Costs

The cost of containers for packaging radioactive materials for disposal is a major cost item for decommissioning. In the case of immediate dismantlement, container costs are nearly 10% of the total cost of decommissioning. The containers assumed to be used in this study are listed in Table I.2-1, together with brief descriptions, particular usage, and unit costs.

⁽¹⁾ Handy-Whitman Index of Public Utility Construction Costs Bulletin No. 105, Section B-6, as amended 1-1-78.

TABLE I.1-1. Salary Data for Typical Staff Positions

<u>Source</u>	<u>Position</u>	<u>Base Pay/Yr(\$)</u>	<u>Overhead Rate (%)</u>	<u>Cost/Yr(\$)</u>
a	Superintendent	43,800	70	74,500
a	Engineer	37,400	70	63,500
a	Asst. Engineer	25,800	70	43,800
b	Health & Safety Supervisor	29,600	70	50,400
b	Industrial Safety Specialist	25,800	70	43,800
b	Health Physicist	23,200	70	39,400
b	Sr. Health Physics Technician	19,300	50	29,000
d	Health Physics Technician	16,800	50	25,100
b	Protective Equipment Attendant	15,500	50	23,200
b	Quality Assurance Supervisor	25,800	70	43,800
b	Quality Assurance Engineer	23,200	70	39,400
b	Quality Assurance Technician	15,500	50	23,200
a	Contracts & Accounting Specialist	22,600	70	38,400
b	Accounting Clerk	13,500	50	20,300
b	Security Supervisor	19,300	70	32,900
c	Security Patrolman	14,200	50	21,300
b	Radioactive Shipment Specialist	19,300	70	32,900
b	Procurement Specialist	18,000	70	30,700
b	Craft Supervisor	23,200	70	39,400
d	Craftsman	18,000	50	27,100
d	Tool Crib Attendant	15,500	50	23,200
b	Shift Engineer	23,200	70	39,400
d	Crew Leader	21,900	50	32,900
d	Utility Operator	18,000	50	27,100
c	Laborer	17,300	50	25,900
b	Secretary	13,500	50	20,300
b	Consultant	67,000	--	67,000

- a. U.S. Dept. of Labor, Bureau of Labor Statistics, Bulletin, March 1975.
b. Author's estimate.
c. R. S. Means Co., Building Construction Cost Data - 1975.
d. Hanford Atomic Metal Trades Council Pay Scales.

NOTE: Base salary data has been adjusted to January 1978 using the Handy-Whitman Index of Public Utility Construction Costs, Pacific Region.

TABLE I.2-1. Shipping Containers Used in This Study

<u>Description</u>	<u>Burial₃ Volume(m³)</u>	<u>Proposed Usage</u>	<u>Estimated Unit Cost(\$)</u>
Shielded Cask Liner 1.38 m O.D. x 1.9 m high	2.84	Shallow Land Burial of Activated Reactor Components	10,000
Stainless Steel Canister 0.76 m O.D. x 3.05 m high	1.38	Deep Geologic Disposal of Activated Reactor Components	5,000
Steel Cask Liner 1.38 O.D. x 1.9 high	2.84	Solidified Resins and Evaporator Bottoms	550
Concrete-lined Steel 0.21 m ³ (55-gallon) drum	0.21	Filter Cartridges	100
Standard Steel 0.21 m ³ (55-gallon) drum	0.21	Compacted Combustible Dry Waste	20
Fiberglassed Wooden Box 1.2 m x 1.2 m x 2.13 m	3.07	Low Specific Activity Material: Piping, Concrete, Contaminated Components	400
Steel Box Assorted Sizes	~3	Low Specific Activity Material: Piping, Concrete, Contaminated Components	6,000

I.2.2 Solidifying Agent Costs

For this study, it was assumed that all evaporator bottoms liquids and ion exchange resins were mixed with a solidifying agent and poured into steel cask liners. The solidifying agent was assumed to be ureaformaldehyde, although portland cement- and bitumen- based solidification systems are also in use in the industry. The cost of the ureaformaldehyde per container (inner volume ~2.3 m³) was assumed to be \$375.

I.3 CASK CHARGES

Much of the material to be shipped to a burial site will be sufficiently radioactive to require transport in a shielded cask. In general, it will be more cost-effective to rent shielded casks than to purchase them, especially the larger casks. The casks postulated for use by this study are listed in Table I.3-1, together with physical descriptions and probable rental charges.

TABLE I.3-1. Shield Casks for Shipment of Radioactive Materials

Description	Empty Weight	Daily Rental	Usage
(1) 1.63 m O.D. x 2.34 m high 100 mm Pb. thickness	16,330 kg (36,000 lb)	\$100	Activated Components, Resins, Evaporator Bottoms
(2) IF-300 Spent Fuel Cask (General Electric Co.)	120,000 kg (265,000 lb)	\$3500 (30 day minimum)	Activated Components/Spent Fuel to Geologic Disposal/ to Repository
(3) 1.24 m O.D. X 1.56 m high 150 mm Pb. thickness	9,300 kg (20,500 lb)	\$100	Filter Cartridges
(4) 1.95 m O.D. x 1.04 m high 50 mm Pb. thickness	7,000 kg (15,410 lb)	\$100	(55-gallon) Drums 0.21 m ³

NOTE: For some shipments a protective overpack may be required. A typical overpack rental is \$125 per day.

I.4 TRANSPORTATION COSTS

Transport of radioactive materials to a burial site is assumed to be accomplished using exclusive-use trucks. The rate schedules, excess gross weight penalties, and charges for a second driver given in Tables 1.4-1, -2, and -3, which were reproduced from the published rates⁽²⁾ of a carrier licensed to transport radioactive materials, are used in this study.

I.4-1 Shipment by Exclusive-Use Trucks

The gross vehicle weight (GVW) for truck shipments is assumed to be less than 20.412 Tonne (45,000 lb) for normal shipments. For the 1600 km (1000 mi) roundtrip distance between the reactor site and the disposal site, the unit cost per shipment is: 1000 miles x 0.97 \$/mile = \$970.

To assure rapid turnaround on the shipping casks, two drivers are required, at an additional cost of \$0.15 per mile, for a total of \$150, as found from Table I.4-3.

The maximum allowed GVW varies from state to state, ranging from 30.21 Tonne (66,600 lb) to 45.36 Tonne (100,000 lb), with the majority of states

⁽²⁾MF-I.C.C. No. 195, Local Commodity Tariff No. 1045-E, Tri-State Motor Transit Co., Joplin, MO, May 1977.

TABLE I.4-1. Mileage Rates for Legal Size and Weight Shipments (effective April 1, 1978)

Item No. 3000-D
Cancels 3000-C (A)

SECTION II - MILEAGE COMMODITY RATES (Con't.)

MILEAGE COMMODITY RATES

COMMODITY: Radioactive Waste and Sensitive Non-Radioactive Waste for Burial
Purposes Only and Empty Containers therefor moving to or From
points of loading, unloading or storage

BETWEEN: All points in the United States to the extent authorized as
published in Scope of Operations published herein.

RATES IN CENTS PER MILES

One-Way Mileage (Not Over)	Column 1	Column 2	One-Way Mileage (Not Over)	Column 1	Column 2
100	319	229	425	147	109
125	293	212	450	141	107
150	269	196	475	137	104
175	244	182	500	132	103
200	212	166	550	128	100
225	201	158	600	125	97
250	192	147	650	122	97
275	184	138	700	119	97
300	175	132	750	117	97
325	171	124	800	112	97
350	165	120	850	111	97
375	159	116	900	109	97
400	152	112	950	108	97
			1000 & Beyond	105	97

NOTES:

(1) Rates apply on legal size and weight shipments only. Overweight shipments not exceeding a gross vehicle weight of 85,000 pounds shall be subject to an additional charge of \$0.21 per mile in addition to all other applicable charges. For rates on shipments exceeding 85,000 pounds gross vehicle weight, see Section II, Item 2000 Application.

(2) Column 1 rates apply to one-way moves.

(3) Subject to restriction, Column 2 rates apply only to continuous excursion moves on one shipper issued bill-of-lading in which a subsequent shipment is made available to carrier within 24 hours after arrival at point of loading or unloading. Only one stop in transit allowed under Column 2 rates. RESTRICTION: Column 2 rates will not apply in connection with shipments moving under Item 970 Application.

(4) Minimum charge per trip to be computed on basis of 100 one-way miles.

(5) Rates named in this item not subject to Paragraph (3), Item 1240 herein.

(6) One way (Column 1) moves between points in the states of Arizona, California, Idaho, Oregon, and Utah on the one hand, and Beatty, Nevada on the other shall be subject to an arbitrary charge of \$0.25 per mile, in addition to all other applicable charges.

(7) One Way (Column 1) moves between points in the states of Idaho and Oregon, on the one hand, and Hanford Works, Washington, on the other, shall be subject to an arbitrary charge of \$0.25 per mile, in addition to all other applicable charges.

(8) Shipments made on shipper furnished trailers will be subject to an allowance of \$0.03 per mile. Such allowance will be based on the mileage used in computing the freight charges and will be made by deducting the amount of the allowance from the freight charges.

For Explanation of Abbreviations and Reference Marks, See Item 820.

TABLE I.4-2. Additional Changes when Gross Shipment Weight Exceeds 45,000 Pounds (effective May 1, 1977)

Item No. 1020

RULES AND REGULATIONS GOVERNING TARIFF (Con't.)

ADDITIONAL CHARGES TO BE ASSESSED BY CARRIER WHEN OVERWEIGHT PERMITS ARE REQUIRED

Any non-divisible shipment that requires overweight permits will be assessed, in addition to all other applicable charges, an overweight charge for each state transported from, to, or through as shown in this item.

Shipments from, to or through the following States shall be assessed a flat charge as indicated per one-way trip. (Charges in dollars and cents).

STATE	CHARGE	STATE	CHARGE	STATE	CHARGE
Alabama	(A) \$10.00	Massachusetts	(A) \$10.00	Ohio	(A) \$12.50
Arizona	10.00	Michigan	(A) 15.00	Oregon	5.00
California	(A) 15.00	Minnesota	(A) 15.00	Rhode Island	(A) 10.00
Colorado	(A) 16.00	Missouri	(A) 18.00	South Carolina	(A) 15.00
Connecticut	(A) 15.00	Nebraska	15.00	South Dakota	10.00
Georgia	(A) 15.00	Nevada	(A) 10.00	Tennessee	(A) 10.00
Iowa	(A) 15.00			Texas	10.00
Idaho	30.00	New Mexico	(A) 10.00	Utah	8.00
Kansas	7.00	New York	(A) 15.00	Vermont	(A) 15.00
Kentucky	(A) 25.00	North Carolina	(A) 10.00	Wisconsin	(A) 15.00
Maine	15.00	North Dakota	(A) 15.00		

Fees from, to, or through States not listed above shall be as follows:

ARKANSAS

A charge per hundred pounds over 45,000 pounds as follows:

0 to 100 miles	\$ 0.25	201 to 270 miles	\$ 0.30
101 to 150 miles	0.30	271 and over miles	0.45
151 to 200 miles	\$ 0.35		

DELAWARE

A charge of \$10.00 plus \$0.02 per ton mile on the excess weight over 45,000 pounds.

FLORIDA

Shipments weighing 45,001 to 71,720 pounds, a flat charge of \$15.00 per load.

Shipments weighing over 71,720 pounds, a flat charge of \$25.00 per load.

ILLINOIS

A charge of \$15.00 plus \$0.20 per mile traveled in Illinois.

INDIANA

A charge of \$5.00 plus \$0.25 per mile travelled in Indiana.

LOUISIANA

Shipments weighing 45,001 to 53,000 pounds a flat charge of \$75.00 per load.

Shipments weighing 53,001 to 57,000 pounds a flat charge of \$100.00 per load.

Shipments weighing 57,001 to 61,000 pounds a flat charge of \$150.00 per load.

MARYLAND

A charge of \$1.00 per ton or fraction thereof on weight in excess of 45,000 pounds.

MISSISSIPPI

A charge of \$0.11 per ton mile on weight in excess of 45,000 pounds. (Cont.)

For Explanation of Abbreviations and Reference Marks, See Item 820.

TABLE I.4-2. (continued)

Item No. 1020 (Cont.)	RULES AND REGULATIONS GOVERNING TARIFF (Cont.)
	<p align="center">ADDITIONAL CHARGES TO BE ASSESSED BY CARRIER WHEN GROSS WEIGHT OF SHIPMENT IS IN EXCESS OF 45,000 POUNDS (Continued)</p> <p>MONTANA On shipments weighing in excess of 45,000 pounds per shipment charges shall be as follows: Distance of 100 miles or less (In Montana) \$13.00 per shipment. Distance of 101 to 200 miles (In Montana) \$23.00 per shipment. Distance of 201 miles or over (In Montana) \$33.00 per shipment.</p> <p>NEW HAMPSHIRE On shipments weighing 45,001 pounds to 65,000 pounds a flat charge of \$15.00 per load. On shipments weighing 65,001 pounds and over a flat charge of \$15.00 plus \$0.02 per hundred pounds on the weight in excess over 45,000 pounds per load.</p> <p>(A) NEW JERSEY A flat charge of \$10.00 plus \$5.00 for each 2000 pounds or fraction thereof in excess of 45,000 pounds.</p> <p>OKLAHOMA A flat charge of \$10.00 plus \$5.00 for each 1,000 pounds or fraction thereof on the weight in excess of 45,000 pounds per load.</p> <p>PENNSYLVANIA (A) A flat charge of \$10.00 plus \$0.03 per ton mile on the weight in excess of 45,000 pounds per load.</p> <p>VIRGINIA A charge of \$0.10 per mile travelled in Virginia.</p> <p>WASHINGTON A flat charge of \$5.00 plus the following 45,001 to 51,000 pounds a charge of \$0.075 per mile 51,001 to 57,000 pounds a charge of 0.150 per mile 57,001 to 63,000 pounds a charge of 0.225 per mile 63,001 to 69,000 pounds a charge of 0.375 per mile 69,001 to 75,000 pounds a charge of 0.525 per mile 75,001 to 81,000 pounds a charge of \$0.675 per mile. 81,001 pounds and over a charge of \$0.825 per mile.</p> <p>WEST VIRGINIA (R) A flat charge of \$5.00 plus \$0.03 per ton mile over 50,000 pounds per load.</p> <p>WYOMING A flat charge of \$10.00 plus \$0.02 per ton mile. Maximum charge \$20.00</p> <p>NOTE: (1) Per Ton Mile means 2,000 pounds of freight transported one (1) mile. (2) Per Ton means a weight of 2,000 pounds.</p> <p align="center">For Explanation of Abbreviations and Reference Marks, See Item 820.</p>

TABLE I.4-3. Rates for a Second Driver per Shipment (effective April 1, 1976)

Item No.	RULES AND REGULATIONS GOVERNING TARIFF (Con't.)
980	<p style="text-align: center;">SECOND DRIVER</p> <p><u>Rates and charges published in this tariff are predicated on carrier-furnished equipment and one driver. When shipper requests an additional driver on the transport vehicle or specifies a delivery schedule that requires two drivers to accomplish the delivery within the specified time, a charge of fifteen cents (\$0.15) per mile will be assessed, subject to a \$ 60.00 minimum per shipment, in addition to all other applicable charges.</u></p> <p>Request for second driver must be inserted in the accompanying bill of lading by the shipper at time of shipment.</p>

having limits in the vicinity of 33.11 Tonne (73,000 lb). Overweight charges by the states vary widely as shown in Table I.4-2. Using the maximum rates in Table I.3-2 (Washington), the state overweight penalty is \$530 for a shipment with a GVW of 33.11 Tonne (73,000 lb). The carrier imposes a surcharge of \$0.21 per mile for shipments with GVW between 20.41 Tonne (45,000 lb) and 38.56 Tonne (85,000 lb), for a cost of \$210. Shipments in excess of 33.11 Tonne (73,000 lb) GVW require special equipment and special permission. Costs would have to be determined on a case-by-case basis.

The GVW of an unloaded truck is assumed to be 12.70 Tonne (28,000 lb), for the purposes of this study. Therefore, the payload per shipment is 7.71 Tonne (17,000 lb), legal, and 20.41 Tonne (45,000 lb), overweight.

The base conditions assumed for this study are summarized in Table I.4-4.

TABLE I.4-4. Basic Shipping Cost per Shipment

Status	Payload		GVW		Cost(\$) ^(a)
	(lb)	(Tonne)	(lb)	(Tonne)	
Legal	17,000	7.711	45,000	20.412	1120
Overweight	45,000	20.412	73,000	33.112	1860

^(a) 1600 km (1000 mile) round-trip distance between reactor site and burial site, two drivers.

I.4-2. Shipment by Rail

For the alternative case where the highly activated reactor components are placed in a deep geologic repository, shipment by railcar is assumed. Based on data given in a study on the shipment of spent reactor fuels,⁽³⁾ the cost for a round-trip shipping distance of 4828 km (3000 miles) is about \$110 per Tonne (\$5 per hundred pounds), or about \$25,100 for a car carrying the GE IF-300 Cask. Should a special train be required, a surcharge of \$12 per km (\$19 per mile) or about \$57,000, is added.

I.5 DISPOSAL COSTS

The basic assumption in this study is that all of the radioactive material removed from the reference PWR during decommissioning is placed in a shallow land burial disposal facility. The alternative of placing the highly activated reactor components in a deep geologic disposal facility is also considered. The unit costs for these two possibilities are given in this section.

I.5.1 Costs for Shallow Land Burial

The disposal costs used in this study are based on the September 1, 1977 price list from Nuclear Engineering Co., Inc., which operates burial sites at Richland, Washington, Beatty, Nevada, and Sheffield, Illinois. The prices in this study are for the Beatty site, and are given in Table I.5-1.

I.5.2 Costs for Deep Geologic Disposal of Highly Activated Reactor Components

Since a facility for deep geologic disposal does not presently exist, the costs presented in this section are somewhat speculative. Based on the analysis and discussion presented in Reference 4, a cost of \$7100/m³ is assumed

(3) W. V. Loscutoff, et al., "A Safety and Economic Study of Special Trains for Shipment of Spent Fuel," BNWL-2263, p. A-2, Battelle, Pacific Northwest Laboratory, Richland, WA, December 1977.

(4) K. J. Schneider and C. E. Jenkins, Study Coordinators, Technology, Safety and Costs of Decommissioning a Reference Nuclear Fuel Reprocessing Plant, Report of U.S. Nuclear Regulatory Commission by Battelle, Pacific Northwest Laboratory, NUREG-0278, Table 7.8-12, p.7-129, October 1977.

TABLE I.5-1. Basis for Burial Cost Estimates^(a)

I. DISPOSAL CHARGES, NON-TRU WASTE

A. Steel Drums, Wood or Fiber Boxes

<u>R/hr at Container Surface</u>	<u>Price/ft³</u>
0.00 - 0.20	\$ 2.65
0.20 - 1.00	3.00
1.01 - 2.00	3.50
2.01 - 5.00	4.55
5.01 - 10.00	5.60
10.01 - 20.00	7.60
20.01 - 40.00	10.05
40.01 - 60.00	15.50
60.01 - 80.00	19.10
80.01 - 100.00	21.00
Over - 100.00	by request

B. Disposable Liners

<u>R/hr at Container Surface</u>	<u>Surcharge Per Liner</u>	<u>Price/ft³</u>
0.00 - 0.20	NONE	\$2.65
0.20 - 1.00	\$ 50.00	2.65
1.01 - 2.00	125.00	2.65
2.01 - 5.00	175.00	2.65
5.01 - 10.00	250.00	2.65
10.01 - 20.00	325.00	2.65
20.01 - 40.00	400.00	2.65
40.01 - 60.00	475.00	2.65
60.01 - 80.00	550.00	2.65
80.01 - 100.00	625.00	2.65
Over - 100.00	by request	by request

TABLE I.5-1. Basis for Burial Cost Estimates (Contd)

- C. Minimum Disposal Charge \$100.00/shipment
- D. Radiation readings are maximum generated at surface of the burial container, irrespective of physical size or configuration.

II. SURCHARGE(S)

- A. Curie Surcharge (per load)
 - 1. Zero - 100 curies no charge
 - 2. 101 - 300 curies \$300.00
 - 3. 301 - License Limits \$300.00 plus \$0.05/Ci
- B. Man-rem exposure (Because of design or physical defect of container or shield) \$3.00/man/millirem
- C. Biological (on containers which are not sealed vapor tight) \$0.50/ft³
- D. Crane
 - 1. 0 - less than 10,000 lb no charge
 - 2. 10,000 to limit of site equipment \$50.00 plus 1¢/lb above 10,000 lb
 - 3. Special equipment by special quotation
- E. Special Slings.
 - Each use \$100.00
- F. Minimum charge for unloading a shielded container \$200.00

(a) Price List (9-1-77) Nuclear Engineering Co.,
San Ramon, Ca.

for the disposal of the highly activated reactor components. Thus, for the canisters presently considered for deep geologic disposal (1.39 m^3 burial volume), the disposal charge is \$9869/canister.

I.5.3 Non-Radioactive Materials

In the dismantlement mode there are significant quantities of non-radioactive materials to be removed from the site (concrete rubble, etc.) or brought to the site (earth for backfilling and surface grading). It is assumed that the non-radioactive concrete rubble is disposed of in a landfill within a reasonable distance of the reactor site. The cost of loading and handling this material is estimated to be \$13 to \$20 per m^3 (\$10 to \$15 per yd^3), depending on the amount of reinforcing steel that must be dealt with.

Earth loaded, hauled, placed and graded on the site for final restoration is estimated to cost \$6.54/ m^3 (\$5/ yd^3).

I.6 SPECIAL EQUIPMENT, SERVICES, SUPPLIES, AND DERIVED UNIT COSTS

The various types of special equipment needed for decommissioning activities are described in some detail in Appendix F.3. The estimated unit costs for these equipment items are summarized in Table I.6-1.

Electricity

The principal services cost item is electric power. A unit cost of \$0.015/kwh is assumed. A significant portion of the electricity is used for heat to evaporate water via the boric acid and the radioactive waste evaporators. At 100% efficiency, 2.73 kwh are needed to evaporate a gallon of water, starting from 20°C. Depending upon the unit cost of electricity locally and the actual efficiency of the boilers and evaporators, this value could be as large as \$40/ m^3 (15¢/gal). For this study, \$26/ m^3 , (10¢/gal) is assumed.

Decontamination Chemicals

The unit costs of the decontamination chemicals are estimated to be:

EDTA - \$4.50/lb (\$2.04/kg)	\$2.88/lb of mixture,
Oxalic Acid - \$2.43/lb (\$1.10/kg) =	1/3 by weight of each.
Citric Acid - \$1.70/lb (\$0.77/kg)	(\$1.30/kg of mixture)

Ion Exchange Resins

The unit cost of ion exchange resins is estimated to be $\$90/\text{ft}^3$ ($\$3180/\text{m}^3$).

Contaminated Water

A unit cost for total disposal of contaminated water can be derived, using the evaporation costs given above and the disposal costs for evaporator bottoms given in Table G.4-5, Appendix G. Assuming a total of ~ 2 million gallons ($757/\text{m}^3$) of borated or contaminated water is evaporated, the cost is $\$200,000$ for evaporation and $\$365,200$ for disposal, for a unit cost of $\sim \$75/\text{m}^3$ ($\sim 28\text{¢}/\text{gal}$), not including staff labor.

Combustible Wastes

The cost per drum for disposal of combustible wastes can be derived from the values given in Table G.4-5, Appendix G: $\$207,431/1350$ drums $\sim \$154$ per drum ($\$20.5/\text{ft}^3$, $\$733/\text{m}^3$). This value is derived assuming that $\sim$ one-third of the drums are shipped in shielded casks.

Contaminated Concrete

The cost per unit volume for disposal of contaminated concrete can be derived from the values given in Table G.4-4, Appendix G: $\$3,177,610/9562 \text{ m}^3$ $\sim \$332/\text{m}^3$ ($\$254/\text{yd}^3$, $\$9.40/\text{ft}^3$).

