Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station

Appendices

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Appendices

MASTER

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APPENDIX A

EVALUATIONS OF DECOMMISSIONING FINANCING ALTERNATIVES

This appendix contains evaluations that estimate the revenue requirements for the alternatives for ensuring the availability of financing for decommissioning, as presented and discussed qualitatively in Chapter 6 of Volume 1. The evaluations in this appendix are in terms of constant dollars, assuming certain interest rates. They are sensitive to local, state, and federal taxes. However, only the impact of federal income taxes at the 46% rate is examined.

A.1 INFORMATION ABOUT REVENUE REQUIREMENTS

An examination of the present nuclear power plant 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 analyses presented in this appendix are based on scenarios appropriate for investor-owned utilities.

The development of the revenue requirements of a regulated investor-owned utility is a rather complex task. The utility is entitled to recover the costs of operation and capital investment in the nuclear power plant and, in addition, to obtain a reasonable return on its capital investment. Capital recovery is normally accomplished via a predetermined depreciation schedule.

The components of the revenue requirements related to a nuclear power plant include such non-taxable items as debt servicing, capital recovery (depreciation), operating expense, and applicable taxes. Revenues in excess of these items are normally controlled by the state's rate commission to allow a reasonable return on the depreciated capital investment in the plant.

A.2 CALCULATION OF REVENUE REQUIREMENTS

In this section, calculations are made of the annual and total revenue required by an investor-owned utility to fund decommissioning via the financing approaches discussed in Chapter 6 of Volume 1: prepayment, internal unfunded

reserve, sinking fund, and payment from revenue when needed. The calculations are only for the purpose of illustration, and no attempt is made to do an exhaustive study of all of the possible variations in scenarios. The effects of inflation are not included in the calculations in this section, but are discussed and presented in Section A.3.

A.2.1 Revenue Requirement for the Prepayment Alternative

In the prepayment alternative, it is assumed that the utility pays the estimated decommissioning cost in year-of-startup dollars into a trust fund operated by an agency of the state or federal government. The prepayment funds are obtained in the same manner as construction funds (i.e., from the sale of bonds or similar funding instruments) and the prepaid decommissioning costs are treated as a capital investment.

For this evaluation, it is assumed that the capital investment is recovered via straight-line depreciation and the annual revenues to cover depreciation are not subject to income taxes. Therefore, the annual revenue, R , needed to cover the decommissioning costs is given by:

$$R = \left\{ \begin{array}{l} \text{Debt Service on the} \\ \text{Prepaid Decommissioning} \\ \text{Costs} \end{array} \right\} + \left\{ \begin{array}{l} \text{Depreciation on the} \\ \text{Prepaid Decommissioning} \\ \text{Costs} \end{array} \right\} \quad (\text{A.1a})$$

or

$$R = \left(\begin{array}{c} \text{Prepaid} \\ \text{Decommissioning} \end{array} \right) \times \left(j + \frac{1}{\text{Depreciation Lifetime}} \right) \quad (\text{A.1b})$$

where j is the interest rate paid on the borrowed capital.

Not included in Equation A.1 is the return on the money in the trust fund. The excess of the trust fund over the decommissioning costs could be returned to the customers after decommissioning is completed and represent a windfall rate reduction at that time; or it could be returned on an annual basis through a reduction in the annual revenue requirement, which depends on the rate of return on

the fund. Because of the customer equality offered, the latter return alternative is chosen for evaluation. Assuming the estimated decommissioning cost remains constant, the annual refund, AR, is given by:

$$AR = \left(\frac{\text{Prepaid Decommissioning Cost}}{\text{Cost}} \right) \times (i) \quad (A.2)$$

where i is the rate of return on the trust fund.

Thus, the reduced annual revenue requirement, RR, is given by a combination of Equations A.1b and A.2:

$$RR = \left(\frac{\text{Prepaid Decommissioning Cost}}{\text{Cost}} \right) \times \left(j + \frac{1}{\text{Depreciation Lifetime}} - i \right) \quad (A.3)$$

where j is the interest rate paid on the borrowed capital and i is the rate of return achieved by the trust fund.

The evaluation results are obviously sensitive to the interest rates paid on the borrowed capital and to the rate of return obtained by the trust fund. If the estimated decommissioning cost is \$44 million and the depreciation lifetime is 30 years, the annual and total revenue requirements during this period are calculated using Equation A.3 and are tabulated in Table A.2-1 for a range of rates of return and interest rates.

A.2.2 Revenue Requirement for the Internal Unfunded Reserve Alternative

The internal unfunded reserve alternative is probably the simplest method for ensuring decommissioning funds since it fits readily into the normal accounting practices of the utilities. The estimated decommissioning cost is amortized over the life of the plant, on a straight-line depreciation schedule. Since these funds are collected in advance of the expenditure, they are not classified as an expense until the actual decommissioning takes place. Thus, these funds must come from the after-tax revenues. As a result, during the

TABLE A.2-1. Annual and Total Revenue Requirements for the Prepayment Alternative as Functions of Rates of Return and Interest Rates

Fund Rate of Return (% per year)	Interest Rate on Borrowed Capital (% per year)				
	2	4	6	8	10
	Annual/Total Revenue Requirements (\$ millions)				
1	1.90/57.2	2.79/83.6	3.67/110.0	4.55/136.4	5.43/162.8
2	1.47/44.0	2.35/70.4	3.23/96.8	4.11/123.2	4.99/149.6
3	--	1.90/57.2	2.79/83.6	3.67/110.0	4.55/136.4
4	--	1.47/44.0	2.35/70.4	3.23/96.8	4.11/123.2
5	--	--	1.90/57.2	2.79/83.6	3.67/110.0

(a) Estimated decommissioning cost = \$44 million; depreciation lifetime = 30 years.

plant's lifetime, the utility pays taxes on the revenue through higher rates charged to the customers, and, at shutdown when the funds are spent, the utility receives a tax credit that is passed on to the customers. Thus, the annual revenue requirement for this alternative is basically that which provides an after-tax amount equal to the estimated decommissioning cost divided by the depreciation lifetime of the plant.

The annual taxed revenue requirement, ATRR, is given by:

$$\text{ATRR} = \frac{\text{Estimated Decommissioning Cost}}{(1 - \text{Tax Rate}) (\text{Depreciation Lifetime})} \quad (\text{A.4})$$

For a decommissioning cost of \$44 million, an income tax rate of 46%, and a depreciation lifetime of 30 years, the annual taxed revenue requirement for decommissioning costs is calculated using Equation A.4 to be \$2.72 million per year for 30 years, for a total of \$81.5 million. If the revenue is not taxed, the annual revenue requirement is calculated to be \$1.47 million per year for 30 years, for a total of \$44.0 million.

It may be possible for the tax credit to be distributed over the same time period as the depreciation lifetime. If so, the annual revenue requirement would be reduced to that of the untaxed case.

A.2.3 Revenue Requirement for the Sinking Fund Alternative

Under the sinking fund alternative, the utility makes annual payments to an external escrow account. For this evaluation, the escrow account is assumed to be under the control of a state or federal agency, thereby exempting the income generated by the escrow account from income taxes. However, it is not presently clear whether the annual payments into the account would be from taxed revenue or from untaxed revenue, so both situations are evaluated. Since, in addition to the annual payments, the sinking fund also depends on the income from interest on its capital to reach the desired final amount at the end of the accumulation period, a range of interest rates (rates of return) for the funds in escrow is considered.

The untaxed annual revenue requirement, ARR, for payment into the escrow account is given by:

$$ARR = S \left\{ \frac{i}{(1 + i)^n - 1} \right\} \quad (A.5)$$

where S is the estimated decommissioning cost in constant dollars, i is the annual rate of return on the escrow account, and n is the number of years that payments are made.

If the revenue is taxed, the annual taxed revenue requirement, ATRR, is given by:

$$ATRR = \frac{S}{(1 - \text{Tax Rate})} \left\{ \frac{i}{(1 + i)^n - 1} \right\} \quad (A.6)$$

For the evaluation, a decommissioning cost of \$44 million and a depreciation period of 30 years are assumed with a tax rate of 46% as applicable. The evaluation of the sinking fund alternative uses Equations A.5 and A.6. The results, shown in Table A.2-2, are very sensitive to the rate of return and to the tax treatment.

TABLE A.2-2. Annual and Total Revenue Requirements for the Sinking Fund Alternative as Functions of Rates of Return and Tax Treatment^(a)

Fund Rate of Return (% per year)	Annual/Total Payment (\$ millions)	
	Untaxed Revenue	Taxed Revenue
1	1.265/37.95	2.34/70.28
2	1.086/32.54	2.01/60.19
3	0.925/27.75	1.71/51.33
4	0.785/23.54	1.46/43.69
5	0.662/19.87	1.22/36.67

(a) Estimated decommissioning cost = \$44 million, depreciation lifetime = 30 years, income tax rate = 46%.

A.2.4 Revenue Requirement for the Payment when Needed Alternative

Under this alternative, there are no annual revenue requirements prior to decommissioning. The decommissioning costs are paid from utility revenues as the work is performed. For comparison purposes, the total cost of dismantlement is assumed to be paid in 1 year. Thus, the revenue requirement is \$44 million in the first year of dismantlement.

A.3 COMPARISON OF ALTERNATIVES

The calculations presented in Section A.2 do not include the effects of inflation on the cost of decommissioning and on the annual revenue required to pay those costs. The effective rate of return on capital in an inflating economy is the difference between the obtainable interest rate and the inflation rate. In recent times, the interest rate on safe, conservative investments has generally exceeded the inflation rate by about 2%/yr, and the interest rate on borrowed capital generally has exceeded the inflation rate by about 4%/yr. On these bases and an estimated decommissioning cost of \$44 million, the resultant annual revenue requirements and the resultant total revenue requirements over 30 years for the financing alternatives are summarized in Table A.3-1.

TABLE A.3-1. Revenue Requirements for the Financing Alternatives(a)

Financing Alternative	Tax Treatment	Annual Payments (\$ millions)	Total Payments (\$ millions)
Prepayment	untaxed	2.35	70.4
Internal Unfunded Reserve	untaxed taxed(b)	1.47 2.72	44.0 81.5
Sinking Fund	untaxed taxed(b)	1.09 2.01	32.5 60.2
Paid When Needed	untaxed	--	44.0

(a) Estimated decommissioning cost = \$44 million, depreciation lifetime = 30 years, effective interest rate on fund = 2%/yr, effective interest rate on borrowed capital = 4%/yr.

(b) Most likely situation regarding taxes.

Considering the results from the three most likely cases given in Table A.3-1, the sinking fund alternative is the least costly. However, if some kind of bonding or insurance is required to ensure funding during the early portion of plant life, the cost of the bond or insurance could easily make the total cost of the sinking fund exceed the cost of the prepayment alternative. The internal unfunded reserve alternative appears to be the most expensive alternative, especially if some form of bonding or insurance during early plant life is required.

Another interesting case is when the facility is placed in safe storage for a number of years and is eventually dismantled. This case is more complex because the funds needed are collected during the operating lifetime of the plant and are disbursed during preparations for safe storage, continuing care, and deferred dismantlement. The unspent funds continue to accumulate interest during the continuing care period.

Let F represent the amount of money that must be in a fund at final reactor shutdown that, with interest accumulated during the safe storage period, is sufficient to pay all decommissioning costs. PSS is the cost of preparing the

plant for safe storage, CC is the annual cost of surveillance and maintenance, and DD is the cost of deferred dismantlement.

At time zero (reactor shutdown), F dollars are in the fund. Immediately, PSS dollars are paid out for preparations for safe storage. The balance remaining is $F - PSS$. At the end of the first year of safe storage, the fund will be increased by $(1 + i)$, where i is the annual interest rate earned by the fund and is decreased by $CC(1 + j)$, and where j is the annual inflation rate. The balance in the fund is $(F - PSS)(1 + i) - CC(1 + j)$. At the end of the second year the fund balance is

$$\left[(F - PSS)(1 + i) - CC(1 + j) \right] (1 + i) - CC(1 + j).$$

Expanding to the end of the n th year, when deferred dismantlement is to take place, the balance in the fund must equal the deferred dismantlement cost, $DD(1 + j)^n$:

$$(F - PSS)(1 + i)^n - CC \sum_{a=1}^n (1 + i)^{n-a}(1 + j)^a = DD(1 + j)^n \quad (A.7)$$

Equation (A.7) can be solved for F, the amount needed in the fund when reactor operation ceases:

$$F = \frac{DD(1 + j)^n + CC \sum_{a=1}^n (1 + i)^{n-a}(1 + j)^a}{(1 + i)^n} + PSS \quad (A.8)$$

The values of F, presented in Table A.3-2, are calculated assuming the values of DD, CC, and PSS estimated in this study; safe storage periods of 10, 30, 50, and 100 years; and either the interest rate is equal to the inflation rate or the interest rate is 2% greater than the inflation rate. From these values of F, the annual revenue required to create the fund is calculated for the prepayment, internal reserve, and sinking fund alternatives.