TABLE I.6-1. Special Tools and Equipment Costs

Item	Estimated Unit Cost (\$000)
Underwater manipulator	300
Underwater plasma cutting torch	20
Underwater oxyacetylene torch	5
Portable plasma cutting torch	20
Portable oxyacetylene cutting torch	1
Arc saw	100
Guillotine pipe saws	4
Closed circuits, high resolution television systems, underwater and in air	(plant equip.)
Shielded vehicle with manipulators exchangeable tools, scoop loader, etc.	50
Underwater lights and viewing windows	5
Submersible pumps with disposable filter cartridges	1.5
Assorted underwater tools such as impact wrenches, bolt cutters, grapples, etc.	25
High velocity water jets	25
Hydraulic concrete surface spalling device	5
Concrete drills, Electric/pneumatic hammers	0.5
Portable filtered ventilation enclosures	1.5
Supplied airbubble suit	0.05
Safety nets	25
Blasting mats	0.5
Shipping casks	0.1/day
Electropolishing decontamination system	100
Mobile radwaste processing unit	10/month
Primary piping jumper	100

APPENDIX J

SAFETY ASSESSMENT DETAILS

In any undertaking involving facilities and systems that are either contaminated with radioactive materials or, as in the case of reactors and components, are radioactive due to neutron bombardment, it is necessary to assure the safety of both the personnel working in the facilities and the general public in the surrounding area. The safety of workers at the site is divided into two areas of concern: radiation safety and industrial safety. Radiation safety measures are designed to minimize the exposure to the workers from ionizing radiation. A detailed examination of the industrial safety aspects of the planned operations and the associated efforts to minimize industrial hazards, such as electrocution, drowning, or asphyxiation, is beyond the scope of this study. The safety of the public is principally related to the potential hazards associated with the release of airborne radioactive materials from decommissioning operations.

The purpose of this appendix is to quantify the airborne radionuclide releases, both planned and accidental, used in estimating the radiation exposures presented here and discussed in Section 11 of this report. In all cases, the scenarios for the airborne release of radionuclides use realistically maximized parameters in estimating the quantity of material released. Airborne radionuclide releases for Immediate Dismantlement and preparations for Safe Storage are estimated and reported in this appendix. The factors used for making these estimates are general enough to cover a broad spectrum of conditions. An attempt is made to quantify the airborne releases using measured release factors under similar conditions or using information about known behavior under the conditions postulated. A detailed probabilistic analysis of the postulated accident scenarios is not within the scope of this study. The estimated radionuclide releases in each category are tabulated in order of decreasing magnitude.

The radionuclide inventories used in this analysis are described in Section 7 and are listed in this appendix for ease of referencing. In addition to this information, the following basic assumptions about decommissioning the reference PWR are made:

1. The integrity of the Reactor Containment Building is not breached until all radioactive material has been removed or adequately contained.
2. The facility has fulfilled existing criteria for the containment of radionuclides for all accidents and natural phenomena during its operational period.
3. The reactor is shut down a few months before major disassembly begins. The reactor will be defueled and disabled. This means making the reactor inoperable, as specified in Regulatory Guide 1.86 by taking the control rod system out of service.⁽¹⁾
4. All of the spent fuel is shipped from the PWR in batches during a 16-month period for Dismantlement, or an 8-month period for Safe Storage.
5. Unneeded hazardous chemicals and equipment are removed after the reference PWR is shut down. Decontamination agents such as phosphoric acid, ethylenediaminetetraacetic acid, oxalic acid, and citric acid are in the available plant. Unneeded ion exchange resin and resin beds are removed.
6. HEPA filters are tested and replaced as required before decommissioning begins.
7. The atmospheric releases from the reference PWR buildings are assumed to pass through a single HEPA filter with transmission factor of 5×10^{-4} .
8. In areas with high levels of radioactive contamination, a temporarily installed "greenhouse," or contamination control envelope, will be used. The contamination control envelope is assumed to have a HEPA filter with a transmission factor of 5×10^{-4} to reduce the airborne radionuclide concentrations in the PWR buildings from selected decommissioning operations.

⁽¹⁾U.S. NRC, Termination of Operating Licenses for Nuclear Reactors. Regulatory Guide 1.86, June 1974.

9. The leakage rate from the contamination control envelope is assumed to be a function of the decommissioning operation involved and the time at which the operation occurs. The leakage rate is assumed to vary between 0.1 and 10% of the airborne concentrations within the contamination control envelope.
10. Surface contamination levels at plant shutdown are used to estimate the airborne radionuclide releases, with no credit taken for radioactive decay.
11. The airborne concentration of dust or liquid droplets is assumed to be similar to those observed at the Elk River reactor of 10 mg/m^3 .⁽²⁾ For blasting operations, the airborne concentrations are assumed to be a factor of 10 higher, or 100 mg/m^3 .⁽³⁾

J.1 AIRBORNE RELEASE SCENARIOS FOR DISMANTLEMENT OF THE REFERENCE PWR

The first option considered in the safety evaluation of decommissioning the reference PWR is Immediate Dismantlement of the facility. This mode will result in removal of the buildings and equipment and restoration of the site so unrestricted use by the general public can be permitted.

During these operations airborne releases can arise from planned operations or from postulated accidents. The accidents considered include both operational and natural phenomena events. Releases from normal operations and accidents have been evaluated and estimates of these releases are given in Tables J.1-1 and J.1-2. An estimate of the frequency of occurrence for each accident is listed in Table J.1-2. The frequencies are given as being high (greater than 10^{-2} per year), medium (between 10^{-2} and 10^{-5} per year), and low (less than 10^{-5} per year) based on published values or engineering judgement or experience since a rigorous probabilistic risk assessment is beyond the scope of this study.

(2) Final Elk River Reactor Program Report. COO-651-93, Revised, p. F-9, United Power Association, Elk River, MD, November 1974.

(3) J. M. Selby, et al., Considerations in the Assessment of Consequences of Effluents from Mixed Oxide Fuel Fabrication Plants. BNWL-1697, Rev. 1, pp. 78-79, Battelle, Pacific Northwest Laboratories, Richland, WA, June 1975.

TABLE J.1-1. Anticipated Airborne Radioactive Releases During Routine Decommissioning Operations^(a)

Operation	Reference Radionuclide Inventory Number ^(b)	Immediate Dismantlement			Preparations for Safe Storage		
		Airborne Radioactivity Generation Rate	Total Airborne Radioactivity in Building during Entire Operation μCi	Estimated Atmospheric Radioactive Release, μCi	Airborne Radioactivity Generation Rate	Total Airborne Radioactivity in Building during Entire Operation μCi	Estimated Atmospheric Radioactive Release μCi
Segmenting of Nonactivated Stainless							
Reactor Coolant Pumps and Primary Piping	4	6.8 $\mu\text{Ci}/\text{min}$	1.5×10^3	75	— ^(c)		
Steam Generators	4	55 $\mu\text{Ci}/\text{min}$	2.5×10^3	0.4	—		
Segmenting of Activated Reactor Vessel and Internals							
Internals	1	1150 $\mu\text{Ci}/\text{min}$	3×10^4	0.5	—		
Vessel	2	20 $\mu\text{Ci}/\text{min}$	4.15×10^2	0.2	—		
Waste Handling of Bioshield Concrete	3	0.04 $\mu\text{Ci}/\text{min}$	3.5×10^2	1.75	—		
Surface Cleaning Operations							
Hand Held Lance							
HX Room (Auxiliary Bldg.)	5	$3.7 \times 10^{-1} \mu\text{Ci}/\text{min}$	7.8×10^3	3.9	—		
Boric Acid Evaporator Room (Fuel Building)	5	$1.9 \times 10^{-1} \mu\text{Ci}/\text{min}$	2.0×10^3	1.0	S ^(d)		
Primary System Reactor Cavity	5	—			$3.7 \times 10^2 \mu\text{Ci}/\text{min}$	1.8×10^4	1
Spent Fuel Pool	5	2.6 $\mu\text{Ci}/\text{min}$	3.4×10^2	0.2	S		
Steam Generator Area (Containment)	5	$7.5 \times 10^{-2} \mu\text{Ci}/\text{min}$	3×10^2	0.2	S		
Ion Exchanger Vault	5	$3.1 \times 10^2 \mu\text{Ci}/\text{min}$	1.0	5.0×10^{-4}	—		
Laundry Room (Control Bldg.)	5	$7.4 \times 10^{-4} \mu\text{Ci}/\text{min}$	4×10^{-4}	S			
Final Chemical Decontamination	4	$4 \times 10^{-1} \mu\text{Ci}/\text{min}$	10^2	7.8×10^{-1}	—		
Sweeping	5	—			$2.8 \times 10^{-2} \mu\text{Ci}/\text{m}^3$	75	4×10^{-2}
(Vacuuming Alternative)	5	—				1	(5×10^{-3})
Household Detergents	5	Insig ^(e)			—		
Aerosol Type Foam	5	Insig			—		
Acid Soaked Sponges	5	—			Insig		
In Situ Chemical Decontamination							
Spray Leak	5	136 $\mu\text{Ci}/\text{min}$	1.36×10^3	0.7	S		
Liquid Leak	5	0.45 $\mu\text{Ci}/\text{min}$	4.5	2.3×10^{-3}	S		
Removal of Bioshield							
Drilling	3	$7.5 \times 10^{-1} \mu\text{Ci}/\text{min}$ (to containment)	1.2×10^{-1}	6.0×10^{-3}	—		
Explosion	3	5.3 $\mu\text{Ci}/\text{explosion}$ (in cavity)	10.6	5.3×10^{-3}	—		
Radiation Survey							
HX Room (Auxiliary Bldg.)	5	$9 \times 10^{-2} \mu\text{Ci}/\text{m}^3$	3.79×10^2	0.2	S		
Boric Acid Evaporator Room (Fuel Bldg.)	5	$2.6 \times 10^{-2} \mu\text{Ci}/\text{m}^3$	92	0.05	S		
Steam Generator Area (Containment)	5	$1.8 \times 10^{-2} \mu\text{Ci}/\text{m}^3$	38	1.9×10^{-2}	S		
Laundry Room (Control Bldg.)	5	$1.8 \times 10^{-4} \mu\text{Ci}/\text{m}^3$	2×10^{-4}	1×10^{-1}	S		
Removal of Concrete							
Steam Generator Enclosure							
Explosives	5	$2.3 \times 10^{-1} \mu\text{Ci}/\text{explosion}$	2×10^{-2}	1×10^{-3}	—		
Drilling	5	$1 \times 10^{-4} \mu\text{Ci}/\text{min}$	5×10^{-3}	2×10^{-4}	Insig		
Pneumatic Jackhammer	5	Insig			S		
Rock Splitters	5	Insig			S		
Interfacility Transport	5	Insig			S		
Fixing Residual Contamination	5	—			Insig		
Equipment Deactivation and Isolation							
Surveillance and Maintenance (53 Years)	5	—			3 $\mu\text{Ci}/\text{inspection}$	621	3.8×10^{-1}
On Site Retrievable Waste Storage	5	—				2.9×10^4	15

(a) The first year dose and fifty year dose commitments calculated for the maximum exposed individual and the population are listed in Tables J.4-1 through J.4-4.

(b) These numbers refer to the radionuclide inventories shown in Tables J.3.1 through J.3.5.

(c) A dash indicates that the operation does not apply to this decommissioning mode.

(d) If the release is the same for the second mode it is marked with an S.

(e) Insignificant means a building release of less than $2 \times 10^{-3} \mu\text{Ci}$, and radiation doses are not calculated.

TABLE J.1-2. Postulated Accidental Airborne Radioactive Releases During Decommissioning^(a)

Incident	Reference Radionuclide Inventory Number ^(b)	Immediate Dismantlement		Preparations for Safe Storage		Estimated Frequency of Occurrence ^(f)
		Airborne Radioactive Release in Building (μCi)	Estimated Radioactive Release to the Atmosphere (μCi)	Airborne Radioactive Release in Building (μCi)	Estimated Radioactive Release to the Atmosphere (μCi)	
Accident during Offsite Transportation	5		3.20×10^4	S ^(c)		—
Explosion of LPG Leaked from Front End Loader	5		3.6×10^3	— ^(d)		Low
Explosion of Onyascetylene during Segmenting of Vessel Shell	2	7.2×10^3	3.6×10^3	—		Medium
Explosion and/or Fire of Ion Exchange Resin	5	7.6×10^4	3.8×10^3	—		Medium
Gross Leak during In Situ Decontamination						
Spray Leak	5	4.1×10^4	2.07×10^3	S		Medium
Liquid Leak	5	1.38×10^3	7×10^{-1}	S		Medium
Segmentation of RCS Piping with Unremoved Contamination	4	2.1×10^4	1.05×10^3	—		High
Loss of Contamination Control Envelope during Onyascetylene Cutting of Vessel Shell	2	4.7×10^3	2.30	—		Medium
Vacuum Bag Rupture	5	—		2×10^3	1	Medium
Pressure Surge Damage to Filters during Bleeding of Activated Concrete Bioshield	3	504	0.30			Low
Accidental Cutting of Contaminated Piping	4	—		3.65×10^3	1.78×10^{-1}	High
Accidental Spraying of Concentrated Contamination with High Pressure Spray	5	—		2.44×10^3	1.22×10^{-1}	High
Accidental Break of Contaminated Piping during Inspection	4	—		2.2×10^3	<0.11	Low
Loss of Integrity of Portable Filtered Ventilation Enclosure	5	60	3×10^{-1}	—		Medium
Fire Involving Contaminated Clothing or Combustible Waste	5	12	6.0×10^{-3}	S		Medium
Loss of Bleeding Mat during Removal of Activated Concrete	3	5.3	2.7×10^{-3}	—		Medium
Detonation of Unused Explosives in Reactor Cavity	3	5.3	2.7×10^{-3}	—		Medium
Fire in Contaminated Sweeping Compound	5	—		0.15	7.5×10^{-3}	Medium
Temporary Loss of Local Airborne Contamination Control						
During Bleeding	3	2.7×10^{-3}	1.35×10^{-4}	—		Low
During Scarfing of Contaminated Concrete Surfaces with Jackhammer	5	Insig ^(e)		S		—
Temporary Loss of Services	3, 4, or 5	Insig		S		—
Dropping of Concrete Rubble	5	Insig		—		—
Natural Phenomena	5	Insig		S		—
Aircraft Crashes	5	Insig		S		—
Drop of Concrete Slab during Placement	5	—		Insig		—

(a)The first year dose and fifty year dose commitments calculated for the maximum exposed individual and the population are listed in Tables J.4-5 and J.4-6

(b)These numbers refer to the radionuclide inventories shown in Tables J.3-1 through J.3-5

(c)If the release is the same for the second mode it is marked with an S

(d)A dash indicates that the accident does not apply to this decommissioning mode

(e)Insignificant means a building release of less than 2×10^{-3} μCi , and radiation doses are not calculated.

(f)Frequency of Occurrence: High $>1 \times 10^{-2}$; Medium 1×10^{-3} to 1×10^{-2} ; Low $<1 \times 10^{-3}$ per year. A dash in this column means that no estimate was made for the specific incident listed.

J.1.1 Postulated Airborne Releases From Routine Dismantlement Operations

The dismantlement of a PWR includes operations such as cutting piping and equipment, drilling concrete, and spraying water or chemical solutions to decontaminate surfaces. These operations can generate considerable amounts of airborne radioactivity. The airborne materials are primarily in the form of solid particles and/or liquid droplets. Airborne contamination control measures are assumed to be used where they are deemed necessary. The use of other appropriate controls is considered in making these release estimates. The operations proposed during Immediate Dismantlement and the potential release situations presented by each operation are tabulated in Table J.1-3. The inventory of anticipated surface contamination in the PWR buildings is shown in Table J.1-4. The areas with the highest radioactive inventory where various decommissioning operations are performed are shown in Table J.1-5.

In the following sections, the Immediate Dismantlement airborne releases, which are listed in Table J.1-1 are described. The airborne releases are estimated by determining the maximum airborne release for each operation, and using this maximum value whenever the operation is performed, even for areas of lower contamination levels.

J.1.1.1 Final Chemical Decontamination

A station installed in the space formerly occupied by the new fuel storage area, is designed to decontaminate nonactivated stainless steel by electro-polishing. One tank, with dimensions of 1.2 m x 1.8 m x 2.4 m (4 ft x 6 ft x 8 ft), contains electrolyte (75 wt.% H_3PO_4), and a second tank holds rinse water. Also, there is a tank for final hose rinsing. A current of 10,000 A is applied to remove 3 to 75 micrometers from all surfaces of a piece being decontaminated. This operation removes the surface and near surface contamination. A recirculation system agitates the electrolyte. There is on-line filtration of the suspended solids and on-line removal of dissolved metals by ion exchange. The ion exchange column is connected to permit back-flushing and regeneration, with the regeneration solution piped into the liquid radioactive waste system for disposal.

TABLE J.1-3. Safety Considerations for Operations During Immediate Dismantlement

Operation	Operation Description	Anticipated Routine Airborne Release	Airborne Release Calculation Format	Potential Hazardous Situations
• Chemical Decontamination				
1 Primary System In Situ Decontamination	Ethylenediaminetetraacetic acid (EDTA) in combination with citric and/or oxalic acid in 1-5% solution fills the primary system. The pressurized 80°C solution is pumped through the system until contamination is removed.	Liquid aerosols, chemical vapors droplets	Solution radioactivity x density x aerosol concentration (1) = airborne radioactivity from 1 l of solution Airborne radioactivity x l/min of leak = total release	Loss of service, spills, leaks, valve and weld failure (pump seal failure, burned out motors) Heating during recirculation
2 Final Decontamination Station	Nonactivated stainless steel material will be electropolished. A tank filled with 75% H ₂ O ₂ will be installed in a new fuel storage area. Current flow will remove surface material from the piece until it is clean. The piece is then rinsed in clean water.	Liquid aerosols, chemical vapors	Solution radioactivity x density x aerosol concentration x gas flow/min = airborne radioactivity/min Airborne/min x total time = total release	Loss of service, spills, leaks, dropping of pieces during transport, hydrogen gas explosion, burning/explosion of ion exchange resins.
• Surface Cleaning Procedures				
1 Sweeping Compounds	Small quantities of loose contamination on floors are removed by sweeping compounds	Dust during sweeping	Surface radioactivity/m ² x resuspension factor = airborne radioactivity/m ³ Airborne radioactivity/m ³ x affected volume in building = total release.	Spilling of contaminated compounds
2 Household Detergents	Limited areas cleaned manually using liquid solutions of household detergents.	Liquid aerosols during wringing of mops, sponges	Not calculated	Spilling of contaminated liquids
3 Aerosol Foaming Type Detergents	Cleaned manually	Liquid aerosols and droplets during wiping. Aerosols during spraying	Not calculated	Used with inadequate ventilation.
4 High Pressure Spray	Variable pressure, high velocity, water jets, hand held, operated at pressures up to 3 x 10 ⁶ kPa (4500 psig)	Liquid aerosol generated by decontaminating "fog" (a)	Surface radioactivity x fraction removed x surface area/min = radioactivity removed/min liters used radioactivity = radioactivity/liter Density x radioactivity/l x aerosol concentration x l/min = radioactivity airborne/min Airborne release/min x time = room release fraction of building = total release	Loss of equipment, services, or local airborne contamination control when used
• Physical Decontamination of Surfaces				
1 Drilling of Concrete Surfaces	Conventional concrete drill will be used to drill holes 2.54 cm (1 in.) wide, 10 cm (4 in.) deep, for insertion of rock splitter. The holes for the explosive will be of different geometries, depending on the need. Some may be drilled parallel to and above 12.7 cm (5 in.) from the concrete to be scarfed, drilling the entire length of the surface. Other holes may be 12.7 cm (5 in.) deep, perpendicular to the surface to be scarfed. These may be on 0.3048 m (1 ft) centers over the entire surface.	Concrete dust during drilling, concrete shards	Volume of hole x density = g/hole of concrete Surface radioactivity per 2.54 cm diameter hole = radioactivity/g Radioactivity x airborne dust concentration ^(a) x airflow = airborne radioactivity per minute generated Radioactivity/min x total time = release	Loss of services equipment failure (drill breaking) or local airborne contamination control
2 Rock Splitter	Concrete will be spalled using a hydraulically operated rock splitter or multisplitters (more than one on a predetermined lattice). The splitter is inserted into a drilled hole and the concrete is sheared over a 10-12.7 cm (4-5 in.) diameter area. It is a portable 1-man device.	Concrete dust	Same as above	Loss of services and local airborne contamination control break in hydraulic lines, dropping contaminated rubble containers
3 Explosives	Commercial explosives both solid and liquid will be used as the situation requires. Area size will be controlled to minimize damage potential in event of accident. Charges sequenced to minimize shock waves. The segment will be covered with blasting mat and a water spray may be used to control airborne concrete dust. The scarfed areas will be protected from recontamination during further operations by painting or covering with heavy plastic or canvas.	Concrete dust	Radioactivity/g x anticipated airborne ^(b) concentration x operation volume = airborne radioactivity/explosion Number of explosions x airborne radioactivity/explosion = total release	Loss of services and local contamination control, over-pressurization from blast, inadvertent detonation of explosives, loss of blasting mat loss of water spray
4 Rock-jacking	A pneumatic jackhammer can be used for removal of walls up to a thickness of 0.6 m (2 ft). Rock-jacking may be limited because of time requirements.	Concrete dust	Not calculated	Loss of service and local airborne controls, break in compressed air lines
• Removal of Contaminated Concrete Rubble	Rubble will be loaded manually or with a shielded front-end loader	Concrete dust	% of solid material becomes dust inside the contamination control envelope	Loss of service, dropping of contaminated concrete rubble, spilling of concrete rubber container, explosion of LPG loaded from front-end loader
• Segmenting Equipment				
1 Arc Saw	91 cm (36 in.) diameter, 64 cm (1/4 in.) thick sawblade, 800 RPM blade cooled with water spray installed in blade guard	Metal fragments and vapors	Length x thickness x cut width x density = wt vaporized/cut Length x cut width x surface radioactivity = radioactivity/cut radioactivity/cut time/cut = airborne radioactivity/min activity/min x operation time = total release	Loss of equipment, services and local airborne contamination interior surfaces are not decontaminated
2 Plasma Torch	Extremely high-temperature, high velocity constricted arc between the electrode in the plasma torch and the piece to be cut. Energy is concentrated upon a small area and its intense heat melts the metal.	Condensed metal, chemical vapors from residual materials surfaces, gases, smoke particles fumes and spatter	Same as above	Loss of equipment, and local airborne contamination control, segmentation of equipment not chemically decontaminated, ignition of combustible materials inadvertently left in area
3 Oxyacetylene Torch	Compressed oxygen and acetylene flame form a cutting jet	Condensed metal, vapors	Same as above	Loss of service, equipment, local airborne control, explosion hazard, fires, aerosols.
4 Linear-Shaped	Remote self-contained explosive Charge units. A tubular housing filled with the explosive cyclotrimethylenetrinitramine, RDX	Liner material that is a jet of hot metal particles, vaporized metal, gaseous metal fragments		Explosions, shock waves, inadvertent detonation, aerosols.

(a) 10 mg/m³ is the air concentration of droplets or dust from routine operations
 (b) 100 mg/m³ is the air concentration of dust generated by blasting operations.

TABLE J.1-4. Inventory of Anticipated Surface Contamination in the Reference PWR Buildings

Dose Point ^(a)	Location	Dimensions, m (ft)	Inventory of Surface Contamination			
			After Shutdown ^(b)			After Manual Chemical Decon Total ^(h) μCi
			DPM/100 cm^2	$\mu\text{Ci}/\text{cm}^2$	Total μCi	
Containment						
1A-05	Floor Drain	— ^(c)	>100K	4.54×10^{-4}		
1A-06	Recirculating Pump	—	>100K	4.54×10^{-4}		
1A-07	General Area	604 m^2 + 4.9 m high wall ^(d,e) (6509 ft^2 + 18 ft wall)	1-32K ⁽ⁱ⁾	1.4×10^{-4}	1.7×10^2	1.7×10^2
1A-10	Pressurizer Relief Tank	—	500K	2.3×10^{-3}		
1A-19	Systematic Piping	—	$\geq 100\text{K}$	4.45×10^{-4}		
4A-01	Reactor Cavity Inside Edge	33.8 m^2 + 15 m wall (348 ft^2 + 50 ft wall)	<7000K ^(f)	3.2×10^{-3}	1.1×10^4	$1.1 \times 10^{(f)}$
4A-03	Steam Generator Area (Floor Surfaces)	$\sim 655 \text{ m}^2$ + 15 m wall ^(g) (7069 ft^2 + 50 ft wall)	20-200K	9.00×10^{-4}	1.6×10^4	1.6×10^3
Control Building						
1A-24	Laundry Room	4.3 x 5.8 x 4.6 (14 x 19 x 15)	≥ 20	9×10^{-3}	1.1×10^{-1}	1.1×10^{-1}
Auxiliary Building						
1A-25	Radiation Sampling Room	5.2 x 6.1 x 4.6 (17 x 20 x 15)	≥ 20	9×10^{-3}	4.28×10^2	43
1A-26	Radiological Chemistry Laboratory	5.3 x 8.8 x 4.6 (17.3 x 29 x 15)	≥ 100	4.5×10^{-7}	3	0.3
2A-11	Waste Gas Decay Room	5.3 x 20 x 4.6 (17.5 x 65 x 15)	5-10	4.5×10^{-3}	0.7	0.07
2A-12	Pipeway	6.1 x 13.7 x 4.6 (20 x 45 x 15)	100-1000	4.5×10^{-4}	58	5.6
2A-15	Volume Control Tank Area	3.81 x 4.7 x 4.6 (12.5 x 15.5 x 15)	100-1000	4.5×10^{-4}	12	1.2
2A-20	Waste Evaporator Room	6.1 x 8.4 x 4.6 (20 x 27.5 x 15)	10-300K	1.4×10^{-3}	1.1×10^{-4}	1.1×10^3
2A-23	Waste Gas Compressor Room	5.0 x 4.7 x 4.6 (16.5 x 15.5 x 15)	10-150	6.8×10^{-3}	2.4	0.2
2A-25	Demineralizers Area	—	10-100	4.5×10^{-3}		
5A-01	Residual Heat Removal HX Room	3.4 x 5.6 x 6.1 (11.3 x 18.3 x 20)	30-1000K	4.5×10^{-3}	5.7×10^3	5.7×10^2
5A-02	Auxiliary Building Sumps	—	$\leq 750\text{K}$	3.4×10^{-3}		
5A-03	RHR Pumps (Contract)	—	4-200K	9.00×10^{-4}		
5A-04	Safety Injection Pump Rooms	3.2 x 5.7 x 6.1 (10.5 x 18.8 x 20)	1000	4.5×10^{-4}	16.5	1.7
6A-01	Boron Injection Tank Room	3.7 x 5.9 x 6.1 (12.3 x 19.3 x 20)	10-50	2.2×10^{-3}	1	0.1
6A-02	Laundry Drain Tank Pump Room	3.9 x 22.7 x 6.1 (12.8 x 75 x 20)	≤ 50	2.2×10^{-3}	4	0.4
6A-05	Charging Pump Room	8.5 x 6.6 x 6.1 (28 x 21.8 x 20)	15-30K	1.4×10^{-4}	2	0.2
Fuel Building						
1A-27	Waste Holdup Tank Rooms (Typical)	6.5 x 21 x 4.6 (21.3 x 67.5 x 15)	≤ 300	1.4×10^{-4}	28	2.8
1A-28	Component Cooling Water HX Area	9.1 x 16 x 4.6 (30 x 52.5 x 15)	≤ 100	4.5×10^{-3}	9.9	0.99
2A-08	Spent Fuel HX	—	10-20	9×10^{-4}		
2A-10	Drumming Room	8.9 x 6.3 x 4.6 (29.1 x 20.8 x 15)	500-5000	2.3×10^{-3}	40.5	4.1
3A-08	Boric Acid Evaporator Room	6.0 x 9.0 x 4.6 (19.6 x 29.6 x 15)	40-500K	2.3×10^{-3}	4.4×10^3	4.4×10^2
3A-14	Spent Fuel Pool Skimmer Filters	—	1-10K	4.5×10^{-4}		
4A-06	Controlled Access Machine Shop	10 x 18 x 12 (34 x 60 x 40)	100-500	2.3×10^{-4}	1.74×10^2	17

(a) Refer to Table 7.4.2 for specific locations

(b) Assume the inventory is on floor and wall (excluding ceiling which is not readily contaminated) if an area is listed the value is extrapolated to entire room

(c) A dash indicates internal surface contamination of an undetermined area, and no total radioactivity can be determined

(d) Estimated area of floor space available for foot traffic and decon, etc

(e) Estimated from drawings, scale derived from Figure A.2.19

(f) The Reactor Cavity will have 90% of activity removed by high pressure spray

(g) The containment dome is not considered a likely surface for deposition of particles

(h) Assume a decontamination factor of 10 by manual decontamination methods

(i) 1-32K = 1,000 to 32,000 DPM/100 cm^2

**TABLE J.1-5. Areas with Highest Radionuclide Inventories
Where Various Operations are Performed**

<u>Locations and Operations</u>	<u>Radionuclide Inventory at Beginning of Operation (μCi)</u>
Containment Building	
Nonactivated Components	
Surface Cleaning Operation Steam Generator Area	1.6×10^4
Arc Saw RCS Piping	1.62×10^6
Oxyacetylene Steam Generator	4.26×10^8
Plasma Torch Primary Piping - 2 to 14" OD	6.2×10^6
Drilling Steam Generator Enclosure	$1.6 \times 10^3^{(a)}$
Explosives/Rock Splitter Steam Generator Enclosure	$1.6 \times 10^3^{(a)}$
Pneumatic Jackhammers Areas where use is indicated by radiation survey.	
Activated Components	
Arc Saw Vessel Internals	3.9×10^{12}
Oxyacetylene Vessel Shell	1.4×10^{10}
Drilling Bioshield	$<1 \times 10^8$
Explosives Bioshield	$<1 \times 10^8$
Front End Loader	$<1 \times 10^8$
Auxiliary Building ^(b)	
Surface Cleaning Operation Residual Heat Removal Hx Room	1.7×10^4
Fuel Building	
Surface Cleaning Operations Boric Acid Evaporator Room	4.4×10^3
Control Building	
Surface Cleaning Operations	1.1×10^{-1}

(a) After manual decontamination operations, DF = 10.

(b) The relative low levels of contamination indicated in buildings other than containment precluded consideration of operations other than decontamination and radiation survey in the safety assessment.

In all, 2769 pieces of contaminated RCS and CVCS piping are electropolished. The piping is cut into 1.5 m (5ft) and 2.1 m (7 ft) pieces after the in situ decontamination. The inventory of radioactivity before the in situ decontamination was 16 Ci for the RCS piping and 60 Ci for the CVCS piping (Table 7.3-8). Assuming 90% decontamination, the inventories immediately before the electropolishing are 16 Ci for the RCS piping and 6.0 Ci for the CVCS piping (Table J.1-6). The 54 pieces of RCS piping have the highest radionuclide inventory level at about 0.3 Ci per piece.

The system requires $2.8 \text{ m}^3/\text{hr}$ per 0.092 m^2 ($100 \text{ ft}^3/\text{min}$ per 1 ft^2) airflow per surface area to remove chemical vapors. For the floor space in this area, a total airflow of $67.2 \text{ m}^3/\text{min}$ ($2400 \text{ ft}^3/\text{min}$) is required for ventilation. Additional airflow is required to dilute the hydrogen gas generated during electropolishing.

The formula to calculate the hydrogen generation rate is:

$$\left(\frac{\text{Surface Area}}{\text{Area}} \right) \left(\frac{\text{Applied Current}}{\text{Current}} \right) \left(\frac{\text{Current Duration}}{\text{Time}} \right) \left(\frac{\text{Conversion}}{\text{Factor}} \right) \left(\frac{\text{Volume of H}_2}{\text{Gas Generated}} \right) \quad (\text{J.1})$$

TABLE J.1-6. Approximate System Internal Surface Radionuclide Inventories During Dismantlement Operations

System	At Shutdown		After in Situ Chemical Decontamination	
	$\mu\text{Ci}/\text{cm}^2$	Total Ci	$\mu\text{Ci}/\text{cm}^2$	Total Ci
Reactor Vessel and Internals	23	130	2.3	13
Steam Generators	23	4400	2.3	440
Piping (except RCS)	6.0	60	0.60	6.0
Pressurizer	4.0	4.0	0.40	0.40
RCS Piping	85	160	8.6	16

An estimate of the airborne releases from electropolishing is made by considering the decontamination of a piece of the largest, most contaminated section of RCS pipe. Using Equation J.1, if 10,000 A are applied to decontaminate a 1.5 m (5 ft) piece of RCS pipe, the hydrogen gas generation rate is

calculated to be $6.2 \text{ m}^3/\text{min}$ ($216 \text{ ft}^3/\text{min}$). The airflow in the new fuel storage area is $119 \text{ m}^3/\text{min}$ ($4260 \text{ ft}^3/\text{min}$), which dilutes the H_2 gas to a concentration of 5.2 volume %. Since the lower flammability limit for H_2 in air is 4.1%,⁽⁴⁾ additional sweep air is provided to dilute the H_2 gas to a concentration below this level. The radioactivity in the electrolyte from decontamination of this RCS section is estimated to be $61 \text{ } \mu\text{Ci}/\ell$, which is the product of the surface area of the pipe ($3.8 \times 10^4 \text{ cm}^2$), the surface contamination level ($8.6 \text{ } \mu\text{Ci}/\text{cm}^2$) and the volume of solution in the tank (5400ℓ). The solution is continuously cleaned by filtration, so the radioactivity level does not increase with time. Sources of the radioactive airborne release from the electrolyte solution are water droplets, splashes, drips, and gases from bubble formation during the evolution of both H_2 and O_2 gases. Since O_2 is generated at half the volumetric H_2 generation rate, the total rate of gas generation available to cause an airborne release is $9.3 \text{ m}^3/\text{min}$ ($324 \text{ ft}^3/\text{min}$). It is assumed that a valid estimation of the airborne concentration of liquid electrolyte is $10 \text{ mg}/\text{m}^3$.⁽⁵⁾ This value agrees with the airborne levels observed at the Elk River Reactor.⁽²⁾ With these assumptions in this maximum case analysis, a total of $4 \times 10^{-3} \text{ } \mu\text{Ci}/\text{min}$ is released to the building ventilation system. The electropolishing decontamination station operates using 2 shifts, 5 days per week, for a total operational time of 60 hr/week for about 108 weeks. Therefore the total release during the entire operation is calculated to be $1.56 \times 10^{-3} \text{ Ci}$ to the building. Assuming a HEPA transmission factor of 5×10^{-4} , the airborne release from the building is $7.8 \times 10^{-1} \text{ } \mu\text{Ci}$.

J.1.1.2 Segmentation of Nonactivated Stainless Steel

Piping sections are cut into sections not exceeding 2.1 m (7 ft) in length. The RCS piping and other primary piping are described in Table A.3-5 of the Facility Description. All other piping systems, some potentially

⁽⁴⁾ H. F. Coward and G. W. Jones, Limits of Flammability of Gases and Vapors. Bulletin 503, U.S. Bureau of Mines, 1952.

⁽⁵⁾ Siting of Fuel Reprocessing and Waste Management Facilities. ORNL-4451, Oak Ridge National Laboratory, Oak Ridge, TN, pp. 8-47, July 1970.

⁽²⁾ Final Elk River Reactor Program Report. COO-651-93, Revised, p. F-9, United Power Association, Elk River, MD, November 1974.

contaminated and some uncontaminated, are also sectioned for removal. Stainless tank and pool liners are cut into sheets no larger than 1.2 m x 2.1 m (4 ft x 7 ft) for disposal. Each steam generator is segmented (Figure G.4-3) and closure plates are welded on. A partial listing of the equipment that is planned to be segmented is described in Table G.1-1 of this document.

Several cutting methods will be used: oxyacetylene torches, plasma torches, arc saws, and linear shaped charges. Table J.1-7 shows a comparison of the kerf (cut) width for the various methods. The arc saw, with a 0.95 cm (3/8 in)⁽⁶⁾ kerf, produces the most material that can become airborne per cut. It is one method recommended for segmenting the RCS piping, the item estimated to have the highest radioactivity level of the items to be cut.

TABLE J.1-7. Cutting Parameters

<u>Cutting Method</u>	<u>Kerf (Cut) Width</u>		<u>Airborne Release</u>
	<u>cm</u>	<u>in.</u>	
Arc saw	0.95	3/8	Metal fragments, vapors
Plasma torch	0.31-0.635	1/8-1/4	Condensed metal, gases, smoke particles
Oxyacetylene torch	0.635	1/4	Condensed metals, vapors
Linear-shaped charges	NA Material fuses		Hot metal particles, fragments, gases from detonating explosive

Therefore cutting RCS piping with an arc saw gives the most potential for airborne material and is used to estimate the airborne release.

The RCS piping is cut into 54 segments, requiring 55 cuts since it is a closed system. The airborne release estimates are based on the segmentation of one of the largest sized sections with an ID of 78.7 cm (31 in.) and thickness of 6.6 cm (2.59 in). Equations J.2 and J.3 show the information required

⁽⁶⁾ M. P. Schlienger and G. A. Beitel, Topical Report: Arc Saw Testing. ARH-LD-147, Atlantic Richfield Hanford Co., Richland, WA, February 18, 1976.

to calculate the total mass of material and activity generated during segmentation operations.

$$\begin{aligned} & \left(\text{Number of cuts} \right) \bullet \left(\text{Length of ID per Cut} \right) \bullet \left(\text{Material Thickness} \right) \bullet \left(\text{Kerf Cut} \right) \bullet \left(\text{Material Density} \right) = \\ & \left(\text{Total Mass of Material Generated} \right) \end{aligned} \tag{J.2}$$

$$\left(\text{Number of Cuts} \right) \bullet \left(\text{Length of ID per Cut} \right) \bullet \left(\text{Kerf (cut) width} \right) \bullet \left(\text{Surface Contamination Level} \right) = \left(\text{Total Activity Generated} \right) \tag{J.3}$$

Using Equation J.2 and information from Table J.1-7, with a density of stainless steel of 8.0 g/cm³, the total mass of material generated by the 55 cuts is 354 kg (781 lb). Using Equation J.3 with the information from Table J.1-7 and the in situ chemical decontamination surface activity levels of Table J.1-6 for RCS piping, the total radioactivity removed for 55 cuts is 0.15 Ci. It is further assumed that all of this material is made airborne.

Ninety percent of the reactor coolant system radioactivity is expected to be in the steam generators, so segmenting the steam generators gives the maximum expected release levels encountered during dismantlement of the reactor coolant system. The segmenting is done in an enclosed contamination control envelope, and the release is filtered through a HEPA filter. Using an oxyacetylene torch, each steam generator is cut into 8 segments, requiring 7 circumferential cuts. Cutting occurs after primary coolant side decontamination, so there is a total of 2.2 µCi/cm² of residual contamination (Table J.1-6). Six of the circumferential cuts bisect the 3388 U-tubes in the steam generator. Using Equation J.3, and calculating the surface area per U-tube cut, the total airborne radioactivity for the 6 cuts on all 4 steam generators is 1.5 Ci. The HEPA filter has a transmission facotr of 5 x 10⁻⁴, so the total estimated release to the Reactor Containment Building is 750 µCi. It is assumed that the containment control envelope leakage rate equals 10% of the amount passing through the HEPA filter, thus an additional 75 µCi are estimated to be released to the Reactor Containment Building from the operation.

J.1.1.3 Segmentation of the Reactor Vessel and Internals

The segmenting of the internals and the reactor vessel are difficult and potentially hazardous operations. To minimize the hazards, the vessel internals are remotely cut under water, with a contamination control envelope in place above the water surface. The pressure vessel is also cut remotely, but not under water, inside a contamination control envelope. An arc saw or explosives are used for cutting the internals. The arc saw, with a 0.95 cm (3/8 in.) kerf, is used to estimate maximum airborne releases. The reactor pressure vessel is cut with oxyacetylene torches, and a 0.635 cm (1/4 in.) kerf is assumed for this safety evaluation.

The radioactivity levels vary with location. Table J.1-8 shows the radioactivity level in the major activated reactor components and the cutting data for estimating the generation rates of airborne material. Table G.1-1 gives a summary of cutting data and shows a cutting rate of 99 cm/hr (39 in./hr) for a nominal material thickness of 7.6 cm (3 in.). Using Equation J.2 and data from Table J.1-6, the mass rate of material cut is calculated to be 5.76×10^3 g/hr (12.7 lb/hr).

Cutting the vessel internals is accomplished under water, so the scrubbing action of the water contains much of the radioactive material, thus keeping the airborne radioactivity minimal. The vessel is cut inside a contamination control envelope with a HEPA filter to remove airborne contamination before the air is discharged to the Reactor Containment Building.

The cover water becomes contaminated with the material generated by the cutting. This is a solid material in suspension and/or in solution or particles settled to the bottom of the reactor cavity where the cutting is accomplished. The movement associated with the cutting agitates the water and causes droplets of contaminated water to be ejected into the air space above the water. The radioactivity in the water is increasing during the cutting operation. To make a conservative estimate of the airborne release, the total release for the entire operation is calculated using the maximum radioactivity level reached by the liquid.

TABLE J.1-8. Radioactivity Levels in Major Activated Reactor Components and Cutting Data for Estimating the Generation Rates of Airborne Material

Material Being Cut	Avg. Ci/g	Material Cutting Rate ^(a)		Airborne Contamination ^(b) Control Systems	
		Ci/hr	Ci/min		
Shroud	0.28	1624	27	Contamination Control Envelope	Cut in Water Remotely
Lower 4.72 m of Core Barrel	0.024	139	2.3	↓	↓
Thermal Shields	0.014	81	1.4		
Cladding	7×10^{-4}	4	0.07		
Lower Grid Plate	0.14	812	14		
Upper Grid Plant	5×10^{-3}	29	0.5	↓	↓
Lower 5.02 m of Pressure Vessel Wall (shield)	7.2×10^{-5}	0.42 ^(c)	0.007 ^(c)		
					Cut in Air Remotely

(a) Normal generation rate of 5.8 kg/hr (12.7 lb/hr).

(b) Table G.1-1.

(c) Cutting rate 3.8 kg/hr (8.4 lb/hr).

For 960 hours of cutting at 5.8 kg/hr, the total material cut is 5.5×10^3 kg. The total radioactivity calculated from the data in Table 7.3-2 (excluding the vessel wall which is cut in air) is calculated to be 9.3×10^{-2} Ci/g. The total radioactivity available in solution/suspension is 5.1×10^5 Ci. In the maximum case, all of this radioactivity is assumed to be in solution or suspension in the water. The volume of water in the reactor pressure vessel during the cutting is 126 m^3 , therefore the radioactivity level in the water is calculated to be $4.05 \times 10^{-3} \text{ Ci/cm}^3$. Thus, at an airborne concentration of water droplets of 10 mg/m^3 and an airflow of $28 \text{ m}^3/\text{min}$ ($1000 \text{ ft}^3/\text{min}$) for 960 hours of cutting, $6 \times 10^7 \text{ } \mu\text{Ci}$ are made airborne at a rate of $298 \text{ } \mu\text{Ci}/\text{min}$. With a HEPA filter transmission factor of 5×10^{-4} , $3 \times 10^4 \text{ } \mu\text{Ci}$ are released to the Reactor Containment building.

The lower 5.02 m of the reactor pressure vessel wall (shield) is cut into 71 segments using an oxyacetylene torch. The operation is done in air, remotely, within a contamination envelope. The 21.6 cm (8.5 in) thick material is cut at a rate of 35 cm/hr (14 in/hr) (Table G.1-2). Using Equation J.3 and the information in Table J.1-8, 4623 $\mu\text{Ci}/\text{min}$ are cut. If the airborne material is similar to that at the Elk River Reactor, there will be about $10 \text{ mg}/\text{m}^3$ of dust.⁽²⁾ Assuming an airflow of $28 \text{ m}^3/\text{min}$ ($1000 \text{ ft}^3/\text{min}$), 20 $\mu\text{Ci}/\text{min}$ are carried to the enclosure ventilation. The entire operation is estimated to require 692 hours, so a total of $8 \times 10^5 \mu\text{Ci}$ are generated. The release to the Reactor Containment Building through the HEPA filter is calculated to be 415 μCi .

Since both of these cutting operations occur in the reactor cavity, and a tight seal can be achieved at this point during dismantlement, leakage past the seal is assumed not to contribute significantly to the airborne release.

J.1.1.4 Waste Handling

Airborne releases of contaminated concrete potentially could occur during the loading of rubble. The use of a fine water spray during loading operations reduces air releases since the surface tension of the water plus the normal adhesive forces of the concrete must be overcome before particles are made airborne. The concrete rubble is loaded by hand or mechanical operations. Since about $8.6 \times 10^5 \text{ kg}$ ($1.9 \times 10^6 \text{ lb}$) of bioshield concrete are removed, mechanical operations are desirable. Some breaking of the larger pieces of concrete rubble is anticipated during mechanical loading of the debris, exposing new contaminated surfaces. Dry material is also exposed by removing upper layers. It is estimated that 0.1% of the radioactivity generated as dust during operations such as drilling concrete, is released during the loading operation. The bioshield concrete has an estimated radioactivity content of 0.11 $\mu\text{Ci}/\text{g}$. Thus, a routine waste handling operation could release about 0.36 g to the Reactor Containment Building containing 0.04 μCi for each minute of loading rubble. Six months are required to remove the bioshield.

⁽²⁾ Final Elk River Reactor Program Report. C00-651-93, Revised, p. F-9, United Power Association, Elk River, MD, November 1974.

Assuming the loading takes place as it is removed, twenty-four 60-hr weeks are required for a total estimated airborne release into the Reactor Containment Building of 3.5×10^3 μCi . A total of 1.75 μCi are released through the building HEPA filter to the environment.

Radioactive liquids were generated during reactor operation and standard methods are assumed to be in place for their handling and disposal during the dismantling operations. Contaminated chemical solutions are generated from decommissioning operations such as the in situ decontamination, the final decontamination station, the segmentation of the vessel internals, and other miscellaneous operations. These solutions potentially have high radioactivity levels and are treated before disposal. They are evaporated to reduce their volume prior to solidification. This type of operation generates airborne radioactive materials as droplets or vapors. The potential release of airborne material depends on the method of operation.

Two liquid waste treatment systems operate during the plant operation. They are the Clean Radioactive Waste (CRW) and Dirty Radioactive Waste (DRW) systems, and are discussed in detail in Appendix A of this report.

Solutions from the final decontamination station were estimated to have a contamination level of 61 $\mu\text{Ci}/\ell$ in Section J.1.1.1. For the sake of illustration, the contamination is assumed to be in a liquid solution whose level of radioactivity makes it a candidate for processing by the CRW system. The distillate is assumed to have a contamination level that is reduced by 10^3 ; therefore the concentration is about 6.1×10^{-2} $\mu\text{Ci}/\ell$. This is a closed system and as such does not normally allow an interface between the contaminated distillate and air outside the system. The concentrates are in the form of sludges that are not readily made airborne. The spent resins are not anticipated to be of higher radioactivity levels than those encountered during the operating life of the plant. Thus, the solid waste system contributes an insignificant airborne radioactivity release.

J.1.1.5 Surface Cleaning

Several different techniques are used for manual chemical decontamination. These include the use of hand-held high-pressure water jets, sweeping compounds, aerosol-type foaming cleaners and household detergents.

Evaluating the release from each separate item does not produce results commensurate with the effort involved. Thus, the maximum release during a specific type of operation in an area with highest contamination level is assumed to be representative of that type of operation throughout the facility. Releases calculated on this basis tend to represent realistic maximum values.