TABLE A.3-2. Revenue Requirements for the Passive Safe Storage/Deferred Dismantlement Alternative^(a)

Safe Storage Period	Funds Required at Shutdown		Annual Revenue Requirement					
			Prepayment ^(b)		Internal Reserve ^(c)		Sinking Fund ^(d)	
	$i = j$	$i = j + 2$	$i = j$	$i = j + 2$	$i = j$	$i = j + 2$	$i = j$	$i = j + 2$
10 ^(e)	57.55	51.43	3.07	2.74	3.55	3.17	2.63	2.35
30 ^(e)	59.05	43.26	3.15	2.31	3.65	2.67	2.70	1.97
50 ^(f)	51.45	34.08	2.74	1.82	3.18	2.10	2.35	1.56
100 ^(f)	55.20	28.73	2.94	1.53	3.41	1.77	2.53	1.31

(a) PSS = \$21.3 million, CC = \$0.075 million.

(b) Effective interest rate paid for capital is 4%, effective interest rate earned by fund is 2%.

(c) Tax rate is 46%, depreciation lifetime is 30 years.

(d) Tax rate is 46%, fund collection period is 30 years, effective interest rate earned is 2%.

(e) DD = \$35.5 million.

(f) DD = \$26.4 million.

Comparing the results presented in Table A.3-1 with those in Table A.3-2 for the case of the earned interest rate exceeding the inflation rate by 2% ($i = j + 2$), it is seen that the annual revenue requirement for the safe storage/deferred dismantlement alternative becomes less than for immediate dismantlement after a safe storage period of at least 30 years, and continues to decrease as the length of the storage period increases. This result is obviously very sensitive to the difference between the interest rate earned by the fund and the inflation rate.

REFERENCES

1. R. S. Wood and K. Kominars, Owners of Nuclear Power Plants, NUREG-0327 October 1977.*

* Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.

APPENDIX B

REFERENCE SITE DESCRIPTION

A reference site is used to assess the public safety and potential environmental effects of conceptually decommissioning various nuclear fuel-cycle facilities.⁽¹⁻³⁾ A summary of the site description found in References 1 through 3 is presented in this appendix. Only information directly relating to the radiation exposure pathway analysis, required in determining acceptable residual contamination levels and in calculating radiation doses for the public radiological safety analysis, is included here. The meteorological parameters and population distribution for the year 2000 are based on information presented in Reference 4. Other necessary site information is based on data reported in the site description of an operating nuclear power station.⁽⁵⁾ Estimates of the residual radioactivity mixtures and levels for the reference site are presented in Appendix E. All information in this appendix is felt to be representative of most existing and potential BWR power stations in the midwestern or middle southeastern United States.

Individual features of an actual site (for a specific facility) may vary from those of this reference site. However, it is believed that this approach (i.e., use of a reference site) results in a more meaningful overall analysis of the potential impacts associated with decommissioning most BWR facilities. A site-specific assessment is required for a similar analysis of a specific BWR power station.

B.1 SITE LOCATION AND SIZE

The reference site is located in a rural area with characteristics similar to those found in the midwestern or middle southeastern United States. The site occupies about 4.7 km² in a rectangular shape of about 2 km by 2.35 km. A moderate-size river flows through one corner of the site.

B.2 DEMOGRAPHY

The site is located in a rural area with a relatively low population density, with highest population densities occurring at distances of 20 to 60 km. Population distribution data for the year 2000 is given in Table B.2-1. The total population residing within an 80-km radius of the BWR site is about 3.52 million.

TABLE B.2-1. Population Distribution Around the Reference Site for the Year 2000(4)

Distance From Site Boundary (km)	Population Density (Persons/km ²)	Total Population In Annulus(a,b)	Cumulative Population
0 to 2	--(c)	10	10
2 to 3	136	2 130	2 140
3 to 5	104	5 230	7 370
5 to 6	230	7 940	15 300
6 to 8	133	11 700	27 000
8 to 20	85	89 300	116 000
20 to 30	239	375 000	491 000
30 to 50	175	878 000	1 370 000
50 to 60	298	1 030 000	2 400 000
60 to 80	127	1 120 000	3 520 000

(a) It is assumed that the population in each annulus is uniformly divided in each of 16 uniform 22.5 degree sectors.

(b) Totals are rounded to three significant figures.

(c) Indicates a population density less than 1.0/km².

B.3 LAND USE

Use of any part of the total site area for anything other than reactor operations is assumed to be prohibited during the 40-year BWR operational lifetime. The major plant facilities are located inside a 0.12-km², fenced portion of the site. The minimum distance from the point of airborne release to the outer site boundary is 1 km. The outer site boundary is fenced and marked.

About 80% of the land within 16 km of the site is used for farming. The main crops are soybeans (60%), corn, oats, and other grains (30%) and hay (10%). It is expected that this area will remain largely agricultural, and that the population will not change significantly because of the BWR operations.

A wildlife refuge and a state forest and campground are located about 14 km from the site. A state park is located about 10 km from the site in the opposite direction.

There are large truck gardens in the area. The nearest dwelling (the residence location of the maximum-exposed individual for the public safety analysis of decommissioning) is a farm located about 1.3 km 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 with a growing season of 5 months is kept for fresh vegetables. River water is used to irrigate the crops on this farm.

B.4 METEOROLOGY

The reference 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. Table B.4-1 shows monthly meteorological statistics.

TABLE B.4-1. Monthly Meteorological Statistics 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>
<u>Air Temperature (°C)</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
<u>Mean Relative Humidity (%)</u>												
	74	75	73	66	62	66	68	70	70	66	73	78

On the average, 12 days per year have maximum temperatures of 32°C and above. Annually, an average of 168 days have minimum temperatures of 0°C and below, with 40 of them at -18°C or below. The January average relative humidities at 7:00 a.m., 1:00 p.m., and 7:00 p.m. are 76, 68, and 70%, respectively. The corresponding humidities for July are 86, 55, and 55%.

The average annual rainfall in the area is 610 mm. The months of May through September have the greatest amounts of rainfall, with an average during this period of 432 to 457 mm (70% of the annual total). The maximum 24-hr total rainfall for the period 1894 through 1965 was 127 mm and occurred in May. Thunderstorms, with an average annual frequency of 36, are the chief source of rain from May through September. Snowfall in the area averages 1070 mm annually, with occurrences recorded in all months except June, July, and August. The extremes in annual snowfall of record are 152 mm minimum and 2235 mm 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 and for the other seasons is about 16 km/hr. The maximum reported wind speed of 148 km/hr 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 is assumed to be associated with tornadoes.

Natural fog that restricts visibility to 0.4 km or less occurs about 30 hr/yr. Icing caused by 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. Thermal inversions occur 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 power reactors are used to calculate the "typical" annual atmospheric dispersion pattern in an average 22.5-degree sector around the site. Dispersion factors at selected distances for the average sector are determined from joint-frequency distributions for each site. This is done by calculating the dispersion factor, $\bar{\chi}/Q'$, 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 are added without regard to direction and divided by the number of sectors involved. Thus, an average dispersion factor is obtained for each selected downwind distance for all 16 sectors, based on data given in Reference 4.

Standard groups of meteorological data are interpolated from the specific site data. The groupings provide four stability classes based on vertical temperature gradient and five wind speed classes based on the Beaufort wind scale.⁽⁶⁾ The stability classes are based on Reference 7 information, with Pasquill classes A, B, and C classified as B (unstable); Pasquill class D (neutral); Pasquill Class E (slightly stable); and Pasquill classes F and G are classified as F (moderately stable).

Some of the wind-speed data is available for only one height. In this event, the measured values are extrapolated to the 10-m level for ground level release calculations and to the 150-m level for reference-BWR stack release calculations. Where measurements at two heights are available, the highest is extrapolated to 150 m and the lowest to 10 m, using a standard power law extrapolation procedure.⁽⁶⁾

The ratio of the maximum sector dispersion factor to the average sector dispersion factor is 2.5. This value is used for all release heights in this study. Investigation of the change in this ratio with increasing distance from the site shows that the ratio remains essentially constant. The dispersion factors for the average sector as a function of height of release and downwind distance are shown in Figure B.4-1.

FIGURE B.4-1. Average ($\bar{X}/Q'$) Values Versus Distance in a Sector from the Reference Site

To assess the effect of increased stack height, atmospheric dispersion factors for stack heights of 150, 200, and 300 m are estimated from the original joint frequency distributions of the information from Reference 4. These values are graphically presented in Figure B.4-1.

No credit for plume rise from either momentum or buoyancy is taken in this study. Where large volumes of heated air are being ejected, the plume rise constant for momentum is estimated to be about $50 \text{ m}^2/\text{sec}$. Assuming an annual average wind speed of 2 to 3 m/sec, the increase in effective stack height because of momentum would be about 15 to 25 m. Plume rise from buoyancy (heat effect) would add at least another 25 to 100 m of effective stack height, depending on the temperature of the exhaust gases. Thus, the $\bar{x}/Q'$ values illustrated in Figure B.4-1 are larger than they would be if credit had been taken for momentum and buoyancy.

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* Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.

** Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and the National Technical Information Service, Springfield, Virginia 22161.

APPENDIX C

REFERENCE BWR FACILITY DESCRIPTION

This appendix describes the reference boiling water reactor (BWR) in sufficient detail to provide an appreciation of the problems associated with decontamination and decommissioning efforts. A 3320-MWt (1155-MWe) BWR is characterized. Descriptions of the structures and the equipment housed therein are included. The information contained here is based on the Washington Public Power Supply System (WPPSS) Final Safety Analysis Report for the WPPSS Nuclear Project No. 2 (WNP-2),⁽¹⁾ from the WNP-2 Environmental Report,⁽²⁾ and from drawings and other data supplied by WPPSS personnel. The single-unit WNP-2 employs a BWR nuclear steam supply system, supplied by the General Electric Company and designated BWR/5.

C.1 GENERAL PLANT DESCRIPTION

The reference site occupies approximately 4.7 km² in a rectangular shape of 2 km by 2.35 km. The plant itself requires an area of 0.12 km² within the larger plant site. A well-labeled perimeter fence exists around the total site to exclude the public. Another fence surrounds the smaller plant area and includes a security entrance to the plant. A moderately sized river runs through one corner of the site. The reference BWR site layout is shown in Figure C.1-1. The principal structures located on the reference plant site are:

- Reactor Building - houses the nuclear steam supply system, the primary containment vessel, new and spent fuel storage, refueling equipment, and emergency core cooling systems
- Turbine Generator Building - houses the power conversion equipment
- Condensate Storage Tanks (2 each)
- Radwaste and Control Building - houses the liquid and solid radwaste systems, components of the condenser off gas system, and the plant's main control room

FIGURE C.1-1. Site Layout of the Reference BWR Plant Facilities

- Diesel Generator Building - houses the standby diesel generators and oil storage tanks
- Service Building - houses the makeup water treatment system, a machine shop, and an office area
- Cooling Towers (6 each)
- Circulating Water Pumphouse - houses the circulating water pumps, plant service water pumps, and fire pumps
- Spray Ponds - cooling ponds provided as the ultimate heat sink

- Standby Service Water Pumphouses - house the redundant standby service water pumps
- Makeup Water Pumphouse - houses the plant makeup water pumps.

C.1.1 Design Criteria

The bases for development and selection of the design criteria used in this plant are: provide protection of public safety, provide for reliable and economic plant performance, and 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.

C.1.2 Operating Characteristics

The design and license power level for the reactor core is approximately 3320 MWt with a corresponding generator electrical output of approximately 1155 MWe. The principle of operation is shown in Figure C.1-2.

C.1.3 Nuclear Steam Supply System

The nuclear steam supply system includes a direct-cycle, forced-circulation BWR that produces steam for direct use in the steam turbine.

The nuclear steam supply system includes those systems most closely associated with the reactor vessel that are designed to contain or be in communication with water going to and steam coming from the reactor core. The nuclear steam supply system includes the following:

- reactor vessel
- reactor vessel internals
- reactor core
- main steam lines from the reactor vessel out to and including the isolation valves outside the primary containment structure

- neutron monitoring system
- reactor water recirculation system
- control rod drive system
- residual heat removal system
- reactor core isolation cooling system
- emergency core cooling systems
- reactor water cleanup system
- reactor vessel pressure relief system.

A NUCLEAR POWER REACTOR PRODUCES STEAM TO DRIVE A TURBINE WHICH TURNS AN ELECTRIC GENERATOR. THE BWR 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, BOILS, AND PRODUCES STEAM THAT IS PIPED TO THE TURBINE.

FIGURE C.1-2. Functional Diagram of a Boiling Water Reactor (BWR)

The fuel for the reactor core consists of slightly enriched uranium dioxide pellets sealed in Zircaloy-2 tubes. The fuel rods are assembled into 764 individual fuel assemblies. The 185 moveable, bottom-entry control rods are cruciform in shape and are dispersed throughout the lattice of fuel assemblies.

The reactor vessel contains the core and supporting structures, the steam separators and dryers, the jet pumps, the control rod guide tubes, and other components. The low-alloy steel reactor vessel is cylindrical with hemispherical heads. Except for the top hemispherical head, the reactor vessel is clad internally with stainless steel.

Four steam lines exit the upper portion of the reactor vessel. Between the reactor vessel and the turbine, each steam line is provided with two isolation valves in series, one on each side of the primary containment structure wall.

The reactor water recirculation system has two loops external to the reactor vessel but inside the primary containment structure. Each external loop contains a pump and a flow control valve. The portion of each loop that is inside the reactor vessel consists of the jet pumps that are located in the annular region between the core shroud and the vessel's inner wall.