The hand-held water lance jet. This device is operated by one man and requires a gauge pressure up to 3.1×10^4 kPa (4500 psig). Appropriate radiation protection is anticipated to be used.

The principal mechanism for airborne activity entrainment using the lance is the suspension of liquid droplets containing contamination. When the lance is sprayed initially, it produces droplets up to 300 micrometers in size or smaller, the size of fogs or mists. The spray particles then break into smaller particles when they impact on a surface. Thus, there are significant amounts of small airborne droplets with considerable variation in the amount airborne.

Direct data are not available to define the quantity of droplets formed. A conservative estimate is made by assuming that a sufficient quantity of droplets are generated to maintain an airborne liquid concentration of 10 mg/m^3 with vigorous mixing in air. This is the maximum mass concentration that is found in air velocities less than 0.046 m/s (0.15 fps).⁽⁵⁾ There is fairly constant weight distribution in a spray of 10 to 20 micrometer particles of about 10 mg/m^3 even with the gross entrainment of larger particles.⁽⁷⁾ Since ten micrometer particles and smaller are in the respirable size range, they are potentially hazardous. The quantity of activity in these airborne droplets is influenced by many factors such as the quantity of radioactivity on the surface, the ability of the liquid to entrain radioactivity, and the contact between the liquid and the surface.

Summarizing the previous discussion: 1) the level of surface contamination is anticipated to be fairly low, 2) the lances are used intermittently,

⁽⁵⁾ Siting of Fuel Reprocessing and Waste Management Facilities. ORNL-4451, Oak Ridge National Laboratory, Oak Ridge, TN, pp. 8-47, July 1970.

⁽⁷⁾ E. H. Garner, S. R. M. Ellis and J. A. Lacey, "The Size Distribution and Entrainment of Droplets." Trans. Inst. Chem. Engrs. 32:222-235, 1954.

producing short bursts of droplets, 3) the airborne concentration does not exceed 10 mg/m^3 , and 4) the radioactivity level depends on the surface contamination of the area.

The operating parameters for the high pressure spray vary with the requirements of the situation. For the surfaces considered in this study, 23 to 30 ℓ/min (6 to 8 gal/min) and 46.5 to 65 m^2/hr (500 to 700 ft^2/hr) are reasonable estimates of the range of the solution flow rate and surface cleaning rate respectively.

This technique is used in various areas including the refueling cavity and spent fuel pool. Releases are calculated for areas with the highest tabulated surface contamination in each building. The releases are tabulated in Table J.1-9, and are calculated assuming a water flow of 26.5 ℓ/min , a density of 1 g/cm^3 and 90% of the surface contamination removed by this method. This operation will not be continuous because time will be required for repositioning equipment and moving personnel during the operation of the jets. This moving additionally contributes to airborne radioactivity. An airborne concentration and airflow will be maintained during periods when the jets are not actually being applied. Therefore, sufficient resuspension could occur to sustain aerosols equivalent to 10 mg from each m^3 of liquid.

Two areas that are selectively cleaned with the hand-jet lance are the reactor cavity from the flange up, and the spent fuel pool. As the water in the pool is lowered, the high pressure spray removes the contamination. Particulates settled on the spent fuel pool floor are removed with an underwater vacuum cleaner. It is assumed that the pool bottom has the same level of contamination as the rest of the pool after this operation.

TABLE J-1.9. Postulated Airborne Release from Selected Areas Using High Velocity Water Jets^(a)

Area (Building)	Total Surface m ²	$\mu\text{Ci}/\text{m}^2$	Contamination Removed/min, μCi	Solution Concentration		Airborne Concentration $\mu\text{Ci}/\text{m}^3$	Aerosol Generation Rate, $\mu\text{Ci}/\text{min}$	Total Operation Time, min	Total Airborne in Room, μCi	Total Airborne in Building, μCi	Total Released to Atmosphere, μCi
				$\mu\text{Ci}/\ell$	$\mu\text{Ci}/\text{cm}^3$						
Steam Generator (Containment)	2447	9	7.5	0.28	2.8×10^{-4}	2.8×10^{-8}	7.5×10^{-2}	2700	203	302	0.2
Hx Room (Auxiliary)	118	45	37.0	1.4	1.4×10^{-3}	1.4×10^{-5}	3.7×10^{-1}	128	47	7.8×10^3	3.9
Borix Acid Evap. Room (Fuel)	153	23	19.0	0.7	7.0×10^{-4}	7.0×10^{-8}	1.9×10^{-1}	166	32	2.0×10^3	1.0
Ion Exchanger Vault	6	3.75×10^4	3.1×10^4	1.2×10^3	1.2×10^0	1.2×10^{-2}	3.1×10^2	6.5	2.0×10^3	1.0 ^(b)	5×10^{-4}
Laundry Room (Control Bldg.)	118	9×10^{-4}	7.5×10^{-2}	2.8×10^{-5}	2.8×10^{-8}	2.8×10^{-10}	7.5×10^{-6}	128	9.5×10^{-4}	4×10^{-3}	1.9×10^{-6}
Reactor Cavity, Spent Fuel Pool	1141	320	2.6×10^{-2}	9.8	9.8×10^{-3}	9.8×10^{-5}	2.6	1240	(c)		0.2

(a) The Water Jet cleans about 0.92 m² per minute and uses 26.5 ℓ per minute of solution

(b) Contamination control envelope in place with a HEPA filter in place

(c) 2886 μCi is released directly into the spent fuel pool exhaust with two banks of HEPA filters in place, for a release through the filters of 7.2×10^{-4} μCi , 340 μCi total is released in the containment

The estimated surface area of the spent fuel rack is 120 m^2 (1300 ft^2). The radioactivity level found at the reactor cavity inside edge is $3.2 \times 10^{-2} \text{ } \mu\text{Ci}/\text{cm}^2$ and is assumed to be the maximum value. This value is used to calculate the airborne release. The calculated release value is $2.6 \text{ } \mu\text{Ci}/\text{min}$, and the total operation releases $3.5 \times 10^3 \text{ } \mu\text{Ci}$ to the ventilation systems.

A contamination control envelope can be used with the hand-jet lance to reduce airborne releases from the reference PWR. An example of this case is the decontamination of the ion exchanger vaults. A leak of 1 liter per day for the 40-year plant life is assumed to occur in one of the vaults, to give a realistically high surface contamination level. Since these vaults are relatively small, with a floor area of only 6 square meters, a contamination control envelope with a HEPA filter system reduces the $2.0 \times 10^3 \text{ } \mu\text{Ci}$ airborne in the room to only $1 \text{ } \mu\text{Ci}$ airborne in the building. The total release to the atmosphere through the building HEPA filter is calculated to be $5 \times 10^{-4} \text{ } \mu\text{Ci}$ for this operation.

Sweeping Compounds. Small quantities of loose floor contamination are effectively removed by using sweeping compounds. The potential airborne contamination from sweeping particles has been determined experimentally for sweeping 10 minutes. The resuspension factor was $1.9 \times 10^{-2} \text{ m}^{-1}$,⁽⁸⁾ and relates to the amount of radioactivity in the air above a surface where the operation takes place.

It is doubtful if sweeping compounds will be used extensively, so they have not been included as an airborne release from dismantlement. Since the resuspension factor for sweeping is about one order of magnitude larger than that used to estimate releases from the radiation survey in a later section, the airborne release is also about one order of magnitude larger. This is a conservatively high estimate of the release since sweeping compounds are composed of fairly large particles of cleaner to which the compound adheres, resulting in a lower resuspension potential for the compound. Sweeping is considered in a later section during preparations for Safe Storage.

⁽⁸⁾ B. R. Fish, R. L. Walker, G. W. Royster, Jr. and J. L. Thompson, "Redispersion of Settled Particulates." Surface Contamination, Proceedings of a Symposium Held at Gatlinburg, TN, June 1964. pp. 75-81, 1967.

Resuspending the sweeping compound does not necessarily constitute a hazard unless the compound is inhaled. Schwendiman⁽⁹⁾ determined the probability of inhaling a single particle during sweeping to be about 4.8×10^{-4} .

Aerosol-Type Foaming Cleaners. Sealed surfaces are washed with aerosol-type foaming cleaners in this study. Some airborne particles are generated during the use of the aerosol-generated foaming cleaners. This is an operation controlled directly by the decontamination personnel. If protection for the operator is indicated, a mask is worn. The phase of spraying the cleanser is used briefly and intermittently, so significant amounts of airborne radioactivity are not generated, and this operation is not considered further in this study.

Household Detergent. Household detergent products are used in areas that can be swabbed with sponges. The reuse of the sponges minimizes the generation of solid contaminated waste. The liquids are disposed of into the liquid radioactive waste system. There appears to be little potential for airborne release during the use of household detergents. The small amount of radioactivity made airborne is due to the low radiation levels expected for manual operations. Thus, the routine release to the ventilation system is expected to be insignificant, and is not considered further in this study.

J.1.1.6 In Situ Decontamination of the Reactor Coolant System

The Reactor Coolant System (RCS) and the Chemical and Volume Control System (CVCS) were selected as the principal systems that are chemically decontaminated and are analyzed for airborne radionuclide releases. The decontamination method selected proposes to circulate decontamination solutions through all systems connected with the RCS, and all possible branches of the CVCS system.

This operation begins after the reactor is defueled, the pressure vessel head is reinstalled, and after the circulation system is isolated from the fuel pool system. The systems are then filled with 337,000 ℓ (89,000 gal)

⁽⁹⁾ L. C. Schwendiman, "Probability of Human Contact and Inhalation of Particles." Health Physics. 1:352-356, 1958.

of ethylenediaminetetraacetic acid (EDTA) in combination with citric and/or oxalic acid in 1-5% solution. The system is pressurized to a gauge pressure of ≤ 2800 kPa (400 psig) and heated to 80°C for the operation. The solution is circulated through the system until analysis of samples indicates that the desired decontamination level has been achieved. One system volume is used to minimize the quantity of contaminated liquids, thus extended recirculation time of several days to a week is expected to be required.

The radioactive inventory of the system is about 4600 Ci (Table J.1-6). It is anticipated that 90% (4140 Ci) of this inventory is removed by chemical decontamination. The removed radioactivity is retained in the circulating liquid, in solution and/or as particulates in suspension. The fraction in each phase is not known directly.

The piping forms a closed loop system containing the circulating decontamination solution. Since the system is enclosed, there is normally no interface between the circulating solution and the air outside the system. This indicates a low potential for airborne release outside the system. However, any fluid system has the potential for leaks. Leakage would allow room air to interface with the decontamination solution and permit the airborne entrainment of droplets.

The design leak rate of the operating Reactor Coolant System is 22.7 kg/day or 1.9×10^{-2} ℓ/min (50 lb/day or 5×10^{-3} gal/min). A measured fluid leakage rate during operations at the Robert Emmett Ginna plant was about 1.9 ℓ/min (0.5 gal/min). A maximum allowable leakage from a known source at a rate of 3.8 ℓ/min (1 gal/min) was established at the Trojan plant.⁽¹⁰⁾ This maximum allowable leakage rate for the operating plant is used as the maximum anticipated system leakage during RCS and CVCS decontamination.

Any leak of radioactive solution can be potentially serious and could indicate material failure such as corrosion. Thus, it is anticipated that if leakage is indicated, the source will be identified and corrective action taken.

⁽¹⁰⁾ TROJAN Final Safety Analysis Report. Vol. 3, Amendment 5, pp. 5.2-51-5.2-52, Portland General Electric Co., Portland, OR, October 1973.

For this analysis, a postulated leak was developed in which the maximum concentration of radioactivity in solution and/or suspension is estimated to be 0.012 Ci/l (0.046 Ci/gal). The leak is assumed to be in the form of a stream, drops, or spray. At 80°C, it is anticipated that a solution is in the vapor phase. A stream or drops of liquid do not present a significant airborne release potential. If it is assumed that the airborne concentration in the droplets associated with this stream is 10 mg/m³ and that the solution density is 1 g/cm³, then 10 mg is about 0.01 ml (10⁻⁵ l). Thus, 1.2 x 10⁻⁷ Ci/m³ is the airborne radioactivity concentration of the release in the building. At a leak rate of 3.8 l/min (1 gal/min), 0.46 µCi/min become airborne when released to the Reactor Containment Building. The Reactor Containment Building ventilation system provides for 1.5 air changes per hour, so the release never accumulates. The volume of the Reactor Containment Building is 5.66 x 10⁴ m³.⁽¹¹⁾

The spray type leak gives the maximum airborne release potential for decontamination solutions. It is already airborne and can be immediately entrained in the ventilation airflow. Less than 0.3% is anticipated to be in the size range that could be transported downwind and constitute an inhalation hazard.⁽¹²⁾ This means that a total of 136 µCi/min is calculated to be transported into the ventilation air flow. Again, the Reactor Containment Building airflow does not allow this release to accumulate.

J.1.1.7 Removal of the Bioshield

Explosives are used to remove the activated concrete biological shield. The shield is enclosed within a ventilation confinement envelope during the blasting. Blasting mats prevent projectiles of debris from penetrating the confinement barrier. Fine fog sprays help settle the dust created by the blast.

(11) TROJAN Final Safety Analysis Report. Vol. 3, Amendment 4, p. 5.2-61, Portland General Electric Co., Portland, OR, September 1973.

(12) R. Brown and L. J. York, "Sprays Formed by Flashing Liquid Sprays." AIChE Journal 8(2):149-153, May 1962.

Long, deep holes are drilled for the insertion of explosives. They are drilled from the top of the bioshield, down, parallel to the inside of the concrete surface. The circumference of the bioshield is about 32 m (104 ft) and, if the holes are 0.76 m (2.5 ft) apart, about forty-two 2.54-cm (1-in.) diameter holes are required. The total penetration depth of the bioshield is about 19.5 m (64 ft). A total of 9.7×10^2 kg of concrete is removed in this manner.

Table G.4-3 shows that less than 100 Ci are associated with 885,000 kg (1,950,000 lb) of concrete. Thus, the total radioactivity in the drilled material is calculated to be 1.07×10^5 μ Ci. At a drilling rate of about 0.305 m/min (1 ft/min), and with a total of 820 m (2690 ft) of drilling, 360 g/min are calculated to be generated by drilling. If the airborne concentration of dry material is again assumed to be 10 mg/m³, 4.8 g or 0.53 μ Ci is the maximum amount that will be airborne in the reactor cavity. With an airflow of 28 m³/min (1000 ft³/min), 0.031 μ Ci per minute would challenge the HEPA filter. The total amount for all of the drilling operations is 83 μ Ci. With a filter transmission factor of 5×10^{-4} , 0.04 μ Ci would enter the Reactor Containment Building.

The maximum leakage around the ventilation confinement closure is assumed to be 0.1% of the total airborne radioactivity generated during drilling, or .083 μ Ci. Thus, a total of 0.12 μ Ci enter the Reactor Containment Building from the HEPA filter and leakage. A total of 6.0×10^{-5} μ Ci is released to the atmosphere.

Blasting takes place in the reactor cavity. A contamination control envelope is used. The Elk River experience indicated that airborne contamination could be controlled using blasting mats and a water fog spray system.⁽¹³⁾ At a postulated aerosol concentration of 100 mg/m³, and with the volume of the reactor cavity of 485 m³, a total of 48.5 g is calculated to be in the air in the reactor cavity. At an activity concentration of 0.11 μ Ci/g, 5.3 μ Ci are made airborne per explosion.

(13) D. McCannon and J. F. Nemec, Experiences in Decontamination/Decommissioning of the Elk River Reactor. April 1, 1974, presented at Second AEC Environmental Conference, Albuquerque, NM, April 1974.

If the explosive charges are placed 0.61 m (2 ft) apart in the drilled holes, 1344 explosions are required. At 5.3 μCi /explosion, the total airborne radioactivity in the cavity is 7100 μCi . Downstream of the enclosure HEPA filter, the release is 3.6 μCi . If the leakage is again assumed to be 0.1%, 7 μCi would additionally enter the Reactor Containment Building for a total of 10.6 μCi . The amount entering the atmosphere through the building HEPA filter is calculated to be 5.3×10^{-3} μCi .

J.1.1.8 Radiation Survey

A radiation survey is done during the first phase of decommissioning. Assuming that it is also the last step before demolition of the following buildings: containment, condensate demineralizer, fuel, auxiliary, control, and turbine. Surveys are conducted at other times in the operations. Radiation surveys involve walking through the plant, which can disturb settled radioactivity. There is a correlation between airborne radioactivity and movements of personnel. The general housekeeping practices followed in the plant determine the amount of loose surface contamination, which varies with location in the plant.

The relationship between the amount of material in the air above a surface and the contamination of that surface has been studied experimentally. This relationship is described as S_f and is shown by Equation J.4.⁽¹⁴⁾

$$S_f = \frac{\text{radioactivity air/m}^3}{\text{radioactivity surface/m}^2} \quad (\text{J.4})$$

⁽¹⁴⁾R. T. Brunskill, "The Relationship Between Surface and Airborne Contamination." Surface Contamination, Proceedings of a Symposium Held at Gatlinburg, TN, June 1964. pp. 93-105, Pergamon Press, NY, 1967.

S_f is defined as the resuspension factor and has units of m^{-1} . Experimentally measured S_f values vary widely depending on the experimental conditions, and range from $2 \times 10^{-3}/m$ to $1 \times 10^{-5}/m$.^(14,15,16) Using the surface contamination data in Table J.1-4, and the maximum experimentally determined value of S_f , the maximum airborne releases for the radiation survey were calculated (Table J.1-1). Again, these are the maximum estimated releases based on the contamination levels shown in Table J.1-4 for each building. The release is assumed to be at a height of one meter above the surface.

J.1.1.9 Removal of Concrete Areas

Contaminated concrete surfaces that cannot be cleaned to release levels by surface washing techniques are removed mechanically. Three techniques are described in Section F.1.3 for the surface removal of concrete. They are rock splitting (Figure F.3-3), explosives, and electric or pneumatic hammers. Each of these techniques has a different potential for the airborne release of radioactivity and each is used in different areas within the plant.

Holes are drilled for the insertion of the rock splitters and explosives. The steam generator enclosure is used as the reference area for the calculation of airborne radioactivity generated by drilling. The drilling occurs within a contamination control envelope, and each hole is estimated to be 2.54 cm (1 in.) in diameter and 10.16 cm (4 in.) deep, creating a volume of 50.5 cm^3 (3.14 in^3). Using the density of concrete as 2.31 g/cm^3 (144 lb/ft^3), the dust generated per hole is calculated to be 119 g (0.26 lb). Assuming that all of the radioactivity is on the surface of the concrete, and using

- (14) R. T. Brunskill, "The Relationship Between Surface and Airborne Contamination." Surface Contamination, Proceedings of a Symposium Held at Gatlinburg, TN, June 1964. pp. 93-105, Pergamon Press, NY, 1967.
- (15) H. Glauberman, W. R. Bootman and A. J. Breslin, "Studies of the Significance of Surface Contamination." Surface Contamination, Proceedings of a Symposium Held at Gatlinburg, TN June 1964. Pergamon Press, NY, pp. 169-178, 1967.
- (16) I. S. Jones and S. F. Pond, "Some Experiments to Determine the Resuspension Factor of Plutonium from Various Surfaces." Surface Contamination, Proceedings of a Symposium Held at Gatlinburg, TN, June 1964. Pergamon Press, NY, pp. 83-92, 1967.

the surface contamination level of the steam generator enclosure, 4.6×10^{-4} μCi are removed in the dust per hole drilled. This radioactivity is assumed to be mixed with the dust generated, giving a dust radioactivity concentration of 4×10^{-6} $\mu\text{Ci/g}$, and 4×10^{-8} $\mu\text{Ci/m}^3$ become airborne. Assuming that it takes one minute to drill a hole, with an airflow of $28 \text{ m}^3/\text{min}$ ($1000 \text{ ft}^3/\text{min}$), 1.1×10^{-6} μCi reach the contamination control envelope HEPA filters per hole drilled.

If about 8300 holes are required, drilled at 0.3 m (1 ft) centers in the steam generator enclosure, then a total of 3.3 μCi reach the contamination control envelope HEPA filters for the entire operation. Assuming a filter efficiency of 5×10^{-4} means that 1.7×10^{-3} μCi are released to the Reactor Containment Building. If leakage accounts for 0.1% of the airborne activity generated, then an additional 3.3×10^{-3} μCi are released. The total released to the Reactor Containment Building from the contamination control envelope HEPA filter plus leakage is 5×10^{-3} μCi . The total release to the atmosphere through the building HEPA filters is 2×10^{-6} μCi .

After the holes are drilled, the rock splitters are inserted and the concrete is scarfed. The concrete fractures and spalls in large pieces, with only a small amount of dust generated. Thus, the airborne release is insignificant when compared to the drilling operation.

The release from explosive removal of the surface concrete is calculated for a maximum air concentration of 100 mg/m^3 .⁽³⁾ Using the same dust radioactivity concentration calculated for drilling of 4×10^{-6} $\mu\text{Ci/g}$, if 500 g (1.1 lb) of dust containing 2.0×10^{-3} μCi are released per explosion, and about 8300 explosions are required, and 17 μCi are made airborne by the operation. A total of 8.3×10^{-3} μCi are released through the contamination control envelope HEPA filter. If leakage is again assumed to be 0.1% of the airborne activity, and additional 1.7×10^{-2} μCi are made airborne, for a total release to the containment of 2.0×10^{-2} μCi .

⁽³⁾J. M. Selby, et al., Considerations in the Assessment of the Consequences of Effluents from Mixed Oxide Fuel Fabrication Plants. BNWL-1697, Rev. 1, pp. 78-79, Battelle, Pacific Northwest Laboratories, Richland, WA, June 1975.

Electric or pneumatic jackhammers are used on floor surfaces. The quantity of suspendible particles generated by this technique is not directly known. Since the point of contact between the jackhammer and the concrete is well defined, an effective means of controlling airborne material such as a waterspray or cover mat could be employed. Therefore, the airborne release is less than for the other reference techniques discussed here, and it is not considered further in this analysis.

J.1.1.10 Defueling and Disabling the Reactor

Defueling the reactor is part of normal plant operations which are not specific only to decommissioning. As such, the safety of this operation is evaluated and included in the final safety analysis report (FSAR) required before the operation of the reactor. The FSAR study is more comprehensive and covers fuel handling in more detail than the scope of this study allows. Thus it is suggested that the FSAR should be consulted for a detailed analysis.

Disabling the reactor means making the reactor inoperable as specified in Regulatory Guide 1.86.⁽¹⁾ The reactor is disabled by taking the control rod system out of service.

J.1.1.11 Interfacility Transport

The observations made in the earlier section entitled "Waste Handling" are applicable here. In addition, loose suspendible material (concrete rubble) is contained prior to transport. Thus, it is anticipated that the contribution of the transport of contaminated materials to the total routine operations airborne release is not significant, compared to other release mechanisms.

J.1.2 Postulated Accidental Airborne Releases During Immediate Dismantlement

During Immediate Dismantlement, situations may arise that lead to accidental airborne releases. The postulated accidental releases for Immediate Dismantlement appear in Table J.1-2. Both the tabulated releases and accident

⁽¹⁾ U.S. NRC, Termination of Operating Licenses for Nuclear Reactors. Regulatory Guide 1.86, June 1974.

scenarios are presented in the following discussion in order of decreasing magnitude of atmospheric release.

J.1.2.1 Operational Accidents

Accidental airborne releases from Immediate Dismantlement operations are the result of events such as equipment failure, human error, and service conditions. The following analysis considers the operational accidents listed in Table J.1-2.