Auxiliary systems purify reactor recirculation water, cool system components, remove residual heat when the reactor is shut down, sample reactor coolant water, provide for emergency water injection, and provide for reactor vessel pressure relief.

C.1.4 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 a deaerating surface condenser, steam jet air ejectors, turbine-driven feedwater pumps, motor-driven condensate and condensate booster pumps, a full-flow condensate demineralizer system, and six stages of feedwater heaters. The steam and turbine systems are designed to receive the heat energy produced in the reactor during

normal operation, as well as during a 25% load reduction on the turbine-generator. The generator is directly coupled to the turbine and has a hydrogen-cooled rotor and a water-cooled stator. It is a three-phased, 60-Hz, 25,000-volt, 1800-rpm unit rated at 1230 MVA at a 0.975 power factor.

C.1.5 Electrical Power

Normal auxiliary power is provided by two normal auxiliary transformers fed from the main generator 25-kV isolated phase bus. The startup transformer is connected to the 230-kV offsite power. These auxiliary systems each have the capacity to carry the full plant auxiliary load. A backup transformer is provided to supply all plant emergency system loads. This transformer is supplied from an offsite 115-kV line to the plant.

Two diesel generators, at 4.16 kV each, are provided as standby sources of emergency power in the event of a loss of normal AC power. A third diesel generator supplies emergency power to the high-pressure core spray system. Each generator has sufficient capacity to operate the equipment necessary to prevent undue risk to the public in an emergency situation. In addition, storage batteries are provided as onsite sources of power in the event of a loss of normal DC power.

C.1.6 Fuel Handling System

The reactor is refueled with equipment designed to handle the 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 from the spent fuel. This system also provides capability for receiving, handling, and storing new fuel.

C.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 produced during reactor operation.

Liquid wastes potentially containing radioactive material are collected, filtered, and concentrated or demineralized as required. The treated water from

the demineralizers or the concentrator distillate may be recycled for use in the plant or may be discharged to the nearby river. The concentrator bottoms, spent demineralizer resins, and spent filters are solidified, packaged, and shipped to a licensed disposal site. Gaseous wastes are collected and discharged to the environment after filtration.

C.2 STRUCTURES

This section describes the structures of the reference BWR. Refer to Figure C.1-1 for the arrangement of the structures on the plant site. Table C.2-1 lists the structural materials used in the various plant structures.

In this section and subsequent sections, equipment key numbers (EKN) are supplied each time a piece of equipment is mentioned to facilitate cross correlation between the text, the figures, and the equipment lists in this appendix. The equipment key numbers in this appendix are identical to numbers assigned by the facility owner (see Reference 1, Figure 1.2-2). A complete listing of all the equipment key numbers and the corresponding equipment identified in this appendix is presented in Section C.6.

TABLE C.2-1. Estimated Structural Material Quantities in the Reference Boiling Water Reactor Facilities

<u>Structure</u>	<u>Concrete, m³</u>	<u>Rebar, Mg</u>	<u>Structural Steel, Mg</u>
Reactor Building	42 804	8 608	902
Primary Containment	1 225	187	693
Turbine Generator Building	46 672	4 717	742
Radwaste and Control Building	26 697	2 746	372
Diesel Generator Building	2 964	408	0
Service Building	2 359	151	395
Circulating Water Pump House	2 963	321	112
Spray Pond Complex	8 321	789	0
Makeup Water Pump House	1 278	163	0
Yard and other buildings	<u>12 825</u>	<u>247</u>	<u>91</u>
Total	147 880	18 351	3 307

The Reactor Building, the Turbine Generator Building, the Radwaste and Control Building, the Diesel Generator Building, and the Service Building are in close proximity to each other, being supported on separate foundation mats and physically separated from each other, above and below grade. Abutting walls between these buildings are separated by several millimeters.

The Reactor Building, the Turbine Generator Building, and the Radwaste and Control Building are the only buildings on the plant site that contain radioactive materials.

C.2.1 Reactor Building

The Reactor Building (Figures C.2-1 through C.2-4) is a Seismic Category I^(a) structure consisting of two containment barriers: the steel primary containment vessel and the building itself, which provides secondary containment. The primary containment vessel (General Electric Co. Mark II containment design) contains a drywell, a suppression chamber, a structural floor separating the drywell from the suppression chamber, a sacrificial shield wall, and a reactor pedestal. The building secondary containment encloses the biological shield wall, the spent fuel storage pool, the dryer and separator storage pool, and the reactor well pool.

The Reactor Building completely surrounds the primary containment vessel. The building provides secondary containment when the primary containment vessel is closed and in service, and serves as a primary containment during periods when the primary containment vessel is open. The reactor building also houses refueling and reactor servicing equipment, new and spent fuel storage facilities, and other reactor safety and auxiliary systems.

(a) The design bases for individual structures and portions of structures are dependent on the systems that the structures support and house. Design bases for Seismic Category I structures are defined as being within the following broad 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. One or more of these may apply to each portion of the structure. The number and quantitative degree of design bases is dependent on the specific needs of components (locally) and systems (generally).

TOP OF PARAPET
STEEL EL. 204.5m

FIGURE C.2-1. Reactor Building Section

EQUIPMENT KEY

- 2. REACTOR PEDESTAL
- 3. VENT PIPES
- 4. LPCS PUMP
- 5. HPCS PUMP
- 8. RHR PUMP
- 10. RCIC PUMP
- 17. RADWASTE BLDG. CONDENSATE SUPPLY PUMP
- 18. CONDENSATE FILTER-DEMINERALIZER BACKWASH PUMP
- 21. CONTROL ROD DRIVE PUMP
- 27. CONTROL INSTRUMENT AIR TANK
- 33. REACTOR WATER RECIRCULATION PUMP
- 34. DRYWELL PERSONNEL LOCK
- 35. DRYWELL EQUIPMENT HATCH
- 39. AIR COMPRESSOR
- 40. MAIN STEAM TUNNEL
- 42. TIP DRIVE MECHANISMS
- 65. AIR LOCK
- 97. FLOOR DRAIN SUMP
- 98. EQUIPMENT DRAIN SUMP
- 100. FAN COIL UNIT
- 107. HEATING AND VENTILATING UNIT
- 124. AIR DRYER
- 700. ELEVATOR
- 701. COVERED HATCH
- 702. OPEN HATCH
- 703. RAILROAD

FIGURE C.2-2. Reactor Building Floor Plans - Sheet 1

REACTOR BLDG. - FLOOR PLAN AT ELEV. 159.1m

REACTOR BLDG. - FLOOR PLAN AT ELEV. 167.0m

EQUIPMENT KEY

1. REACTOR VESSEL
2. REACTOR PEDESTAL
46. CONTROL ROD DRIVE MODULES
47. RWCU PUMP
49. SACRIFICIAL SHIELD WALL
50. RBCC WATER PUMP
51. RBCC WATER HEAT EXCHANGERS
54. RHR HEAT EXCHANGER
55. FUEL POOL CIRCULATION PUMP
56. FUEL POOL HEAT EXCHANGER
58. STANDBY LIQUID CONTROL STORAGE TANK
60. STANDBY LIQUID CONTROL PUMPS
61. REGENERATIVE HEAT EXCHANGERS
62. NON-REGENERATIVE HEAT EXCHANGERS
63. REACTOR WELL
66. STEAM DRYER STORAGE
67. STEAM SEPARATOR STORAGE
68. SPENT FUEL SHIPPING CASK STORAGE
74. UPPER AND LOWER SHIELD PLUG LAYDOWN
75. REACTOR VESSEL HEAD INSULATION LAYDOWN
76. FLUX MONITOR SHIPPING CRATE
77. NEW FUEL STORAGE VAULT
79. DRYWELL HEAD LAYDOWN
82. UPPER POOL PLUG LAYDOWN
83. LOWER POOL PLUG LAYDOWN
84. REACTOR VESSEL HEAD LAYDOWN
85. SPENT FUEL CASK WASHDOWN AREA
88. SLOT PLUG LAYDOWN
89. STUD TENSIONER SPREADER
90. SPENT FUEL STORAGE POOL
91. RBCC WATER SURGE TANK
92. STANDBY GAS FILTER UNIT
93. EXHAUST AIR DAMPERS
94. HYDROGEN RECOMBINER
95. EXHAUST AIR PLENUM
96. SUMP VENT FILTER UNIT
100. FAN COIL UNIT
107. HEATING AND VENTILATING UNIT
109. REACTOR BUILDING VENT
112. DRYER AND SEPARATOR STORAGE POOL
700. ELEVATOR
701. COVERED HATCH
702. OPEN HATCH

REACTOR BLDG. - FLOOR PLAN AT ELEV. 174.4m

REACTOR BLDG. - FLOOR PLAN AT ELEV. 185.0m

KEY PLAN

FIGURE C.2-3. Reactor Building Floor Plans - Sheet 2

FIGURE C.2-4. Reactor Building Elevation Sections

The Reactor Building is a cast-in-place, reinforced concrete structure up to and including the operating floor at elevation 185.0 m. Reinforced concrete shield walls extend above this level, 2.43 m high on the north exterior side and 1.83 m high on the west exterior side of the building, immediately inside the insulated metal exterior siding of the steel superstructure.

Above the operating floor, the building structure is constructed of structural steel members and insulated metal siding and roof decking. Joints in the superstructure paneling are designed to ensure air leak tightness. The steel superstructure supports the overhead bridge crane (115-Mg capacity) and houses refueling operations.

The foundation mat, shown in Figure C.2-5, is a common foundation that supports the steel primary containment vessel and the Reactor Building that encloses it. The foundation mat is 4.9-m thick reinforced concrete.

Of the eight main floor levels, from and including the top of the foundation mat to and including the operating floor, the foundation mat level is the only one below the final plant grade, which is at elevation 134.4 m.

The Reactor Building is rectangular in plan and elevation. The approximate inside plan dimensions of the concrete structure are 38.7 m by 43.6 m. The steel superstructure is approximately 39.6 m by 44.5 m in plan. From the basement level, which is the top of the foundation mat at elevation 128.7 m, to elevation 143.6 m, the exterior concrete walls are 1.22 m thick; from elevation 143.6 m to the operating floor, elevation 185.0 m, the exterior wall thicknesses vary from 0.46 m to 0.91 m.

The height of the concrete structure is approximately 56.3 m, measured from the top of the foundation mat, elevation 128.7 m, to the operating floor, elevation 185.0 m. The height of the steel superstructure is approximately 19.5 m from the operating floor to the top of the superstructure parapet.

Interior walls and floor slabs are of reinforced normal or high-density concrete of various thicknesses, depending on shielding requirements. In the building interior, the floor slabs are supported by reinforced concrete columns and by a biological shield wall that surrounds the primary containment vessel.

FIGURE C.2-5. Reactor Building Foundation

The biological shield wall is a reinforced concrete structure that varies in thickness from 1.52 m to 1.83 m and extends from the top of the foundation mat to the operating floor. The top of the shield wall forms the reactor well pool cavity and is closed by means of removable, segmented, reinforced concrete shield plugs. The wall is reinforced by horizontal hoop and vertical reinforcing steel and is tied to the foundation mat by metal dowels that penetrate both structures to provide continuity.

The spent fuel storage pool (EKN 90) and the dryer and separator storage pool (EKN 112) are supported by two deep reinforced concrete beams that extend from the east and west exterior walls to the biological shield wall at each elevation above the foundation mat. These beams are constructed monolithically with and supported by the wall buttresses and the biological shield wall. Above elevation 159.1 m, an intermediate column under each deep beam provides additional support for the pool support system.

C.2.1.1 Primary Containment Vessel

The primary containment vessel contains a drywell that houses the reactor vessel, the reactor coolant recirculating loops, and connections of the reactor primary system. It also contains a pressure suppression chamber that stores 3160 m³ (maximum) of water and a submerged vent system that connects the drywell and the suppression pool (see Figure C.2-1).

The primary containment vessel is a free-standing steel pressure vessel with a dividing floor forming an over-under configuration inside. The vessel is enclosed in the biological shield wall, but it is separated from the wall by an annulus of compressible insulation material approximately 50 mm thick.

The drywell is located directly above the wetwell. The drywell configuration is basically a truncated cone with removable ellipsoidal top closure head. The suppression chamber (wetwell) is cylindrical with an ellipsoidal base. The primary containment vessel is anchored to the concrete mat foundation. The bottom of the suppression chamber is lined on the inside with reinforced concrete. The physical dimensions of the steel primary containment vessel are:

- The diameter of the cylindrical portion at the base of the cone is approximately 26.2 m.
- The diameter at the top of the cone is approximately 11.4 m and then narrows to 9.7 m to carry the removable head.
- The ellipsoidal bottom head, with a ratio of 2:1, has an inside height of approximately 6.5 m.
- The removable ellipsoidal top closure head has an inside height of approximately 4.7 m.
- The drywell shell height is approximately 30 m.
- The suppression chamber shell height is approximately 22 m.
- The overall shell height is approximately 52 m.

The primary containment vessel shell plate thicknesses vary, with typical thicknesses as follows:

- bottom ellipsoidal head: from 22.2 mm to 38.1 mm
- suppression chamber (wetwell) cylinder: from 33.3 mm to 38.1 mm
- drywell conical section: from 19.1 mm to 38.1 mm
- removable top ellipsoidal head: 23.8 mm.