Accidents Involving Fuel

The release of radioactivity in postulated accidents involving the spent fuel storage pool, shipping casks and refueling are discussed extensively in the Final Safety Analysis Report.⁽¹⁷⁾ The Reactor Safety Study⁽¹⁸⁾ also considers operational fuel handling accidents. Several operations are proceeding during the time that fuel is stored in the fuel storage pool awaiting offsite shipment. Most of these operations take place in the Reactor Containment Building and are effectively blocked from contact with the spent fuel pool. The electropolishing decontamination station located in the new fuel storage area, adjacent to the spent fuel pool is estimated to give the highest probability for an operational accidental airborne release involving spent fuel. The accident with the maximum postulated airborne release is one that takes place 96 hours after the reactor is shut down, before the radioactivity in the fuel assemblies has decayed significantly. Since the electropolishing decontamination station is scheduled to begin operation two months after completion of defueling, an accident involving the decontamination station would involve lower radioactivity levels in the assemblies, and consequently smaller releases than the maximum postulated accident considered in the FSAR. Therefore, the potential airborne releases of radioactivity

(17) TROJAN Final Safety Analysis Report. Vol 8, Amendment 13, pp. 15.5-40, 15.5-51, Portland General Electric Co., Portland, OR, May 1974.

(18) U.S. Nuclear Regulatory Commission, Reactor Safety Study; an Assessment of Risks in U.S. Commercial Nuclear Power Plants. WASH-1400, NUREG-75/014, October 1975.

involving fuel handling as postulated on page I 103/104 of the Reactor Safety Study⁽¹⁸⁾ are considered to maximize the atmospheric releases for fuel handling accidents during decommissioning. These releases are tabulated in Tables J.1-10 and J.1-11. Since these operations are considered in these references, and are not unique to or amplified by decommissioning, they are not considered further in this study.

TABLE J.1-10. Atmospheric Releases of Radionuclides from Loss of Spent Fuel Cooling⁽¹⁸⁾

<u>Chemical Group or Element</u>	<u>Ci Released</u>
Noble Gases	4.56×10^7
Halogens	5.25×10^5
Alkali Metals	3.13×10^5
Alkaline Earth	7.86×10^4
Tellurium, Antimony	3.12×10^5
Noble Metals	1.42×10^5
Actinides and Lanthanides	8.97×10^4

TABLE J.1-11. Atmospheric Releases of Radioactivity from Fuel Handling Accidents

<u>Accident</u>	<u>Release in Curies</u>		
	<u>Noble Gases</u>	<u>Halogens</u>	<u>Alkali Metals</u>
Dropping Heavy Item Into SFP	1.37×10^6	3.41	-
Shipping Cask Accident	1.47×10^2	-	2.77×10^1

Accident During Offsite Transportation of Radioactive Materials

The materials to be shipped are classified as combustible or noncombustible, activated or contaminated, solids.

⁽¹⁸⁾ U.S. NRC, Reactor Safety Study: An Assessment of Risks in U.S. Commercial Nuclear Power Plants. WASH-1400, NUREG-75/014, October 1975.

Noncombustibles include the activated reactor vessel and internals, activated concrete, contaminated concrete, and contaminated piping and equipment. Combustible solids, include ion exchange resins (solidified with cement or ureaformaldehyde), filter cartridges, HEPA filters, miscellaneous cleaning materials, and protective clothing. All evaporator bottoms are assumed to be fixed in a solid matrix during decommissioning.

Shielded disposable containers are constructed to ship and bury segments of the reactor vessel and vessel internals, as well as concrete from the biological shield. Steel containers are fabricated for some segments. The bulk of the material is packaged in standard glass fiber-lined, reinforced plywood boxes (Figure F.2-3). Some equipment sections from non-radiation areas in the PWR are sealed with welded closures and require no further containment, however much of the material is doubly packaged. All of the material is assumed to be shipped by truck to the disposal site in this study.

Trucks can become involved in two types of accident situations: with or without fire. The nonfire accident has the lowest airborne release potential. Much of the radioactive inventory is on stainless steel surfaces or in coarser concrete rubble--forms that are not readily entrained by aerodynamic forces. If the concrete rubble is directly injected into the air, some fine particles are made airborne.

A fire situation gives a greater possibility of airborne release. Since trucks are involved, there is the possibility of a large-scale gasoline fire in the event of an accident. If the waste containers rupture, the radioactive inventory is subject to forces which could lead to an airborne release.

The spent resins, with a total of about 42,000 Ci, are removed in 20 shipments. This represents about 2100 Ci per shipment, and is one of the higher radionuclide inventories anticipated to be shipped in each load. The activated materials represent about 9300 Ci per shipment, but are in a matrix that is highly unlikely to become airborne. The filter cartridges have about 800 Ci radioactivity per shipment. Twenty-two containers of solidified evaporator bottom liquids are estimated at surface radiation levels of 50 R/hr

containing about 80 Ci per shipment, and combustible waste which is estimated to have about 6 Ci per shipment. Therefore, an accident involving the burning of solidified ion exchange resins is believed to have the highest potential airborne radioactive release.

Concrete and ureaformaldehyde are the suggested solidifying agents. Concrete is noncombustible even in a gasoline fire. Ureaformaldehyde is difficult to ignite, and it swells and cracks as it burns.⁽¹⁹⁾ It is anticipated to burn in a fire of the magnitude of a large scale gasoline fire, with the combustible ion exchange resin contributing to the fire.

In the postulated accident, the shipping container is breached in the impact resulting from the collision, and the heat of the fire causes the inner containment to rupture. All of the shipment becomes involved in a fire that burns to the point of self-extinguishment. The fractional release for the entire shipment during this type of event is estimated to be 1.5×10^{-3} .⁽²⁰⁾ All of the shipment is involved in the fire, so 3.15 Ci are estimated to be released to the atmosphere.

Explosion of LPG Leaked from Front End Loader

It is inconceivable that the large quantities of concrete rubble will be removed by hand operations. The mechanical technique used in this study for loading is a front-end loader. It is anticipated that by the time these procedures are put into practice, an alternative technique with a safer "fuel" supply will be available and used instead of the front-end loader. Currently, liquified propane gas is used as the fuel at a gauge pressure of 689.4 kPa (100 psig) at 21°C (70°F).⁽²¹⁾ Its specific volume is 542 cc/g (8.7 ft³/lb)⁽²¹⁾ and its limits of flammability in air are 2.1% to 9.5%.⁽²²⁾

(19) C. J. Hilado, Flammability Handbook for Plastics. Technomic Publishing Co., Inc., p. 32, 41, Stamford, CT, 1969.

(20) J. Mishima, "Potential Aerosol Generation Mechanisms from Damaged Packages," from Transport Packaging of Radioactive Materials, IAEA-SR-10/15, Vienna, pp. 29-37, 1976.

(21) Linde Specialty Gas Catalog. Union Carbide Corporation, Linde Division, New York, NY, 1974.

(22) M. G. Zabetakis, Flammability Characteristics of Combustible Gases and Vapors. Bulletin 627, U.S. Bureau of Mines, 1965.

Some maximum pressures and rates of pressure rise for 3 propane-air mixtures in this range are:⁽²³⁾

3%--510 kPa (74psi), 8,618 kPa/sec (12psi/sec)

5%--661 kPa (96 psi), 17,235 kPa/sec (2500 psi/sec)

7%--331 kPa (48 psi), 241 kPa/sec (35 psi/sec)

Much of the contaminated concrete rubble is generated in the containment, so this is the postulated location of the loading. The volume of the containment is $5.66 \times 10^4 \text{ m}^3$ ($2 \times 10^6 \text{ ft}^3$) and it would require 1189 m^3 ($4.2 \times 10^4 \text{ ft}^3$) and 2186 kg (4827 lb) of propane to produce a flammable mixture. This exceeds the fuel holding capacity of the front end loader specified in this study. If 4.53 kg (10 lb) of propane are vaporized, the gas would produce a flammable mixture in 113 m^3 (4000 ft^3), or approximately 0.2% of the containment. An explosion of the mixture produces a maximum pressure less than 241 kPa (35 psi), which would cause no structural damage nor result in impairment of filter capacity.⁽²⁴⁾

The mixture generated could be drawn into the purge exhaust system ventilating the containment. Deflagration of the air-propane mixture could generate pressure in the ductwork and might cause filter failure. A potential failure mechanism is the failure by the high temperature generated by the burning of the propane-air mixture on the glass fiber filter media. Loss of filters causes simultaneous loss of the blowers due to increased air velocity and pressure, plus potential damage from debris. Loss of the blowers tends to aerodynamically isolate the area and releases to the atmosphere. Remedial measures such as sealing off outlets would also limit the release.

Airborne contamination could be released instantaneously or over a relatively long time period as a consequence of such an occurrence. Contamination shaken or blown from its source in the path of the deflagration is instantaneously

⁽²³⁾G. H. Tryon, ed., Fire Protection Handbook. National Fire Protection Association, Boston, MA, 1969.

⁽²⁴⁾C. A. Burchsted and A. B. Fuller, Design, Construction and Testing of High Efficiency Air Filtration Systems for Nuclear Applications. ORNL-NSIC-65, Nuclear Safety Information Center, Oak Ridge, TN, 1970.

available while radionuclides entrained by aerodynamic phenomena in the area are released continuously after the explosion until the facility is isolated (blowers stopped and outlets sealed). Areas potentially containing significant quantities of radioactivity are the ducts and filters. The information required to estimate the contamination on the duct surface is not available, but it is postulated to be equal to the amount of radioactivity collected on the filters.

Filters downstream of an explosion have an inventory of radioactivity. If they are replaced before loading begins, the inventory is limited to the dust generating operations of the concrete removal. The mass of material deposited on the filters without causing serious operational problems, such as pressure drops exceeding about 10 cm (4 in.) water gauge at rated flows, vary considerably with filter construction and the size of the deposited material.^(24,25) A generous estimate of deposited material assumed for this study is 2.3 kg (5 lb).

If the prefilters and filters in the purge exhaust filter banks (50 filters per bank) rupture simultaneously, 2.5×10^4 μCi could be released to the atmosphere. If an equal amount of the radioactivity is released from the duct by this explosion, the total atmospheric release is estimated to be 5.0×10^4 μCi . This is calculated to be more than the routine release estimated for bioshield decommissioning operations. The total release is 3.6×10^3 μCi for a postulated accident just before the end of the concrete bioshield operation.

Explosion of Oxyacetylene During Segmentation of the Reactor Pressure Vessel

The reactor pressure vessel is cut into segments using an oxyacetylene torch. Acetylene is a gas having a flammability range of 2.5 to 80 volume %

(24) C. A. Burchsted and A. B. Fuller, Design, Construction and Testing of High Efficiency Air Filtration Systems for Nuclear Applications. ORNL-NSIC-65, Nuclear Safety Information Center, Oak Ridge, RN, 1970.

(25) H. A. Lee, Guide to Fire Protection in Caves, Canyons and Hot Cells. ARH-3020, Atlantic Richfield Hanford Company, Richland, WA, July 1974.

mixed in air.^(26,27) Violent explosions can occur when acetylene and oxygen are incorrectly mixed. The degree of explosive violence depends on how closely the gas mixture approximates the ratio for complete combustion. Some maximum pressures and pressure rises from 3 acetylene-air mixtures in the explosive range are:⁽²³⁾

5%--544 kPa (79 psi), 17,235 kPa/sec (2,500 psi/sec)

13%--1034 kPa (150 psi), 8,273 kPa/sec (1,200 psi/sec)

25%--689 kPa (100 psi), 9,652 kPa/sec (1,400 psi/sec)

Since the segmentation takes place in the reactor pressure vessel cavity, this is a likely location for an explosion. Oxyacetylene is used to segment steam generators and pressurizers so these are also candidate explosion sites. The reactor vessel wall segmentation is selected to illustrate this event because of the higher radioactivity involved. It is assumed that the acetylene is stored in an area that does not contain radioactivity, so there is no radioactive release potential from a postulated storage accident.

Oxyacetylene explosions occur from a variety of causes such as reverse flow,⁽²⁸⁾ obstructions in the nozzle, and flashback (a flare going back up one of the hoses).⁽²⁹⁾ The dust from this explosion is postulated to have an airborne dust concentration in the cavity of 100 mg/m³.⁽³⁾ The radioactivity of the airborne dust is calculated to be about 3500 μ Ci.

(26) Fire Protection Guide on Hazardous Materials. Sixth Edition, National Fire Protection Association, Boston, MA, 1975.

(27) R. C. Weast, ed., Handbook of Chemistry and Physics. 54th Edition, CRC Press, Cleveland, OH, 1973.

(23) G. H. Tryon, ed., Fire Protection Handbook. National Fire Protection Association, Boston, MA, 1969.

(28) E. S. Young, "Preventing Reverse Flow Explosions in Gas Cutting Equipment." Plant Engineering, p. 505, December 14, 1972.

(29) R. Cecil Smart, The Technology of Industrial Fire and Explosion Hazards. Chapman and Hall, LTD., London, pp. 41-43, and pp. 128-132, 1947.

(3) J. M. Selby, et al., Considerations in the Assessment of the Consequences of Effluents from Mixed Oxide Fuel Fabrication Plants. BNWL-1697, Rev. 1, pp. 78-79, Battelle, Pacific Northwest Laboratories, Richland, WA, June 1975.

The reactor pressure vessel segmentation takes place in the reactor cavity which has a design pressure of 1034 kPa (150 psi)⁽³⁰⁾ and has a 1.5 safety design factor. This means that no damage to the reactor cavity results from an acetylene explosion with the maximum pressure of 1034 kPa (150 psi). This pressure is sufficient to cause failure of the HEPA filter⁽²⁴⁾ in the contamination control envelope. The mass inventory of the filter is 2.27 kg (5 lb) (see LPG explosion). The total radioactivity inventory level (assuming a pair of filters in the contamination control envelope) is 0.33 Ci.

The inside surface of the contamination control envelope is contaminated with radioactive dust that is made airborne by heat, mechanical, and aerodynamic disturbances created by the explosion. It is assumed that the accident occurs with the maximum radioactivity deposited, just before completion of the cutting operations. The total material generated in the operation is 2.7×10^6 g. A fraction of this is deposited on the inside surface of the contamination control envelope. Assuming that 0.2% of total material generated is deposited, 5.4×10^3 g with 72 μ Ci/g or 3.9×10^5 μ Ci total are deposited on the filters. Thus the total 0.72 Ci release consists of an airborne activity of 3500 μ Ci, 0.33 Ci deposited on the filters, and 3.9×10^5 μ Ci deposited on the contamination control envelope.

The transmitted pressure, P_t , is given in terms of shock wave Mach Number as:⁽³¹⁾

$$\frac{P_t}{P_1} = \frac{2 \gamma m t^2 - (\gamma - 1)}{(\gamma + 1)} \quad (J.5)$$

(30) TROJAN Final Safety Analysis Report. Vol. 2, Amendment 9, Figure 3.8-16, Portland General Electric Co., Portland, OR, February 1974.

(24) C. A. Burchsted and A. B. Fuller, Design, Construction and Testing of High Efficiency Air Filtration Systems for Nuclear Applications. ORNL-NSIC-65, Nuclear Safety Information Center, Oak Ridge, TN, 1970.

(31) M. R. Busby, J. E. Kahn, and J. P. Belk, "Effects of Explosion Generated Shock Waves in Ducts." Proceedings of the 14th ERDA Air Cleaning Conference. CONF-760822, pp. 210-219, 1976.

where:

- γ = the specific heat ratio, C_p/C_v
- M_t = the transmitted shock wave Mach Number
- P_1 = the pressure in the gas downstream
- P_t = the transmitted static pressure.

Using 1034 kPa (147 psi) as the transmitted static pressure, and 2.5 as the Mach Number, the downstream pressure is calculated to be 145 kPa (21 psi). Since the Reactor Containment Building is designed to hold a pressure of 414 kPa (about 59 psi), no other damage should occur. The distance between the top of the reactor pressure vessel cavity and the purge filter is about 30 m (100 ft), which should allow enough room for the dissipation of the shock wave, thus no filter damage is expected.

Explosion and or Fire of Ion Exchange Resin

Both anion and cation exchange resins have been involved in explosion incidents.⁽³²⁾ The lowest ignition temperature found for the nitrate form is 135°C (275°F). An ion exchange resin can self-ignite, depending on factors such as the resin form, the heating ratio, the loading, the volume, and the aging process.⁽³³⁾ Reported accidents concerned resins involved with concentrated HNO_3 , an oxidizing agent, but the mechanisms of the initiated explosion have not yet been determined.

Phosphoric acid is neither flammable nor explosive mixed in air but its chemical reactions can generate heat.⁽³⁴⁾ The reactions of phosphoric acid and ion exchange resins have not been investigated. There is a low probability that sufficient heat could be generated within the resins to initiate an explosion.

(32) J. Barghusen and A. A. Jonke, "Incident in Plutonium Processing at Hanford." Reactor Fuel Processing. 7:297-304, 1964.

(33) J. T. Fullam, Thermal Stability of Nitrate Form Anion Resins in a Vented System. RL-SEP-823 (TID-4500, 46th Edition), Hanford Atomic Products Operation, Richland, WA, December 1965.

(34) Battelle-Northwest Safety Guides. BNWL-MA-43, Battelle, Pacific Northwest Laboratories, Richland, WA, June 1967.

The radionuclides in the electropolishing solution are removed to the ion exchange resin. Electropolishing all the primary piping gives a resin inventory of 22.4 Ci (16.2 Ci from RCS piping and 6.2 Ci from other piping).

The explosive release of an ion exchange column in a nuclear waste facility resulted in an estimated release in the building of about 1 μCi .⁽³⁵⁾ This column was loaded with 175 g of ^{241}Am with a specific activity of 3.45 Ci/g. The fraction of this inventory released to the atmosphere was 1.7×10^{-6} . Using the same fractional release in this explosion, 38.1 μCi are estimated to be released to the atmosphere with a total of $7.6 \times 10^4 \mu\text{Ci}$ released in the building.

Gross Leak During In Situ Decontamination

A massive failure of piping is a low probability event. There have been no such accidents in 30 years of experience with nuclear systems. Massive failures are unknown in high-pressure central steam boilers which provide considerably more experience than is available in the nuclear industry.⁽³⁶⁾

Pipe failures can result from: 1) design, 2) materials, 3) base metal defects, 4) fabrication, or 5) service conditions.⁽³⁷⁾ The in situ chemical decontamination takes place 40 years after the piping was put in service and introduces a chemical solution to that piping that was not anticipated during its design. It is unlikely that a system that held about $1.5 \times 10^4 \text{ kPa}$ (2200 psi) would leak at a pressure of 2758 kPa (400 psi), the RCS pressure

(35) Chemical Explosion of an Ion Exchange Resin Column and Resulting Americium Contamination of Personnel in the 242-Z Building, August 30, 1976. Energy Research and Development Administration, Richland, WA, October 19, 1976.

(36) M. Eisenbud, "The Primary Air Pollutants - Radioactive, Their Occurrence Sources, and Effects." Air Pollution Third Edition, Volume 1, Air Pollutants, Their Transformation and Transport. pp. 197-231, A. C. Stern, Editor, Academic Press, New York, NY, 1976.

(37) H. Thielsch, "Why High Temperature Piping Fails." Pressure Vessels and Piping Design and Analysis, Volume 4, Quality Assurance - Applications, Components. H. H. Waite, editor, The American Society of Mechanical Engineers, pp. 350-375, New York, NY, 1976.

during chemical decontamination. The decontamination solution should not cause excessive corrosion damage to the piping. Tests of decontamination solutions used in this study have shown a uniformly slow corrosion rate for stainless steel of 0.05 to 0.5 mg/cm²/hr.⁽³⁸⁾ Using 0.5 mg/cm²/hr and the density of 8.027 g/cm³ for stainless steel, the corrosion depth rate is calculated to be 6×10^{-5} cm/hr. If this rate continued for the approximately 2 weeks that may be required, the corrosion is 0.02 cm (assuming continuous circulation). This value is considerably below the thickness of the 5 cm (2 in.) pipe with a wall thickness of 0.87 cm (0.344 in.). Assuming the system does retain its integrity, a gross leak above the postulated leak rate during routine reactor operations would not be likely to occur from the corrosion mechanism.

Accidental leakage could also occur as a result of human errors such as poor judgment where exposure time is controlled to minimize localized corrosive attack, lack of adequate instructions (e.g., incorrect mixing of solutions), or lack of proper control (e.g., inadequate temperature control).

The gross leak of 10 times the magnitude of the routine in situ decontamination leak is a low probability accident but is considered to be a conservative basis for this safety assessment. The leak during routine operations was estimated to give a maximum airborne release range of from 0.45 to 136 μ Ci/min depending on leak type. Therefore a gross leak of 10 times that magnitude gives a range of 4.5 to 136 μ Ci/min. If 30 minutes elapsed before the leak is discovered and stopped, 4.1×10^4 μ Ci and 136 μ Ci are released to the containment.

Segmentation of RCS Piping with Unremoved Contamination

The contamination in the RCS piping is not evenly distributed within the piping. Circulating corrosion products deposit throughout the primary system and can preferentially deposit in a low velocity flow area or heat transfer surface.⁽³⁹⁾ Such a pocket could remain unremoved during the decontamination, again because of low velocity flow.

⁽³⁸⁾ J. A. Ayres, Decontamination of Nuclear Reactors and Equipment. Ronald Press Co., New York, NY, pp. 186-189, 1970.

⁽³⁹⁾ A. J. Kennedy, "PWR Crud Characterization and Transport Modeling." From Proceedings of the System Contamination Workshop. Atlanta, GA, pp. 95-111, March 15-17, 1976.

The maximum radioactivity level before chemical decontamination was estimated to be $86 \mu\text{Ci}/\text{cm}^2$. The surface area of the cut is 238 cm^2 (37 in.^2), $2.1 \times 10^4 \mu\text{Ci}$ and $5.7 \times 10^3 \text{ g}$ of stainless steel are vaporized by the arc saw during this segmentation. If this contamination is not removed, it is released to the Reactor Containment Building since no contamination control envelope is assumed to be used for this operation. The total release to the atmosphere is estimated to be $10.5 \mu\text{Ci}$.

Loss of Contamination Control Envelope During Oxyacetylene Cutting of the Reactor Pressure Vessel

A hazardous aspect of oxyacetylene cutting is the projection of small globules of molten metal to a distance of up to 9.1 m (30 ft) from the source.⁽²⁹⁾ Since this operation takes place within a contamination control envelope, it is postulated that a globule of molten metal pieces the envelope. The routine radioactivity generation rate during cutting is $1150 \mu\text{Ci}/\text{min}$ from Table J.1-1. Since this is a remotely controlled operation, the release is not immediately apparent, and is assumed to continue for about five minutes. Thus $4.7 \times 10^3 \mu\text{Ci}$ are released in the Reactor Containment Building before operations are halted. It is also assumed that some of the radioactivity previously deposited on the inside of the contamination control envelope is available for release. If the additional release fraction is assumed to be equal to the release fraction from a contaminated waste fire, or 1.5×10^{-4} ⁽⁴⁰⁾, an additional $0.7 \mu\text{Ci}$ are released to the Reactor Containment Building. The total release in the Reactor Containment Building is therefore estimated to be $4.7 \times 10^3 \mu\text{Ci}$, and $2.3 \mu\text{Ci}$ are released to the atmosphere through the building HEPA filters.

⁽²⁹⁾ R. Cecil Smart, The Technology of Industrial Fire and Explosion Hazards. Chapman and Hall, Ltd., London, pp. 41-43 and pp. 128-139, 1947.

⁽⁴⁰⁾ J. Mishima and L. C. Schwendiman, Fractional Airborne Release of Uranium (Representing Plutonium) During the Burning of Contaminated Waste. BNWL-1730, Battelle, Pacific Northwest Laboratories, Richland, WA, April 1973.

Pressure Surge Damage to Filters During Blasting of Activated Concrete Bioshield

The ventilation enclosures are designed to accommodate air pressure from blasting. At the Elk River Reactor,⁽⁴¹⁾ some signs of pressure effect occurred during blasting. There was partial collapsing of a prefilter installed in a local contamination control system.

Using the estimated air concentration from blasting of 100 mg/m^3 ⁽³⁾, 5.34 μCi are estimated to be released to the containment. The mass of collected dust released from the filters is assumed to be similar to the previously analyzed accident for the Explosion of LPG leaked from a front end loader. For two filters containing 2.3 kg (5 lb) of contaminated dust, the calculated maximum release is 499 μCi . Thus, a total of 504 μCi is estimated to be released to the Reactor Containment Building.