The primary containment vessel is reinforced with internal vertical and horizontal stiffeners. Metal circumferential rings, attached to the inside face of the vessel, support pipe whip protection framework. The top closure head of the drywell is bolted to a steel flange attached to the top of the containment vessel. The vessel is provided with two concentric circular skirts on the bottom ellipsoidal head integral with the vessel. The skirts are anchor-bolted to the concrete foundation mat and are backed up by concrete fill.

The primary containment vessel contains 171 penetrations ranging in diameter from 19.1 mm to 3.81 m. Most of these penetrations also penetrate the concrete biological shield. A 3.81-m diameter penetration and a 2.69-m diameter penetration provide, respectively, equipment and personnel access to the drywell. A 1.07-m diameter penetration provides access to the wetwell

Table C.2-2 gives the size, quantity, general description, and general location of the rest of the primary containment vessel penetrations that penetrate the biological shield. Figure C.2-6 shows the specific location of the major penetrations through the primary containment vessel and the biological shield.

TABLE C.2-2. Primary Containment Vessel Penetration Information

Nominal Penetration Diameter, mm	Type, Location, and Quantity of Penetrations				Total for Each Diameter
	Piping		Instrument & Electrical		
	Drywell	Wetwell	Drywell	Wetwell	
20	2	--(a)	--	--	2
30	--	--	8	--	8
40	5	1	--	--	6
50	6	1	--	4	11
80	--	2	--	1	3
100	2	--	--	--	2
150	2	4	4	4	14
200	--	1	--	--	1
250	3	--	25	3	31
300	5	7	22	2	36
360	--	--	3	1	4
410	4	--	--	--	4
460	1	5	4	--	10
560	1	--	--	--	1
610	--	10	--	--	10
660	1	--	--	--	1
710	7	--	--	--	7
760	2	--	--	--	2
910	1	--	--	--	1
1 070	2	--	--	--	2
1 120	4	--	--	--	4
Totals	48	31	66	15	160

(a) A dash indicates no penetrations in that category.

FIGURE C.2-6. Major Primary Containment Vessel Penetrations

C.2.1.2 Primary Containment Vessel Internals

The major structural components inside the primary containment vessel are shown in Figure C.2-1, and are discussed here.

Reactor Pedestal. The reactor pedestal is a hollow, right-cylindrical reinforced concrete foundation that supports the reactor pressure vessel and the sacrificial shield wall. The bottom of the reactor pressure vessel skirt and the sacrificial shield wall are anchor-bolted directly to the top of the reactor pedestal. The bottom of the pedestal is keyed into the reinforced concrete liner inside the bottom head of the primary containment vessel and is tied directly to the reactor building foundation mat via the inner primary containment vessel skirt.

Except at haunches for the support of radial beams and at any other special structural features, the reactor pedestal has an inside diameter of 6.17 m and a shell thickness of 1.54 m. Its height is approximately 25.6 m. The shell is reinforced by horizontal hoop and vertical meridional reinforcing steel.

The inside and outside surfaces of the reactor pedestal are coated with a special decontaminable epoxy coating.

Sacrificial Shield Wall. The sacrificial shield wall is a vertical, cylindrical shell structure that sits atop the reactor pedestal and surrounds the lower two-thirds of the reactor pressure vessel. It is constructed of layers of inner and outer steel rings welded together and filled with concrete. Inner and outer skin plates and internal structural steel beam and box columns provide support and continuity between the ring layers and between the many penetrations that allow piping access to the reactor pressure vessel.

Except for the skin plates and other special structural aberrations, the sacrificial shield wall is 0.61 m thick and has a nominal inside diameter of 7.87 m. It has a height of approximately 14.7 m.

Drywell Floor. The drywell floor is a leak-tight pressure barrier dividing the primary containment vessel into a drywell portion above the floor and a suppression chamber (wetwell) below the floor directly under the drywell. The drywell floor structural system consists of:

- an outer annulus made of a 0.61-m thick reinforced concrete slab supported by structural steel beams, by reinforced concrete columns, and by the reinforced concrete reactor pedestal
- an inner circular reinforced concrete slab, inside the reactor pedestal, 1.52 m thick, lower in elevation than the outer slab by approximately 2.08 m, and constructed monolithically with, and supported by, the reactor pedestal.

Additional elements supporting the drywell floor are:

- a continuous circular closure girder embedded in the floor along its outer periphery
- a peripheral seal assembly
- shear lugs intermittently located along the outer periphery
- 17 reinforced concrete support columns equally spaced around the reactor pedestal in the suppression chamber.

Eighty-four 0.61-m and 18 0.71-m diameter downcomer vent pipes penetrate the drywell floor into the suppression chamber. The drywell floor is coated with a special decontaminable epoxy coating.

Radial Beam Framing Systems. Structural steel beams span radially from support points on the sacrificial shield wall and the reactor pedestal to the primary containment vessel to form radial beam framing systems at various elevations in both the suppression chamber and the drywell. They provide 1) supports for mechanical and electrical equipment, 2) platform areas for access to equipment, and 3) supports for pipe whip restraints.

Stabilizer Truss. The stabilizer truss is a circular steel truss with 16 horizontal members hinged at eight panel points at the top of the sacrificial shield wall and at eight panel points at the containment vessel wall by means of horizontal pin plate/gusset plate connections. The gusset plates are rigidly welded to the top of the sacrificial shield wall at the one end of each member and to the containment vessel at the other end.

Reinforced Concrete Base Lining. The reinforced concrete lining inside the bottom head of the primary containment vessel provide foundation and structural continuity for concrete structures in the suppression chamber, namely the reactor pedestal and the drywell floor support columns. The concrete liner is anchored to the primary containment vessel by means of headed stud shear connectors welded to the vessel bottom head.

C.2.2 Turbine Generator Building

The Turbine Generator Building, shown in Figures C.2-7 through C.2-10, is a modified non-Category I structure with two floors above grade. It is designed to provide adequate protection to the main steam lines that are designed as Seismic Category I from the outermost containment isolation valves in the Reactor Building up to the turbine main steam stop valves. The Turbine Generator Building houses the turbine generator, steam condensing equipment, moisture separator reheaters, and feedwater equipment. A 182-Mg capacity overhead crane services the Turbine Generator Building.

The building exterior consists of reinforced concrete covered with metal siding from the foundation up to the operating floor level, elevation 152.7 m. Above the operating floor, the exterior walls are insulated metal siding supported by structural steel. The roof is insulated metal decking with built-up roofing and is supported by steel trusses. The building is rectangular in plan (~58.8 m by ~91.4 m) and in elevation (~42.5 m high). The ground floor is at elevation 134.4 m.

The Turbine Generator Building structure and the reinforced concrete turbine generator pedestal are supported on a common foundation mat. The mat is constructed of reinforced concrete of thicknesses ranging between 2.74 m and 3.66 m. The mat top is at elevation 134.3 m. The mat reinforcing bar is placed in an orthogonal pattern top and bottom, similar to that of the Reactor Building mat (refer to Figure C.2-5).