Loss of Integrity of Portable Filtered Ventilation Enclosure During Segmentation of the Steam Generators

The portable filtered ventilation enclosure is used to confine and collect particulate material produced by the cutting processes. It is also designated for use during operation of the rock splitter, and it is illustrated in Figure F.3-5. The ventilation is through a large squirrel-cage exhaust blower coupled with a HEPA filter preceded by glass fiber roughing filter. This ventilation unit is mounted on a cart and is coupled to the work station with a flexible hose. Another glass fiber roughing filter is installed in the inlet of the enclosure. The enclosure minimizes the potential airborne contamination release.

A loss of the roughing filters and the HEPA filter simultaneously seems less likely and would result in lower releases than a breach in the integrity of the flexible duct. This could be by mechanical means such as abrasion. In This study, a small hole is assumed to develop from this mechanism or by other

⁽⁴¹⁾ J. Nemec, Demolition of Radioactive and Contaminated Concrete Structures by Use of Explosives. United Power Association, p. 9, April 15, 1974.

⁽³⁾ J. M. Selby, et al., Considerations in the Assessment of the Consequences of Effluents from Mixed Oxide Fuel Fabrication Plants. BNWL-1697, Rev. 1, pp. 78-79, Battelle, Pacific Northwest Laboratories, Richland, WA, June 1975.

physical means. Damage to the filter is assumed to go unnoticed. Damage to the duct is more substantial, and is likely to be immediately apparent. Each breach would lead to particulate releases of different magnitudes. In the event that a substantial breach occurs, the exhaust fan continues to draw air in forcing air to diffuse through the breach. With an assumed airflow through the HEPA filter of $28 \text{ m}^3/\text{min}$ ($1000 \text{ ft}^3/\text{min}$), $55 \text{ } \mu\text{Ci}/\text{min}$ are generated during the segmenting of the steam generator. Ten percent of this amount is assumed to leak to the Reactor Containment Building through the breach, or a rate of about $5.5 \text{ } \mu\text{Ci}/\text{min}$. It is further assumed that the breach is immediately apparent and that segmenting is terminated promptly. If the air flow continues for about 10 minutes, $60 \text{ } \mu\text{Ci}$ is released to the Reactor Containment Building, and $3 \times 10^{-2} \text{ } \mu\text{Ci}$ are released to the atmosphere through the building HEPA filters.

Fire Involving Contaminated Clothing or Combustible Waste

Absorbent materials, such as rags or paper wipes, are used for a variety of purposes during the PWR dismantlement and are discarded after use. Materials such as rags and mops that have come into contact with contaminated surfaces, hold small quantities of radionuclides. Protective clothing such as coveralls, caps, hoods, or shoe covers become contaminated through use. Plastics are generally considered combustible, although many kinds do not burn readily or are self-extinguishing.

It is postulated that combustible contaminated waste containing low levels of radioactivity ignites, and burns, in a working area. Large amounts of these wastes are not likely to accumulate in one place, so it is assumed one cubic meter containing $8 \times 10^4 \text{ } \mu\text{Ci}$ is involved in the fire. The maximum fractional airborne release measured during the burning of contaminated waste under similar conditions was 1.5×10^{-4} .⁽⁴⁰⁾ Thus it is estimated that up to $12 \text{ } \mu\text{Ci}$ could be released to the ventilation system from this fire. A total of $6 \times 10^{-3} \text{ } \mu\text{Ci}$ are released to the atmosphere through the building HEPA filter.

(40) J. Mishima and L. C. Schwendiman, Fractional Airborne Release of Uranium (Representing Plutonium) During the Burning of Contaminated Waste. BNWL-1730, Battelle, Pacific Northwest Laboratories, Richland, WA, April 1973.

Loss of Blasting Mat During Removal of Activated Concrete

Blasting for removal of contaminated concrete surfaces is conducted in the reactor cavity. If the protective blasting mat is lost during blasting, the confinement barriers could possibly be breached by projectiles of debris. The consequences of explosion accidents are generally limited by the material that is made airborne. A reasonable upper mass limit of 100 mg/m^3 is used to estimate the airborne radioactivity.⁽¹³⁾ About 48.5 g containing $0.11 \text{ } \mu\text{Ci/g}$ or $5 \text{ } \mu\text{Ci}$ could be airborne in the volume of the Reactor Containment Building. These values are based on the assumption that the fine fog water sprays are inoperative. If the sprays are operative, the release would be greatly reduced.

Detonation of Unused Explosives in the Reactor Cavity

An individual charge of the explosive used to scarf the bioshield may not be detonated on command. This failure usually is detected by a qualified explosive engineer by visual inspection of the surface being scarfed. If an errant charge is detonated during the inspection when the water spray is turned off and the blasting mat and contamination control envelope are not in place, a temporary high dust loading in the cavity could result. Assuming that the entire volume of the reactor pressure vessel is filled with dust to 100 mg/m^3 , 48.5 g of bioshield concrete dust containing $0.11 \text{ } \mu\text{Ci/g}$ are made airborne as a result of this event. The total airborne radioactivity in the reactor cavity is about $5 \text{ } \mu\text{Ci}$. There would be additional nonradiation hazards to employees if an unexploded charge detonated when workers were in the area.

Temporary Loss of Local Airborne Contamination Control During Blasting

A contamination control envelope is required in the Reactor Containment Building during the explosive removal of the biological shield contaminated concrete. Loss of the fine fog spray, airborne contamination control, increases the amount of dust made airborne.

If such an accident occurred, approximately 48.5 g concrete dust containing $5.335 \text{ } \mu\text{Ci}$ activity could be made airborne in the reactor cavity. However,

⁽¹³⁾ D. McConnon and J. F. Nemec, Experiences in Decontamination/Decommissioning of the Elk River Reactor. April 1, 1974, presented at the Second AEC Environmental Conference, Albuquerque, NM, April 1974.

loss of the fine water spray does not mean loss of the contamination control envelope, and therefore, this release is mitigated by the HEPA filter. From the HEPA filter a total of 2.7×10^{-3} μCi is passed into the Reactor Containment Building.

Temporary Loss of Local Airborne Containment Control During Jackhammer Scarfing of Concrete Surfaces

Since this is a manual operation, loss of the airborne control is readily apparent to the operator, and the operation is suspended before a significant release occurs. Such releases would be much smaller than for the preceeding postulated accidents.

Temporary Loss of Services

Since most of the operations are manual, loss of services such as water, power or airflow would be immediately evident and would be compensated for by the workers. Thus, such a condition would constitute a lesser hazard for airborne releases than other postulated accidents.

Dropping of Contaminated Concrete Rubble

If a container of contaminated concrete rubble is accidentally spilled, the accumulated material would fall to the floor and cause fine particles that are adhering to rubble to become suspended in the air. The quantity of such material is believed to be small since most of the readily suspendible particles are removed during routine operations. Thus, it is anticipated that this accident would result in lower airborne contamination than other accidents that were evaluated.

J.1.2.2 Natural Phenomena

The reference PWR has a final safety analysis report (FSAR) that was written before the start of plant operations. This FSAR evaluated natural phenomena and their effect on plant safety. The following general design criterion were met: "Structures, systems and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunamis, and seiches without loss

of capability to perform safety functions."⁽⁴²⁾ It is assumed that this integrity was preserved during the operating life of the plant. It is further assumed that the structural integrity is preserved during decommissioning as long as required for safety. It is felt that, while natural phenomena can cause severe damage to the reference PWR resulting in severe radiological public safety impacts, these are low probability events. Also, the impact of these events will be less than the impacts calculated in the design basis accidents for the operating PWR in the FSAR. For this reason, no detailed analysis of the impacts of these phenomena is attempted beyond the brief discussions that follow.

Floods

The FSAR considered runoff floods, surges, seiches, wave action, tsunamis, wind-generated wave action and combinations of these phenomena. For the FSAR, a probable maximum flood was proposed. The plant safety was assured to be safe from damage by such a flood or other potential flooding, so the site is assumed to be above maximum flood level.

Earthquakes

All systems and components designated seismic Category I were designed to exhibit no loss of functions for earthquake ground accelerations of 0.25 g. Category I structures at the PWR are: Auxiliary Building, Fuel Building, Intake Building, Reactor Containment Building, and Containment Internals.

The Reactor Containment and Auxiliary Buildings are separated by a 7.62 cm (3 in.) expansion joint (Figure A.2-11). This gap is adequate to prevent the two structures from coming in contact with one another during an earthquake.

The Fuel Building is structurally connected to the Auxiliary Building (Figure A.2-19). Expansion joint bellows at the fuel transfer tube (between the containment and fuel building) provide for relative movement between containment, containment internals, and the spent fuel pool. The bellows consider all loading conditions including the safe shutdown earthquake and hydraulic pressure.

⁽⁴²⁾ Trojan Final Safety Analysis Report. Vol. 2, Amendment 3, pp. 31-36, Portland General Electric Co., Portland, OR, September 1973.

A reasonable estimate of the probability of an earthquake that would occur and cause damage to the plant is 10^{-7} per year.⁽¹⁸⁾ At this level of probability, earthquake induced accidents are extremely unlikely, and are not further analyzed.

Tornadoes

All power reactors in the United States are now designed to withstand the effects of sizeable tornadoes. Significantly less than 1% of all tornadoes are expected to be as large as a design basis tornado. The probability of such a tornado striking a nuclear power plant is on the average less than 5×10^{-6} per year.⁽¹⁸⁾

Tests at Los Alamos showed that a 20 cm x 20 cm x 8 cm (8 in. x 8 in. x 3-1/16 in.) HEPA filter could withstand 9-second pressure pulse with the maximum pressure of 21 kPa (3 psi) and hold without visible damage or reduction in efficiency.⁽⁴³⁾ This pressure pulse is substantially greater than the pulse the filter would experience in the event of a tornado.

High Winds

The design wind velocity for all Category I and Category II structures is 168 km/hr (105 mi/hr) at 9.1 m (30 ft) above nominal ground elevation. Therefore, significant releases during dismantling due to high winds are not expected.

J.1.2.3 Aircraft Crashes

Each reactor site has its own potential risk from aircraft crashes based on its proximity to airports and air traffic lanes. However, current information leads to the conclusion that the aircraft risk is low for most reactor sites.⁽⁴⁴⁾ Thus, the risk is not escalated by dismantlement operations. The aircraft crash probability for most plants is very low at 10^{-6} to 10^{-8} per year, and no further analysis is attempted.

⁽¹⁸⁾ U.S. Nuclear Regulatory Commission, Reactor Safety Study; an Assessment of Risks in U.S. Commercial Nuclear Power Plants. WASH-1400, NUREG 75/014, October 1975.

⁽⁴³⁾ C. A. Burchsted, J. E. Kahn and A. B. Fuller, Nuclear Air Cleaning Handbook. ERDA-76-21, Oak Ridge National Laboratory, Oak Ridge, TN, 1976.

⁽⁴⁴⁾ I. B. Wall, "Probabilistic Assessment of Aircraft Risk for Nuclear Power Plants." Nuclear Safety. 15:(3):276-284, May-June 1974.

J.1.2.4 Man-Caused Events

Activities of man outside the plant border could intrude and cause accidental releases. A detailed analysis of the man-caused events is not within the scope of the study. This would require speculation and deliberate acts, which cannot truly be assessed.

J.2 AIRBORNE RELEASE SCENARIOS FOR PREPARATIONS FOR SAFE STORAGE

Cleaning the facility and immobilizing radioactivity is the major operation performed in preparation for the Safe Storage approach. Onsite retrievable storage of decommissioning wastes and in situ decontamination of the Reactor Coolant System are considered in this analysis. The assumptions made about the Initial Facility Status before the Immediate Dismantlement operations are valid for the operations of preparing for Safe Storage. The airborne releases are segregated into those taking place as the result of planned routine operations and accidents.

J.2.1 Postulated Airborne Releases From Routine Preparations for Safe Storage

The routine operations in preparing for Safe Storage may be categorized as those of decontaminating surfaces, draining and drying vessels, painting surfaces to reduce the mobility of unremoved contamination, and mechanical sealing. Unremoved contamination in various plant areas are determined in the field and the specific decommissioning action is based on the cost and safety effectiveness of immediate removal versus immobilization and deferred removal. Immobilization methods include multiple layers of waterproof paint or other adhesive coatings and the attachment of appropriate warning labels and signs. The airborne radioactive releases from these operations are in the form of liquid aerosols. Insignificant amounts of solid particles are also released in the limited mechanical decontamination operations.

The airborne releases are presented in Tables J.1-1 and J.1-2 in order of decreasing magnitude of atmospheric release. Since the airborne release from the continuing care period may extend over many years, it should not be compared with the releases taking place during a few weeks or months of

decommissioning operations. The releases during continuing care are therefore at the end of the release tabulation. Airborne releases for the vacuuming alternative and onsite retrievable storage have been considered and are included here.

J.2.1.1 Surface Cleaning

The surface cleaning operations listed for preparation for Safe Storage in Table J.1-1 are discussed in the following sections.

The Hand-Held Water Lance

In addition to the areas described in the dismantlement approach, the hand-held lance is used to decontaminate portions of the exterior of the primary coolant system. It is not used in areas that are identified as having high levels of surface radioactivity. Since these areas are not quantified, it is assumed that the entire exterior surface of the primary coolant system is decontaminated by this method. This method gives the highest airborne radioactive release, so this is a maximum estimate of the airborne release from decontamination. Using these guidelines, the total release for decontaminating this system is calculated to be $1.8 \times 10^3 \mu\text{Ci}$.

Sweeping Compounds

Surface contamination is removed from areas with low levels of surface radioactivity, which have a maximum dose rate of less than 100 mR/hr and surface contamination level of less than $1.5 \mu\text{Ci}/\text{m}^2$. About a third of the floor area, 2600 m^2 , of the Fuel, Auxiliary, and Control Buildings are assumed to be swept. Using the resuspension factor of $1.9 \times 10^{-2} \text{ m}^{-1}$ and assuming that the resuspended radioactivity is confined to the lower meter of the room, $2.9 \times 10^{-2} \mu\text{Ci}/\text{m}^3$ become airborne during the operation, for a total release of 75 μCi . The levels of radioactivity ($9 \mu\text{Ci}/\text{m}^2$) found in the Reactor Containment Building preclude sweeping that area.

Vacuuming

Some areas are vacuumed as an alternative to sweeping. The airborne release from vacuuming is calculated by assuming the same 2600 m^2 of floor area used in estimating the sweeping airborne release.

Since this is an alternative, the option with the larger calculated airborne release value is the assumed release for the area. Vacuuming removes the loose contamination on the floor, estimated as 50% of the surface inventory. The vacuum system is equipped with a HEPA filter on the exhaust to remove contaminated particulates. The estimated amount of contamination removed by this method is 2000 μCi . Assuming no leakage from the vacuum, the exhaust downstream of the HEPA filter releases 1 μCi during the entire operation.

Acid Soaked Sponges

Some areas with low contamination levels are manually cleaned with acid soaked sponges. There appears to be a low potential for routine airborne release using this technique. Some small liquid drops are generated during the wringing of solution out of the sponges, and during wiping. The level of contamination in the liquid is very low as limited by the restrictions on radiation levels for manual operations. Thus, the release is not considered significant.

J.2.1.2 Mechanical Decontamination

Mechanical decontamination methods are used to remove radioactive contamination from areas that contribute significantly to the radiation exposure of surveillance and maintenance personnel. Drilling, rock splitting, and jackhammering are some of the methods used. In this application, the area of use is well defined so releases are controlled by the use of local airborne control methods. Additionally, the operator wears a face mask and protective clothing to minimize his exposure. Thus, it is not anticipated that this technique results in either the airborne release of a significant amount of radioactivity or excess operator exposure.

J.2.1.3 Fixing Residual Contamination

Residual low level contamination that remains on surfaces outside of the isolated areas is immobilized by covering the surfaces with paint or other protective coatings. This operation minimizes future airborne contamination, and it is postulated that this operation does not release significant amounts of airborne contamination since no operations which radically disturb surface contamination occur.

J.2.1.4 Waste Handling

Most of the materials that are packaged for disposal are combustible, contaminated solids. They are used on a continuing basis during reactor operation so standard handling procedures that are in place will be used. The routine releases from these sources are not significant compared to other operations considered.

Similarly, radioactive liquids are generated during reactor operation. Standard methods are in place and are used for handling and disposing of liquid wastes generated during these operations. The maximum radionuclide inventory in liquid wastes generated during the use of the hand-held lance was previously estimated as 9.8 $\mu\text{Ci}/\ell$ from cleaning the reactor pressure vessel cavity and spent fuel pool. This inventory is considerably lower than the peak concentrations of fission and corrosion products in reactor coolant liquids that are treated during reactor operations.⁽⁴⁵⁾ The systems are adequate to handle them without increasing airborne releases.

Loading contaminated concrete rubble is the major contributor to airborne releases in the dismantlement operations. During the preparation for Safe Storage, significant quantities of concrete rubble are not generated, so this operation does not contribute to airborne releases.

J.2.1.5 Equipment Deactivation and Isolation

Deactivation and isolation techniques include closing installed valves, locking or removing handles, installing blank flanges, and disconnecting electric power and other utilities. Pipes that have contained contaminated materials are blanked where flanges are provided. No piping with major contamination is cut. The only piping that is cut is uncontaminated pipe essential to isolate a system.

The operation is mainly one of disconnecting, closing and sealing. It takes place after chemical decontamination has been completed. Thus, no significant airborne release occurs.

⁽⁴⁵⁾ TROJAN Final Safety Analysis Report, Vol. 6, Table 11-1-2, Portland General Electric Co., Portland, OR, May 1974.

J.2.1.6 Surveillance and Maintenance

Surveillance inspections are assumed to take place quarterly for a length of time that varies from a few years to perhaps as long as 100 years. A status review is suggested at about 50 years, or about ten half-lives of ^{60}Co decay. Release estimates are based on accumulated 50-year releases, since by then significant decay has occurred so that no further significant airborne release occurs from these operations.

Only low level radioactivity areas of the plant are involved in these inspections. Sweeping is assumed to remove 90% of the floor contamination in an area of about 2600 m^2 of the Fuel, Auxiliary, and Control Building floors. From Table J.1-4, the characteristic floor contamination level is $1.5\text{ }\mu\text{Ci/m}^2$, so after sweeping, the remaining surface contamination level is $0.15\text{ }\mu\text{Ci/m}^2$. The reactor containment building floor is also assumed to have 90% of the surface contamination removed by decontamination efforts. From Table J.1-4, the steam generator floor has $9\text{ }\mu\text{Ci/m}^2$ so the remaining level after decontamination is $0.9\text{ }\mu\text{Ci/m}^2$ in an area of 1200 m^2 . It is calculated that about $3\text{ }\mu\text{Ci}$ becomes airborne during each inspection of the facility.

Since the ventilation system is deactivated, radioactive material loss from the PWR buildings is the result of diffusion. This is a slow process and much of the contamination settles to the floor again. If 1% of the $3\text{ }\mu\text{Ci}$ made airborne for each inspection eventually reaches the HEPA filters, $1.5 \times 10^{-5}\text{ }\mu\text{Ci}$ reach the atmosphere per inspection. The total release to the atmosphere is $6 \times 10^{-5}\text{ }\mu\text{Ci/year}$, and a total of $3.8 \times 10^{-3}\text{ }\mu\text{Ci}$ are released for the 50 year period of continuing care. This estimate is a maximum since it does not include radioactive decay effects.

J.2.1.7 Onsite Retrievable Storage

Several facility areas are suggested for onsite storage of solidified wastes. These are the spent fuel pool, the Containment Building refueling cavity, and the reactor vessel interior.

After the reactor vessel is drained and dried, the core shroud provides about 33 m^3 (1165 ft^3) of storage space for containers of waste materials. The vessel is activated from neutron exposure and the vessel wall also has

surface contamination from the deposition of activated corrosion products and fission products. This surface radioactivity is estimated to be approximately $22.25 \mu\text{Ci}/\text{cm}^2$ (Table J.1-6). A portion of this inventory is dislodged by mechanical and aerodynamic forces during the operations required for storage of the containers. The interior surface area of the portion of the vessel suggested for storage is 45 m^2 , so the total surface radioactivity subject to resuspension forces is approximately 10 Ci. A fractional release of 2.9×10^{-3} ⁽²⁰⁾ is a maximum estimate of the material that may become airborne during the placement of shipping containers. Therefore, $2.9 \times 10^4 \mu\text{Ci}$ are released from the interior surface of the shroud during placement of the solidified wastes in this storage area.

The inventory of the refueling cavity and spent fuel pool after cleaning is $3 \times 10^3 \mu\text{Ci}$, so that area contributes 9 μCi to the total release.

J.2.1.8 In Situ Decontamination of the Primary System

The chemical decontamination of the primary system internal surfaces is assumed to generate the same airborne radionuclide releases as those calculated for this operation during Immediate Dismantlement in Section J.1.1.6.

J.2.2 Postulated Accidental Airborne Releases During Preparation for Safe Storage

The accident scenarios for airborne releases presented in the following discussion are listed in Table J.1-2. A comparison of the airborne releases and accidents between Immediate Dismantlement and preparing for Safe Storage operations is also presented in Table J.1-2.

J.2.2.1 Operational Accidents

Postulated accidents during normal safe storage operations are discussed below. The causes of such accidents include equipment failure, human error, and malfunctioning service conditions.

⁽²⁰⁾ J. Mishima, "Potential Aerosol Generation Mechanisms from Damaged Packages," from Transport Packaging of Radioactive Materials. IAEA-SR-10/15. Vienna, pp. 29-37, 1976.

Vacuum Bag Rupture

Metal shards picked up in vacuuming cause abrasion of the filter bag that could lead to a rupture. This could be accompanied by puncture of the vacuum cleaner. In this case, all of the collected material is assumed to be ejected into the air by the mechanical and aerodynamic forces of the $1.4 \text{ m}^3/\text{min}$ air flow. The material is assumed to be immediately made airborne by the action. If this accident occurs just before completion of the operation, the entire radioactivity inventory of about $200 \text{ } \mu\text{Ci}$ would be entrained in the building ventilation system. A total of $1 \text{ } \mu\text{Ci}$ would be released to the atmosphere.

Accidental Cutting of Contaminated Piping

Human error is assumed to lead to the accidental cutting of a piece of contaminated pipe. A large piece of pipe is not likely to be cut accidentally, but a mistake is postulated to be made on a pipe 25.4 cm (10 in.) or smaller. Three nominal pipe sizes, 5.1 cm , 15.2 cm , and 25.4 cm (2, 6, and 10 in.) that may contain as much as 25.3 Ci of internal corrosion products, or $8 \text{ } \mu\text{Ci}/\text{m}^2$. In cutting a piece of 25.4 cm (10 in.) diameter pipe with an oxyacetylene torch, 44.5 cm^2 with a total inventory of about $360 \text{ } \mu\text{Ci}$ could become airborne, and about $0.2 \text{ } \mu\text{Ci}$ would be released to the atmosphere.

Accidental Spraying of Concentrated Contamination with High Pressure Spray

Since only parts of the exterior surface of the primary system are decontaminated with the high pressure spray, there is the opportunity to spray an area not designated for decontamination. The concentrated area is postulated to be in the surrounding thermal insulation that has hidden a slow leak for a number of years. It is assumed to be a small area approximately 0.25 m^2 . The leak is assumed to have occurred and the deposited radioactivity was retained in the insulation as the liquid evaporated. This process leads to a gradual buildup of radioactive deposition.

The peak concentration of ^{137}Cs in the reactor coolant is assumed to be $1.3 \text{ } \mu\text{Ci}/\text{g}$.⁽⁴⁵⁾ Over a period of time the leak amounts to $18.9 \text{ } \ell$ (5 gal).

⁽⁴⁵⁾ TROJAN Final Safety Analysis Report, Vol. 6, Table 11-1-2, Portland General Electric Co., Portland, OR, May 1974.

Assuming a coolant density of 1, a total of $2.5 \times 10^4 \mu\text{Ci}/\text{cm}^3$ accumulates. The area would require 6.6 l to decontaminate with the high pressure spray, giving a solution radioactivity concentration of $3.7 \mu\text{Ci}/\text{cm}^3$. Using the same method to determine the airborne release as in Section J.1.1.5 results in a calculated airborne release to the building ventilation system of 240 μCi .

Accidental Break Of Contaminated Piping During Inspection

The Safe Storage Continuing Care period is assumed to continue for 50 years. Is assumed that during this time pipe is weakened by corrosion and becomes fragile so that it is very easily damaged. The pipe is postulated to become damaged by hitting it with a piece of equipment, dropping tools or radiation monitoring equipment, or even from a worker jostling the pipe. This damage is assumed to cause a fairly large section of pipe, with several square meters of surface area, to give way and allow the dry material inside to escape. The release affects a floor surface area as large as 5 m^2 , depending on the precipitating events.

A maximum amount of radioactivity released is obtained by assuming that the radioactive inventory of a small pipe that breaks equals the radioactive inventory of the inside of the RCS piping, or $86 \mu\text{Ci}/\text{cm}^2$ at shutdown. No in situ chemical decontamination is assumed, and the piping is assumed to be in useable condition when the reactor was shutdown. The radioactivity deposited in the piping is both corrosion products and fission products. ^{60}Co with a 5.3 year half life is assumed to be a prominent contributor to this radioactivity. Several years of radioactivity decay is assumed to take place before this type of event is anticipated, and a maximum estimate of the radioactivity at the time of the breach is $43 \mu\text{Ci}/\text{m}^2$. If 5 m^2 is breached, then 2.15 Ci is available to escape from the pipe. Of this, an estimated 10%, or 0.215 Ci, falls from the pipe. The material is dry and dusty and easily airborne. An estimated 10% of that removed from the pipe, or 0.0215 Ci, becomes airborne in the immediate vicinity of the break.

The ventilation system is not operating during the continuing care period, so the release dissipates by diffusion to the atmosphere. This is a slow process and probably does not lead to an immediate atmospheric release. Much of

the release falls to the floor. It is estimated that less than 1% or 2.15×10^2 μCi reaches the HEPA filter bank, so less than .11 μCi eventually reaches the atmosphere.

This accident is postulated to occur in the Reactor Containment Building. There is a possibility that a portion of the radioactive material is carried to the outside environment on the clothing and shoes of the workers involved. This movement does not lead to a significant atmospheric release because few inspections occur during this time period.

Fire in Contaminated Sweeping Compound

Sweeping compound is composed of sawdust treated with oil or other additives to enhance the pickup of contamination. It is postulated to catch fire by spontaneous combustion. The sawdust has an inventory of contamination from the floor surfaces. It is assumed that 250 cm^3 of sweeping compound is used for every square meter of surface, for a total of $6.5 \times 10^5 \text{ cm}^3$. It is also assumed that the density of the sawdust compound is about the same as light wood, or approximately 0.6 g/cm^3 .⁽⁴⁶⁾ Therefore, the weight of sawdust used is about $3.9 \times 10^5 \text{ g}$. Approximately half of the total 2000 μCi contamination on the floor is assumed to adhere to this sawdust, giving a concentration of $0.003 \mu\text{Ci/g}$ of sawdust. During the fire, the fractional release is taken as 1.5×10^{-4} , so 0.15 μCi is the total release from the sawdust accumulated from all the sweepings involved in a fire.

J.2.2.2 Dropping a Concrete Slab During Placement in Onsite Retrievable Waste Storage

Precast concrete slabs are used for top shields and a sealing surface for the waste storage location. One of these slabs is postulated to drop 6 m (20 ft) while it is being placed. It is unlikely that a shipping container designed to withstand rigorous transportation impacts is damaged in this free fall accident. Some surface particles become airborne, but these do not increase the routine release significantly, and are not considered further in this study.

⁽⁴⁶⁾ G. B. McKinnon, Fire Protection Handbook. National Fire Protection Association, Boston, MA, p. 3.202, 1976.

J.3 REFERENCE RADIONUCLIDE INVENTORIES

The residual radionuclide inventories at different locations within the reference PWR and on its site are presented and discussed in Section 7. They are repeated in this appendix to help illustrate the dependence of the calculated doses on the radionuclide mixtures. Reference radionuclide inventory numbers are assigned here to aid in tracing the relationships between these inventories, postulated atmospheric releases, and the calculated doses. The numbers assigned to these radionuclide inventories appear in Tables J.1-1 and J.1-2 and in the calculated dose tables presented in Section J.4. Reference radionuclide inventories 1 through 5 include lists of: 1) stainless steel activation products (Table J.3-1), 2) carbon steel activation products (Table J.3-2), 3) concrete activation products (Table J.3-3), 4) activated corrosion products (Table J.3-4), and 5) deposited surface contamination (Table J.3-5).

TABLE J.3-1. Reference Radionuclide Inventory 1, Stainless Steel Activation Products in the Core Stand

Radionuclide	Fractional Radioactivity Normalized to Reactor Shutdown at Decay Times of:				
	Shutdown	10 Years	30 Years	50 Years	100 Years
⁵⁴ Mn	2.6×10^{-2}	7.7×10^{-6}	--(a)	--	--
⁵⁵ Fe	4.9×10^{-1}	3.7×10^{-2}	2.2×10^{-4}	1.3×10^{-6}	--
⁵⁹ Fe	1.7×10^{-2}	--	--	--	--
⁵⁸ Co	5.7×10^{-2}	--	--	--	--
⁶⁰ Co	3.6×10^{-1}	9.6×10^{-2}	6.9×10^{-3}	5.0×10^{-4}	6.8×10^{-7}
⁵⁹ Ni	2.8×10^{-4}	2.8×10^{-4}	2.8×10^{-4}	2.8×10^{-4}	2.8×10^{-4}
⁶³ Ni	4.5×10^{-2}	4.2×10^{-2}	3.6×10^{-2}	3.2×10^{-2}	2.2×10^{-2}
⁶⁵ Zn	4.5×10^{-5}	1.3×10^{-9}	--	--	--
⁹³ Mo	1.4×10^{-7}	1.4×10^{-7}	1.4×10^{-7}	1.4×10^{-7}	1.4×10^{-7}
⁹⁴ Nb	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}
TOTAL	1.0	1.8×10^{-1}	4.3×10^{-2}	3.3×10^{-2}	2.2×10^{-2}

(a) A dash indicates values less than 1×10^{-10} .

**TABLE J.3-2. Reference Radionuclide Inventory 2, Carbon Steel
Activation Products Lower Vessel Wall**

Radionuclide	Fractional Radioactivity Normalized to Reactor Shutdown at Decay Times of:				
	Shutdown	10 Years	30 Years	50 Years	100 Years
⁵⁴ Mn	5.3×10^{-2}	1.6×10^{-5}	--(a)	--	--
⁵⁵ Fe	8.2×10^{-1}	6.3×10^{-2}	3.7×10^{-4}	2.1×10^{-6}	--
⁵⁹ Fe	3.1×10^{-2}	--	--	--	--
⁵⁸ Co	7.5×10^{-3}	--	--	--	--
⁶⁰ Co	8.5×10^{-2}	2.3×10^{-2}	1.6×10^{-3}	1.2×10^{-4}	1.6×10^{-7}
⁵⁹ Ni	3.6×10^{-5}	3.6×10^{-5}	3.6×10^{-5}	3.6×10^{-5}	3.6×10^{-5}
⁶³ Ni	4.3×10^{-3}	4.0×10^{-3}	3.5×10^{-3}	3.0×10^{-3}	2.1×10^{-3}
⁹³ Mo	1.5×10^{-6}	1.5×10^{-6}	1.5×10^{-6}	1.5×10^{-6}	1.5×10^{-6}
TOTAL	1.0	9.0×10^{-2}	5.5×10^{-3}	3.2×10^{-3}	2.1×10^{-3}

(a) A dash indicates values less than 1×10^{-10} .

**TABLE J.3-3. Reference Radionuclide Inventory 3, Calculated
Radioactivity Levels in Activated Biological
Shield(a)**

Radionuclide	Fractional Radioactivity Normalized to Reactor Shutdown at Decay Times of:				
	Shutdown	10 Years	30 Years	50 Years	100 Years
³⁹ Ar	1.14×10^{-3}	1.11×10^{-3}	1.05×10^{-3}	1.00×10^{-3}	8.78×10^{-4}
⁴¹ Ca	2.01×10^{-4}	2.01×10^{-4}	2.01×10^{-4}	2.01×10^{-4}	2.01×10^{-4}
⁴⁵ Ca	1.05×10^{-1}	2.30×10^{-8}	--(b)	--	--
⁵⁴ Mn	4.83×10^{-3}	1.05×10^{-6}	--	--	--
⁵⁵ Fe	8.65×10^{-1}	6.64×10^{-2}	3.91×10^{-4}	2.30×10^{-6}	--
⁶⁰ Co	1.92×10^{-2}	5.15×10^{-3}	3.71×10^{-4}	2.67×10^{-5}	3.73×10^{-8}
⁵⁹ Ni	3.42×10^{-5}	3.42×10^{-5}	3.42×10^{-5}	3.42×10^{-5}	3.42×10^{-5}
⁶³ Ni	4.02×10^{-3}	3.75×10^{-3}	3.27×10^{-3}	2.84×10^{-3}	2.01×10^{-3}
TOTAL	1.0	7.70×10^{-2}	5.32×10^{-3}	4.10×10^{-3}	3.12×10^{-3}

(a) The radionuclides listed include only those whose half-life and/or initial concentration result in a significant contribution after one year's decay and/or one hundred years decay.

(b) A dash indicates values less than 1×10^{-10} .

TABLE J.3-4. Reference Radionuclide Inventory 4, Estimates of the Deposition of Neutron-Activated Corrosion Products on Piping Internal Surfaces (a)

Radionuclide	Fractional Radioactivity Normalized to Reactor Shutdown at Decay Times of:				
	Shutdown	10 Years	30 Years	50 Years	100 Years
⁵¹ Cr	2.4×10^{-2}	-- (b)	--	--	--
⁵⁴ Mn	3.6×10^{-2}	1.1×10^{-5}	--	--	--
⁵⁹ Fe	8.2×10^{-3}	--	--	--	--
⁵⁸ Co	4.6×10^{-1}	--	--	--	--
⁶⁰ Co	3.2×10^{-1}	8.6×10^{-2}	6.2×10^{-3}	4.4×10^{-4}	6.0×10^{-7}
⁹⁵ Zr	5.6×10^{-2}	--	--	--	--
⁹⁵ Nb	5.6×10^{-2}	--	--	--	--
¹⁰³ Ru	2.6×10^{-2}	--	--	--	--
¹³⁷ Cs	1.2×10^{-3}	9.5×10^{-4}	6.0×10^{-4}	3.8×10^{-4}	1.2×10^{-4}
¹⁴¹ Ce	6.6×10^{-2}	--	--	--	--
TOTAL	1.0	8.7×10^{-2}	6.8×10^{-3}	8.2×10^{-4}	1.2×10^{-4}

(a) Based on surface corrosion product activities on a steam generator in Reference 8.

(b) A dash indicates values less than 1×10^{-10} .

Note: The activities are based on actual data from the Turkey Point Reactors extrapolated to 7 years of commercial operation.

TABLE J.3-5. Reference Radionuclide Inventory 5, Radioactive Surface Contamination in the Reference PWR

Radionuclide	Fractional Radioactivity Normalized to Reactor Shutdown at Decay Times of:				
	Shutdown	10 Years	30 Years	50 Years	100 Years
⁵⁴ Mn	1.4×10^{-3}	4.2×10^{-7}	--	--	--
⁵⁵ Fe	2.2×10^{-2}	1.7×10^{-3}	9.9×10^{-6}	5.7×10^{-8}	--
⁵⁹ Fe	8.7×10^{-4}	--	--	--	--
⁵⁸ Co	7.5×10^{-3}	--	--	--	--
⁶⁰ Co	7.5×10^{-2}	2.0×10^{-2}	1.4×10^{-3}	1.0×10^{-4}	1.4×10^{-7}
⁸⁹ Sr	1.2×10^{-3}	--	--	--	--
⁹⁰ Sr	6.9×10^{-4}	5.4×10^{-4}	3.4×10^{-4}	2.1×10^{-4}	6.3×10^{-5}
⁹⁰ Y	6.9×10^{-4}	5.4×10^{-4}	3.4×10^{-4}	2.1×10^{-4}	6.3×10^{-5}
⁹⁵ Zr	2.5×10^{-4}	--	--	--	--
⁹⁵ Nb	2.5×10^{-4}	--	--	--	--
^{129m} Te	3.1×10^{-4}	--	--	--	--
¹³¹ I	1.4×10^{-2}	--	--	--	--
¹³⁴ Cs	1.2×10^{-1}	4.1×10^{-3}	4.8×10^{-6}	5.4×10^{-9}	--
¹³⁶ Cs	1.1×10^{-3}	--	--	--	--
¹³⁷ Cs	7.5×10^{-1}	5.9×10^{-1}	3.7×10^{-1}	2.4×10^{-1}	7.4×10^{-2}
TOTAL	1.0	6.2×10^{-1}	3.7×10^{-1}	2.4×10^{-1}	7.4×10^{-2}

(a) A dash indicates values less than 1×10^{-10} .

J.4 CALCULATED RADIATION DOSES

The first-year doses and 50-year dose commitments, which were discussed in Appendix E, were calculated using the atmospheric releases and reference radionuclide inventories shown previously in this appendix. These calculated radiation doses are presented here and summarized in Section 11. A detailed discussion of the radiation dose models, parameters, and assumptions associated with the radiation dose calculations is given in Appendix E.

The airborne radioactive releases from routine decommissioning operations, listed in Table J.1-1, are used with the corresponding reference radionuclide inventories to calculate the radiation doses listed in Tables J.4-1 through J.4-4. Calculated radiation doses to the maximum exposed individual are listed in Tables J.4-1 and J.4-2, and calculated population radiation doses are based on the population distribution within a circle of 80 km radius around the reference PWR, with average individual exposure pathway parameters.

Airborne radioactive releases from postulated accidents during decommissioning operations, listed in Table J.1-2, are used to calculate the radiation doses listed in Tables J.4-5 and J.4-6. The calculations were done for the maximum exposed individual. The expected range of frequency of occurrence of each accident is also listed in these tables.

TABLE J.4-1. Radiation Doses to the Maximum Exposed Individual from Airborne Radionuclides during Normal Dismantlement Operations

Operation or Location	Reference Radionuclide Inventory Number	Release to Atmosphere (μCi)	First Year Dose, mrem				Fifty Year Dose Commitment, mrem			
			Total Body	GI-LLI	Bone	Lung	Total Body	GI LLI	Bone	Lung
Segmentation of Nonactivated Stainless										
Coolant Pumps & Primary Piping	4	7.5 × 10 ¹	1.7 × 10 ⁻⁵	3.1 × 10 ⁻⁵	1.6 × 10 ⁻⁵	4.3 × 10 ⁻⁵	1.7 × 10 ⁻⁵	3.1 × 10 ⁻⁵	1.6 × 10 ⁻⁵	5.8 × 10 ⁻⁵
Steam Generators	4	4.0 × 10 ¹	9.1 × 10 ⁻⁸	1.6 × 10 ⁻⁷	8.6 × 10 ⁻⁸	2.3 × 10 ⁻⁷	9.2 × 10 ⁻⁸	1.6 × 10 ⁻⁷	8.6 × 10 ⁻⁸	3.1 × 10 ⁻⁷
Segmentation of Activated Reactor										
Internals	1	5.0 × 10 ⁻¹	1.1 × 10 ⁻⁷	1.8 × 10 ⁻⁷	1.1 × 10 ⁻⁷	2.5 × 10 ⁻⁷	1.1 × 10 ⁻⁷	1.8 × 10 ⁻⁷	1.4 × 10 ⁻⁷	3.5 × 10 ⁻⁷
Vessel	2	2.0 × 10 ⁻¹	1.2 × 10 ⁻⁸	2.1 × 10 ⁻⁸	1.2 × 10 ⁻⁸	3.0 × 10 ⁻⁸	1.2 × 10 ⁻⁸	2.1 × 10 ⁻⁸	1.6 × 10 ⁻⁸	4.2 × 10 ⁻⁸
Waste Handling Bioshield Concrete	3	1.8 × 10 ⁰	4.7 × 10 ⁻⁸	7.7 × 10 ⁻⁸	5.1 × 10 ⁻⁷	8.0 × 10 ⁻⁸	1.1 × 10 ⁻⁶	7.7 × 10 ⁻⁸	1.1 × 10 ⁻⁶	1.6 × 10 ⁻⁶
Surface Cleaning Operations										
Hand Held Lance										
HX Room	5	3.9 × 10 ⁰	3.0 × 10 ⁻⁶	9.3 × 10 ⁻⁷	2.8 × 10 ⁻⁶	1.3 × 10 ⁻⁶	4.5 × 10 ⁻⁶	9.3 × 10 ⁻⁷	4.7 × 10 ⁻⁶	1.7 × 10 ⁻⁶
Evaporator Room	5	1.0 × 10 ⁰	7.7 × 10 ⁻⁷	2.4 × 10 ⁻⁷	7.1 × 10 ⁻⁷	3.3 × 10 ⁻⁷	1.2 × 10 ⁻⁶	2.4 × 10 ⁻⁷	1.2 × 10 ⁻⁶	4.4 × 10 ⁻⁷
Reactor Cavity	5	2.0 × 10 ⁻¹	1.5 × 10 ⁻⁷	4.8 × 10 ⁻⁸	1.4 × 10 ⁻⁷	6.6 × 10 ⁻⁸	2.3 × 10 ⁻⁷	4.8 × 10 ⁻⁸	2.4 × 10 ⁻⁷	8.8 × 10 ⁻⁸
Steam Generator Area	5	2.0 × 10 ⁻¹	1.5 × 10 ⁻⁷	4.8 × 10 ⁻⁸	1.4 × 10 ⁻⁷	6.6 × 10 ⁻⁸	2.3 × 10 ⁻⁷	4.8 × 10 ⁻⁸	2.4 × 10 ⁻⁷	8.8 × 10 ⁻⁹
Final Chemical Decontamination	4	7.8 × 10 ⁻¹	1.8 × 10 ⁻⁷	3.2 × 10 ⁻⁷	1.7 × 10 ⁻⁷	4.5 × 10 ⁻⁷	1.8 × 10 ⁻⁷	3.2 × 10 ⁻⁷	1.7 × 10 ⁻⁷	6.0 × 10 ⁻⁷
Ion Exchanger Vault	5	5.0 × 10 ⁻⁴	3.8 × 10 ⁻¹⁰	1.2 × 10 ⁻¹⁰	3.6 × 10 ⁻¹⁰	1.6 × 10 ⁻¹⁰	6.0 × 10 ⁻¹⁰	1.2 × 10 ⁻¹⁰	6.6 × 10 ⁻¹⁰	2.2 × 10 ⁻¹⁰
Laundry Room	5	1.9 × 10 ⁻⁸	1.5 × 10 ⁻¹²	4.5 × 10 ⁻¹³	1.3 × 10 ⁻¹²	6.3 × 10 ⁻¹³	2.2 × 10 ⁻¹²	4.5 × 10 ⁻¹³	2.3 × 10 ⁻¹²	8.4 × 10 ⁻¹³
In Situ Chemical Decontamination										
Spray Leak	5	7.0 × 10 ⁻¹	5.4 × 10 ⁻⁷	1.7 × 10 ⁻⁷	5.0 × 10 ⁻⁷	2.3 × 10 ⁻⁷	8.1 × 10 ⁻⁷	1.7 × 10 ⁻⁷	8.5 × 10 ⁻⁷	8.1 × 10 ⁻⁷
Liquid Leak	5	2.3 × 10 ⁻¹	1.8 × 10 ⁻⁹	5.5 × 10 ⁻¹⁰	1.6 × 10 ⁻⁹	7.6 × 10 ⁻¹⁰	2.7 × 10 ⁻⁹	5.5 × 10 ⁻¹⁰	2.8 × 10 ⁻⁹	1.0 × 10 ⁻⁹
Removal of Bioshield										
Explosive	3	5.3 × 10 ⁻¹	1.4 × 10 ⁻¹⁰	2.3 × 10 ⁻¹⁰	1.5 × 10 ⁻⁹	2.4 × 10 ⁻¹⁰	3.4 × 10 ⁻⁹	2.3 × 10 ⁻¹⁰	3.2 × 10 ⁻⁹	4.7 × 10 ⁻⁹
Drilling	3	6.0 × 10 ⁻⁵	1.6 × 10 ⁻¹²	2.6 × 10 ⁻¹²	1.7 × 10 ⁻¹¹	2.7 × 10 ⁻¹²	3.8 × 10 ⁻¹¹	2.6 × 10 ⁻¹²	3.6 × 10 ⁻¹¹	5.3 × 10 ⁻¹¹
Radiation Survey										
HX Room	5	2.0 × 10 ⁻¹	1.5 × 10 ⁻⁷	4.8 × 10 ⁻⁸	1.4 × 10 ⁻⁷	6.6 × 10 ⁻⁸	2.3 × 10 ⁻⁷	4.8 × 10 ⁻⁸	2.4 × 10 ⁻⁷	8.8 × 10 ⁻⁸
Boric Acid Evap. Room	5	5.0 × 10 ⁻²	3.8 × 10 ⁻⁸	1.2 × 10 ⁻⁸	3.5 × 10 ⁻⁸	1.6 × 10 ⁻⁸	5.8 × 10 ⁻⁸	1.2 × 10 ⁻⁸	6.1 × 10 ⁻⁸	2.2 × 10 ⁻⁸
Steam Generator Area	5	1.9 × 10 ⁻²	1.5 × 10 ⁻⁸	4.5 × 10 ⁻⁹	1.3 × 10 ⁻⁸	6.3 × 10 ⁻⁹	2.2 × 10 ⁻⁸	4.5 × 10 ⁻⁹	2.3 × 10 ⁻⁸	8.4 × 10 ⁻⁹
Laundry Room	5	1.0 × 10 ⁻⁷	7.7 × 10 ⁻¹⁴	2.4 × 10 ⁻¹⁴	7.1 × 10 ⁻¹⁴	3.3 × 10 ⁻¹⁴	1.2 × 10 ⁻¹¹	2.4 × 10 ⁻¹⁴	1.2 × 10 ⁻¹¹	4.4 × 10 ⁻¹⁴
Removal of Concrete Areas										
Explosives	5	1.0 × 10 ⁻⁵	7.7 × 10 ⁻¹²	2.4 × 10 ⁻¹²	7.1 × 10 ⁻¹²	3.3 × 10 ⁻¹²	1.2 × 10 ⁻¹¹	2.4 × 10 ⁻¹²	1.2 × 10 ⁻¹¹	4.4 × 10 ⁻¹²
Drilling	5	2.0 × 10 ⁻⁸	1.5 × 10 ⁻¹²	4.8 × 10 ⁻¹³	1.4 × 10 ⁻¹²	6.6 × 10 ⁻¹³	2.3 × 10 ⁻¹²	4.8 × 10 ⁻¹³	2.4 × 10 ⁻¹²	8.8 × 10 ⁻¹³
Pneumatic Jack Hammer	5	—(a)								
Rock Splitters	5	—(a)								

(a) A dash means that the atmospheric release value was less than $1 \times 10^{-17} \mu\text{Ci}$

TABLE J.4-2. Radiation Doses to the Maximum Exposed Individual from Airborne Radionuclides during Normal Safe Storage Operations

Operation or Location	Reference Radionuclide Inventory Number	Release to Atmosphere (μ Ci)	First Year Dose, mrem				Fifty Year Dose Commitment, mrem			
			Total Body	GI-LLI	Bone	Lung	Total Body	GI-LLI	Bone	Lung
Surface Cleaning Operations										
Hand Held Lance										
Primary System	5	1.0×10^0	7.7×10^{-7}	2.4×10^{-7}	7.1×10^{-7}	3.3×10^{-7}	1.2×10^{-6}	2.4×10^{-7}	1.2×10^{-6}	4.4×10^{-7}
Evaporator Room	5	1.0×10^0	7.7×10^{-7}	2.4×10^{-7}	7.1×10^{-7}	3.3×10^{-7}	1.2×10^{-6}	2.4×10^{-7}	1.2×10^{-6}	4.4×10^{-7}
Reactor Cavity	5	2.0×10^{-1}	1.5×10^{-7}	4.8×10^{-8}	1.4×10^{-7}	6.6×10^{-8}	2.3×10^{-7}	4.8×10^{-8}	2.4×10^{-7}	8.8×10^{-8}
Steam Generator Area	5	2.0×10^{-1}	1.5×10^{-7}	4.8×10^{-8}	1.4×10^{-7}	6.6×10^{-8}	2.3×10^{-7}	4.8×10^{-8}	2.4×10^{-7}	8.8×10^{-8}
Laundry Room	5	1.9×10^{-6}	1.5×10^{-12}	4.5×10^{-13}	1.3×10^{-12}	6.3×10^{-13}	2.2×10^{-12}	4.5×10^{-13}	2.3×10^{-12}	8.4×10^{-13}
Sweeping	5	4.0×10^{-2}	3.1×10^{-8}	9.5×10^{-9}	2.8×10^{-8}	1.3×10^{-8}	4.6×10^{-8}	9.5×10^{-9}	4.8×10^{-8}	1.8×10^{-8}
Vacuum Alternate	5	5.0×10^{-4}	3.8×10^{-10}	1.2×10^{-10}	3.5×10^{-10}	1.6×10^{-10}	5.8×10^{-10}	1.2×10^{-10}	6.1×10^{-10}	2.2×10^{-10}
In Situ Chemical Decontamination										
Spray Leak	5	$7.0 \times 10^{-1(a)}$	5.4×10^{-7}	1.7×10^{-7}	5.0×10^{-7}	2.3×10^{-7}	8.1×10^{-7}	1.7×10^{-7}	8.5×10^{-7}	8.1×10^{-7}
Liquid Leak	5	$2.3 \times 10^{-3(a)}$	1.8×10^{-9}	5.5×10^{-10}	1.6×10^{-9}	7.6×10^{-10}	2.7×10^{-9}	5.5×10^{-10}	2.8×10^{-9}	1.0×10^{-9}
Radiation Survey										
Hx Room	5	2.0×10^{-1}	1.5×10^{-7}	4.8×10^{-8}	1.4×10^{-7}	6.6×10^{-8}	2.3×10^{-7}	4.8×10^{-8}	2.4×10^{-7}	8.8×10^{-8}
Boric Acid Evap. Room	5	5.0×10^{-2}	3.8×10^{-8}	1.2×10^{-8}	3.5×10^{-8}	1.6×10^{-8}	5.8×10^{-8}	1.2×10^{-8}	6.1×10^{-8}	2.2×10^{-8}
Steam Generator Area	5	1.9×10^{-2}	1.5×10^{-8}	4.5×10^{-9}	1.3×10^{-8}	6.3×10^{-9}	2.2×10^{-8}	4.5×10^{-9}	2.3×10^{-8}	8.4×10^{-9}
Laundry Room	5	1.0×10^{-7}	7.7×10^{-14}	2.4×10^{-14}	7.1×10^{-14}	3.3×10^{-14}	1.2×10^{-13}	2.4×10^{-14}	1.2×10^{-13}	4.4×10^{-14}
Removal of Concrete Areas										
Drilling	5	—(b)								
Pneumatic Jack Hammer	5	—(b)								
Rock Splitters	5	—(b)								
Onsite										
Retrievable Waste Storage	5	1.5×10^1	1.1×10^{-5}	3.6×10^{-6}	1.1×10^{-5}	4.9×10^{-6}	1.7×10^{-5}	3.6×10^{-6}	1.8×10^{-5}	6.6×10^{-6}

(a) This is in situ Decontamination of the Safe Storage mode, and will not necessarily be used

(b) A dash means that the atmospheric release value was less than $1 \times 10^{-7} \mu\text{Ci}$

TABLE J.4-3. Radiation Doses to the Population from Airborne Radionuclides Released during Normal Dismantlement Operations(a)

Operation or Location	Reference Radionuclide Inventory Number	Release to Atmosphere (μCi)	First Year Dose, man-rem				Fifty Year Dose Commitment, man-rem			
			Total Body	GI-LLI	Bone	Lung	Total Body	GI-LLI	Bone	Lung
Segmentation of Nonactivated Stainless										
Coolant Pumps & Primary Piping	4	7.5 × 10 ¹	1 × 10 ⁻⁵	2 × 10 ⁻⁵	9 × 10 ⁻⁶	4 × 10 ⁻⁵	1 × 10 ⁻⁵	2 × 10 ⁻⁵	9 × 10 ⁻⁶	6 × 10 ⁻⁵
Steam Generators	4	4.0 × 10 ⁻¹	5 × 10 ⁻⁸	1 × 10 ⁻⁷	5 × 10 ⁻⁸	2 × 10 ⁻⁷	5 × 10 ⁻⁸	1 × 10 ⁻⁷	5 × 10 ⁻⁸	3 × 10 ⁻⁷
Segmentation of Activated Reactor										
Internals	1	5.0 × 10 ⁻¹	6 × 10 ⁻⁸	1 × 10 ⁻⁷	6 × 10 ⁻⁸	2 × 10 ⁻⁷	6 × 10 ⁻⁸	1 × 10 ⁻⁷	8 × 10 ⁻⁸	3 × 10 ⁻⁷
Vessel	2	2.0 × 10 ⁻¹	7 × 10 ⁻⁹	1 × 10 ⁻⁸	7 × 10 ⁻⁹	3 × 10 ⁻⁸	7 × 10 ⁻⁹	1 × 10 ⁻⁸	1 × 10 ⁻⁸	4 × 10 ⁻⁸
Waste Handling Bioshield Concrete	3	1.8 × 10 ⁰	2 × 10 ⁻⁷	3 × 10 ⁻⁷	6 × 10 ⁻⁷	3 × 10 ⁻⁷	1 × 10 ⁻⁶	3 × 10 ⁻⁷	1 × 10 ⁻⁶	2 × 10 ⁻⁶
Surface Cleaning Operations										
Hand Held Lance										
HX Room	5	3.9 × 10 ⁰	2 × 10 ⁻⁶	6 × 10 ⁻⁷	2 × 10 ⁻⁶	9 × 10 ⁻⁷	3 × 10 ⁻⁶	6 × 10 ⁻⁷	4 × 10 ⁻⁶	1 × 10 ⁻⁶
Evaporator Room	5	1.0 × 10 ⁰	6 × 10 ⁻⁷	1 × 10 ⁻⁷	5 × 10 ⁻⁷	2 × 10 ⁻⁷	9 × 10 ⁻⁷	1 × 10 ⁻⁷	9 × 10 ⁻⁷	3 × 10 ⁻⁷
Reactor Cavity	5	2.0 × 10 ⁻¹	1 × 10 ⁻⁷	3 × 10 ⁻⁸	1 × 10 ⁻⁷	5 × 10 ⁻⁸	2 × 10 ⁻⁷	3 × 10 ⁻⁸	2 × 10 ⁻⁷	7 × 10 ⁻⁸
Steam Generator Area	5	2.0 × 10 ⁻¹	1 × 10 ⁻⁷	3 × 10 ⁻⁸	1 × 10 ⁻⁷	5 × 10 ⁻⁸	2 × 10 ⁻⁷	3 × 10 ⁻⁸	2 × 10 ⁻⁷	7 × 10 ⁻⁸
Final Chemical Decontamination	4	7.8 × 10 ⁻¹	1 × 10 ⁻⁷	2 × 10 ⁻⁷	1 × 10 ⁻⁷	4 × 10 ⁻⁷	1 × 10 ⁻⁷	2 × 10 ⁻⁷	1 × 10 ⁻⁷	6 × 10 ⁻⁷
Ion Exchanger Vault	5	5.0 × 10 ⁻⁴	3 × 10 ⁻¹⁰	5 × 10 ⁻¹¹	2 × 10 ⁻¹⁰	1 × 10 ⁻¹⁰	4 × 10 ⁻¹⁰	5 × 10 ⁻¹¹	4 × 10 ⁻¹¹	2 × 10 ⁻¹⁰
Laundry Room	5	1.9 × 10 ⁻⁶	1 × 10 ⁻¹²	3 × 10 ⁻¹³	1 × 10 ⁻¹²	4 × 10 ⁻¹³	2 × 10 ⁻¹²	3 × 10 ⁻¹³	2 × 10 ⁻¹²	6 × 10 ⁻¹³
In Situ Chemical Decontamination										
Spray Leak	5	7.0 × 10 ⁻¹	4 × 10 ⁻⁷	1 × 10 ⁻⁷	4 × 10 ⁻⁷	2 × 10 ⁻⁷	6 × 10 ⁻⁷	1 × 10 ⁻⁷	6 × 10 ⁻⁷	2 × 10 ⁻⁷
Liquid Leak	5	2.3 × 10 ⁻¹	1 × 10 ⁻⁹	3 × 10 ⁻¹⁰	1 × 10 ⁻⁹	5 × 10 ⁻¹⁰	2 × 10 ⁻⁹	3 × 10 ⁻¹⁰	3 × 10 ⁻⁹	8 × 10 ⁻¹⁰
Removal of Bioshield										
Explosive	3	5.3 × 10 ⁻¹	7 × 10 ⁻¹⁰	8 × 10 ⁻¹⁰	2 × 10 ⁻⁹	8 × 10 ⁻¹⁰	3 × 10 ⁻⁹	8 × 10 ⁻¹⁰	3 × 10 ⁻⁹	6 × 10 ⁻⁹
Drilling	3	6.0 × 10 ⁻¹	8 × 10 ⁻¹²	9 × 10 ⁻¹²	2 × 10 ⁻¹¹	9 × 10 ⁻¹²	3 × 10 ⁻¹¹	9 × 10 ⁻¹²	3 × 10 ⁻¹¹	7 × 10 ⁻¹¹
Radiation Survey										
HX Room	5	2.0 × 10 ⁻¹	1 × 10 ⁻⁷	3 × 10 ⁻⁸	1 × 10 ⁻⁷	5 × 10 ⁻⁸	2 × 10 ⁻⁷	3 × 10 ⁻⁸	2 × 10 ⁻⁷	7 × 10 ⁻⁸
Boric Acid Evap. Room	5	5.0 × 10 ⁻²	3 × 10 ⁻⁸	7 × 10 ⁻⁹	3 × 10 ⁻⁸	1 × 10 ⁻⁸	4 × 10 ⁻⁸	7 × 10 ⁻⁹	5 × 10 ⁻⁸	2 × 10 ⁻⁸
Steam Generator Area	5	1.9 × 10 ⁻²	1 × 10 ⁻⁸	3 × 10 ⁻⁹	1 × 10 ⁻⁸	4 × 10 ⁻⁹	2 × 10 ⁻⁸	3 × 10 ⁻⁹	2 × 10 ⁻⁸	6 × 10 ⁻⁹
Laundry Room	5	1.0 × 10 ⁻⁷	6 × 10 ⁻¹⁴	1 × 10 ⁻¹⁴	5 × 10 ⁻¹⁴	2 × 10 ⁻¹⁴	9 × 10 ⁻¹⁴	1 × 10 ⁻¹⁴	9 × 10 ⁻¹⁴	3 × 10 ⁻¹⁴
Removal of Concrete Areas										
Explosives	5	1.0 × 10 ⁻³	6 × 10 ⁻¹²	1 × 10 ⁻¹²	5 × 10 ⁻¹²	2 × 10 ⁻¹²	9 × 10 ⁻¹²	1 × 10 ⁻¹²	9 × 10 ⁻¹²	3 × 10 ⁻¹²
Drilling	5	2.0 × 10 ⁻⁶	1 × 10 ⁻¹²	3 × 10 ⁻¹³	1 × 10 ⁻¹²	5 × 10 ⁻¹³	2 × 10 ⁻¹²	3 × 10 ⁻¹³	2 × 10 ⁻¹²	7 × 10 ⁻¹³
Pneumatic Jack Hammer	5	—(b)								
Rock Splitters	5	—(b)								

(a)The calculated radiation doses are shown to one significant figure

(b)A dash means that the atmospheric release value was less than $1 \times 10^{-7} \mu\text{Ci}$

TABLE J.4-4. Radiation Doses to the Population from Airborne Radionuclides Released during Normal Safe Storage Operations^(a)

Operation or Location	Reference Radionuclide Inventory Number	Release to Atmosphere (μ Ci)	First Year Dose, man-rem				Fifty Year Dose Commitment, man-rem			
			Total Body	GI-LLI	Bone	Lung	Total Body	GI-LLI	Bone	Lung
Surface Cleaning Operations										
Hand Held Lance										
Primary System	5	1.0×10^0	6×10^{-7}	1×10^{-7}	5×10^{-7}	2×10^{-7}	9×10^{-7}	1×10^{-7}	9×10^{-7}	3×10^{-7}
Reactor Cavity	5	2.0×10^{-1}	1×10^{-7}	3×10^{-8}	1×10^{-7}	5×10^{-8}	2×10^{-7}	3×10^{-8}	2×10^{-7}	7×10^{-8}
Steam Generator Area	5	2.0×10^{-1}	1×10^{-7}	3×10^{-8}	1×10^{-7}	5×10^{-8}	2×10^{-7}	3×10^{-8}	2×10^{-7}	7×10^{-8}
Laundry Room	5	1.9×10^{-6}	1×10^{-12}	3×10^{-13}	1×10^{-12}	4×10^{-13}	2×10^{-12}	3×10^{-13}	2×10^{-12}	6×10^{-13}
Sweeping	5	4.0×10^{-2}	2×10^{-8}	6×10^{-9}	2×10^{-8}	1×10^{-8}	4×10^{-8}	6×10^{-9}	4×10^{-8}	2×10^{-8}
Vacuum Alternate	5	5.0×10^{-4}	3×10^{-10}	7×10^{-11}	3×10^{-10}	1×10^{-10}	4×10^{-10}	7×10^{-11}	5×10^{-10}	2×10^{-10}
In Situ Decontamination										
Spray Leak	5	$7.0 \times 10^{-1(b)}$	4×10^{-7}	1×10^{-7}	4×10^{-7}	2×10^{-7}	6×10^{-7}	1×10^{-7}	6×10^{-7}	2×10^{-7}
Liquid Leak	5	$2.3 \times 10^{-3(b)}$	1×10^{-9}	3×10^{-10}	1×10^{-9}	5×10^{-10}	2×10^{-9}	3×10^{-10}	2×10^{-9}	3×10^{-10}
Radiation Survey										
Hx Room	5	2.0×10^{-1}	1×10^{-7}	3×10^{-8}	1×10^{-7}	5×10^{-8}	2×10^{-7}	3×10^{-8}	2×10^{-7}	7×10^{-8}
Boric Acid Evap. Room	5	5.0×10^{-2}	3×10^{-8}	7×10^{-9}	3×10^{-8}	1×10^{-8}	4×10^{-8}	7×10^{-9}	5×10^{-8}	2×10^{-8}
Steam Generator Area	5	1.9×10^{-2}	1×10^{-8}	3×10^{-9}	1×10^{-8}	4×10^{-9}	2×10^{-8}	3×10^{-9}	2×10^{-8}	6×10^{-9}
Laundry Room	5	1.0×10^{-7}	6×10^{-14}	1×10^{-14}	5×10^{-14}	2×10^{-14}	9×10^{-14}	1×10^{-14}	9×10^{-14}	3×10^{-14}
Removal of Concrete Areas										
Drilling	5	—(c)								
Pneumatic Jack Hammer	5	—(c)								
Rock Splitters	5	—(c)								
On Site										
Retrievable Waste Storage	5	1.5×10^1	9×10^{-6}	2×10^{-6}	8×10^{-6}	4×10^{-6}	1×10^{-5}	2×10^{-6}	1×10^{-5}	5×10^{-6}

^(a)The calculated radiation doses are shown to one significant figure

^(b)This is in situ Chemical Decontamination of the Safe Storage mode, and will not necessarily be used

^(c)A dash means that the atmospheric release value was less than $1 \times 10^{-7} \mu\text{Ci}$

TABLE J.4-5. Radiation Doses to the Maximum Exposed Individual from Accidental Airborne Radionuclide Releases during Dismantlement Operations

Incident	Reference Radionuclide Inventory Number	Release to Atmosphere (μCi)	First Year Dose, mrem				Fifty Year Dose Commitment, mrem				Expected Frequency of Occurrence ^(a)
			Total Body	GI-LLI	Bone	Lung	Total Body	GI LLI	Bone	Lung	
Explosion of LPG Leaked from Front End Loader	5	3.6×10^3	3.6×10^{-2}	3.5×10^{-3}	3.5×10^{-2}	4.7×10^{-2}	4.4×10^{-2}	3.6×10^{-3}	4.7×10^{-2}	5.4×10^{-2}	Low
Explosion of Oxyacetylene during Segmenting of Vessel Shell	2	3.6×10^2	4.3×10^{-5}	4.1×10^{-5}	6.6×10^{-5}	6.1×10^{-3}	6.9×10^{-3}	4.1×10^{-4}	2.4×10^{-4}	6.9×10^{-3}	Medium
Explosion and/or Fire of Ion Exchange Resin	5	3.8×10^1	3.8×10^{-4}	3.8×10^{-5}	3.7×10^{-4}	5.0×10^{-4}	4.6×10^{-4}	3.8×10^{-5}	5.0×10^{-4}	5.7×10^{-4}	Medium
Gross Leak during In Situ Chemical Decontamination											
Spray Leak	5	2.1×10^1	2.1×10^{-4}	2.1×10^{-5}	2.1×10^{-4}	2.8×10^{-4}	2.5×10^{-4}	2.1×10^{-5}	2.8×10^{-4}	3.2×10^{-4}	Medium
Liquid Leak	5	7.0×10^{-2}	7.1×10^{-7}	7.1×10^{-8}	6.9×10^{-7}	9.2×10^{-7}	8.5×10^{-7}	7.1×10^{-8}	9.2×10^{-7}	1.1×10^{-6}	Medium
Segmentation of RCS Piping with Unremoved Contamination	4	1.1×10^1	4.6×10^{-8}	5.5×10^{-5}	5.0×10^{-8}	7.3×10^{-4}	4.8×10^{-8}	5.5×10^{-5}	5.1×10^{-8}	7.9×10^{-4}	High
Loss of Contamination Control Envelope during Oxyacetylene Cutting of Vessel Shell	2	2.3×10^0	2.8×10^{-7}	2.6×10^{-8}	4.2×10^{-7}	3.9×10^{-7}	4.5×10^{-7}	2.6×10^{-6}	1.6×10^{-6}	4.4×10^{-5}	Medium
Pressure Surge Damage to Filters during Blasting of Activated Concrete Bioshield	3	3.0×10^{-1}	1.3×10^{-4}	1.4×10^{-7}	4.5×10^{-4}	2×10^{-6}	3.6×10^{-4}	1.4×10^{-7}	2×10^{-7}	2.2×10^{-6}	Low
Loss of Integrity of Portable Filtered Ventilation Enclosure	5	3.0×10^{-2}	3.0×10^{-7}	3.0×10^{-8}	2.9×10^{-7}	3.9×10^{-7}	3.6×10^{-7}	3.0×10^{-8}	3.9×10^{-7}	4.5×10^{-7}	Medium
Fire Involving Contaminated Clothing or Combustible Waste	5	6.0×10^{-3}	6.1×10^{-8}	6.1×10^{-9}	5.9×10^{-8}	7.9×10^{-8}	7.3×10^{-8}	6.1×10^{-9}	7.9×10^{-8}	9.1×10^{-8}	Medium
Loss of Blasting Mat during Removal of Activated Concrete	3	2.7×10^{-3}	1.2×10^{-10}	1.3×10^{-9}	4.1×10^{-10}	1.8×10^{-8}	3.2×10^{-10}	1.3×10^{-9}	1.3×10^{-9}	2.0×10^{-8}	Medium
Detonation of Unused Explosives in Reactor Cavity	3	2.7×10^{-3}	1.2×10^{-10}	1.3×10^{-9}	4.1×10^{-10}	1.8×10^{-8}	3.2×10^{-10}	1.3×10^{-9}	1.8×10^{-9}	2.0×10^{-8}	Medium
Temporary Loss of Local Airborne Contamination Control											
During Blasting	3	1.4×10^{-6}	6.2×10^{-14}	6.7×10^{-13}	2.1×10^{-13}	9.2×10^{-12}	1.7×10^{-13}	6.7×10^{-13}	9.4×10^{-13}	1.0×10^{-11}	Low

^(a)Frequency of occurrence High $>1 \times 10^{-2}$, Medium 1×10^{-2} to 1×10^{-5} , Low $<1 \times 10^{-5}$ per year

TABLE J.4-6. Radiation Doses to the Maximum Exposed Individual from Accidental Airborne Radionuclide Releases During Safe Storage Operations

Incident	Reference Radionuclide Inventory Number	Release, (μCi)	First Year Dose, mrem				Fifty Year Dose Commitment, mrem				Expected Frequency of Occurrence ^(a)
			Total Body	GI-LLI	Bone	Lung	Total Body	GI-LLI	Bone	Lung	
Gross Leak During In Situ Decontamination											
Spray Leak	5	2.1×10^1	2.1×10^{-4}	2.1×10^{-5}	2.1×10^{-4}	2.8×10^{-4}	2.5×10^{-4}	2.1×10^{-5}	2.8×10^{-4}	3.2×10^{-4}	Low
Liquid Leak	5	7.0×10^{-2}	7.1×10^{-7}	7.1×10^{-8}	6.9×10^{-7}	9.2×10^{-7}	8.5×10^{-7}	7.1×10^{-8}	9.2×10^{-7}	1.1×10^{-6}	Low
Vacuum Bag Rupture	5	1.0×10^0	1.1×10^{-6}	1.0×10^{-6}	9.8×10^{-6}	1.3×10^{-5}	1.2×10^{-5}	1.0×10^{-6}	1.3×10^{-5}	1.5×10^{-5}	Medium
Accidental Cutting of Contaminated Piping	4	1.8×10^{-1}	7.6×10^{-8}	9.0×10^{-7}	8.1×10^{-8}	1.2×10^{-5}	7.9×10^{-8}	9.0×10^{-7}	8.3×10^{-8}	1.3×10^{-5}	High
Accidental Spraying of Concentrated Contamination with High Pressure Spray	5	1.2×10^{-1}	1.3×10^{-7}	1.2×10^{-7}	1.2×10^{-6}	1.6×10^{-6}	1.5×10^{-6}	1.2×10^{-7}	1.6×10^{-6}	1.8×10^{-6}	High
Accidental Break of Contaminated Piping during Inspection	4	1.1×10^{-1}	4.6×10^{-8}	5.5×10^{-7}	5.0×10^{-8}	7.3×10^{-6}	4.8×10^{-8}	5.5×10^{-7}	5.1×10^{-8}	7.9×10^{-6}	Low
Fire Involving Contam. Clothing or Combustible Waste	5	6.0×10^{-3}	6.1×10^{-8}	6.1×10^{-9}	5.9×10^{-8}	7.9×10^{-8}	7.3×10^{-8}	6.1×10^{-9}	7.9×10^{-8}	9.1×10^{-8}	Medium
Fire in Contaminated Sweeping Compound	5	7.5×10^{-5}	8.2×10^{-11}	7.5×10^{-11}	7.4×10^{-10}	9.8×10^{-10}	9.0×10^{-11}	7.5×10^{-11}	9.8×10^{-10}	1.1×10^{-9}	Medium

^(a)Frequency of occurrence: High $>1 \times 10^{-2}$, Medium 1×10^{-3} to 1×10^{-4} , Low $<1 \times 10^{-4}$ per year

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