Internal walls of the Turbine Generator Building include reinforced concrete walls, concrete block walls, and metal partitions. The turbine (EKN 201), the main condenser (EKN 200), the moisture separator reheaters (EKN 270), the feedwater heater system (EKN 253, 254) and the condenser gas removal and

EQUIPMENT KEY

200	MAIN CONDENSER
212	CONDENSATE PUMPS
213	CONDENSATE BOOSTER PUMPS
214	REACTOR FEEDWATER PUMPS
215	MECHANICAL VACUUM PUMPS
216	STEAM JET AIR EJECTORS
217	GLAND SEAL STEAM CONDENSER
218	CATALYTIC RECOMBINERS
226	SEAL WATER LIQUID TANK
227	PUMPED DRAIN TANK
228	CONDENSATE STORAGE TANKS
273	EQUIPMENT DRAIN SUMP
274	FLOOR DRAIN SUMP
700	ELEVATOR
703	RAILROAD TRACKS

FIGURE C.2-7. Turbine Generator Building
Floor Plan, Elevation 134.4 m

EQUIPMENT KEY

- 200. MAIN CONDENSER
- 224. OFF GAS CONDENSER
- 225. OFF GAS WATER SEPARATOR
- 228. CONDENSATE STORAGE TANKS
- 252. HIGH-PRESSURE FEEDWATER HEATERS
- 253. LOW-PRESSURE FEEDWATER HEATERS
- 254. STEAM EVAPORATORS
- 255. EVAPORATOR DRAIN TANKS
- 256. REHEATER DRAIN TANKS
- 258. TURBINE BYPASS VALVE ASSEMBLY
- 263. MOISTURE SEPARATOR DRAIN TANK
- 700. ELEVATOR
- 701. COVERED HATCH
- 702. OPEN HATCH

FIGURE C.2-8. Turbine Generator Building
Floor Plan, Elevation 143.6 m

EQUIPMENT KEY

- 201. TURBINE GENERATOR UNIT
- 202. MAIN STEAM AND FEEDWATER PIPE CHASE
- 253. LOW-PRESSURE FEEDWATER HEATERS
- 270. MOISTURE SEPARATOR REHEATER
- 276. HEATING AND VENTILATION UNIT
- 700. ELEVATOR
- 701. COVERED HATCH
- 702. OPEN HATCH

FIGURE C.2-9. Turbine Generator Building
Floor Plan, Elevation 152.7 m

SECTION A-A

SECTION B-B

EQUIPMENT KEY

- 200. MAIN CONDENSER
- 201. TURBINE GENERATOR UNIT
- 213. CONDENSATE BOOSTER PUMP
- 214. REACTOR FEEDWATER PUMPS
- 215. MECHANICAL VACUUM PUMP
- 216. STEAM JET AIR EJECTOR CONDENSER
- 217. GLAND SEAL STEAM CONDENSER
- 218. CATALYTIC RECOMBINERS
- 252. HIGH-PRESSURE FEEDWATER HEATERS
- 253. LOW-PRESSURE FEEDWATER HEATERS
- 254. STEAM EVAPORATORS
- 255. EVAPORATOR DRAIN TANKS
- 256. REHEATER DRAIN TANK
- 270. MOISTURE SEPARATOR REHEATER
- 272. LOUVER PENTHOUSES
- 276. HEATING AND VENTILATION UNITS

FIGURE C.2-10. Turbine Generator Building, Elevation Sections

handling equipment (EKN 216, 218) are surrounded by concrete walls at least 1.07 m thick for shielding purposes. The walls that surround the turbine generator area on elevation 152.7 m extend up to elevation 159.9 m.

There are two steel tanks for storage of condensate (EKN 228) adjacent to each other and enclosed in a diked retaining area located outside the Turbine Generator Building, as shown in Figures C.2-7 and C.2-8. The retaining area is a reinforced concrete, Seismic Category I structure consisting of a structural slab on soil and four perimetral walls. The retaining area is 18.6 m by 34.8 m in plan. The four perimetral walls are approximately 5.5 m high. The slab is 0.61 m thick and the walls are 0.46 m thick. Construction joints are keyed. Continuous waterstops are provided in all horizontal joints between the slab and walls.

C.2.3 Radwaste and Control Building

The Radwaste and Control Building is shown in Figures C.2-11 through C.2-13. The control room tower and the radwaste area are designed to Seismic Category I criteria. The other portions of the building are designed to less stringent standards. The Radwaste and Control Building contains the following areas associated with the daily plant operation: main control room; heating, ventilating and air conditioning rooms; cable spreading room; main reactor building power distribution center; emergency battery rooms; 4160-volt and 6900-volt switchgear and relay rooms; 480-volt substation; laboratory, instrument and computer rooms; and offices and facilities for shift operating personnel. Both safety related and nonsafety related systems are housed in the Radwaste and Control Building, some of which are: the condenser off gas filter system, the radioactive waste and chemical waste systems, the reactor water cleanup demineralizer system, the fuel pool cooling and cleanup demineralizer system, the condensate demineralizer system, and the fire protection system.

The Radwaste and Control Building is rectangular in shape, being approximately 63.7 m by 48.8 m in plan and 32 m in overall height from the top of the building foundation mat at elevation 133.2 m. The foundation mat, nominally 2.4 m thick and with orthogonal reinforcing similar to that in the reactor building foundation (refer to Figure C.2-5), is the only level of this building below the final plant grade. There are three main floor levels above this, and an additional partial level tops the control room tower.

RADWASTE AND CONTROL BUILDING -
FLOOR PLAN AT ELEVATION 142.3m

RADWASTE AND CONTROL BUILDING -
N. W. MEZZANINE PLAN AT ELEVATION 137.8m

RADWASTE AND CONTROL BUILDING -
FLOOR PLAN AT ELEVATION 133.2m

EQUIPMENT KEY

- 300 FLOOR DRAIN COLLECTOR TANK
- 301 WASTE COLLECTOR TANK
- 302 CONDENSATE PHASE SEPARATOR TANK
- 304 CONDENSATE BACKWASH RECEIVING TANK
- 305 SPENT RESIN TANK
- 306 WASTE SLUDGE PHASE SEPARATOR TANK
- 307 CLEANUP PHASE SEPARATOR TANK
- 308 CHEMICAL WASTE TANK
- 309 DISTILLATE TANK
- 310 DETERGENT DRAIN TANK
- 311 DECONTAMINATION SOLUTION CONCENTRATOR WASTE TANK
- 312 WASTE SURGE TANK
- 313 WASTE SAMPLE TANK
- 314 FLOOR DRAIN SAMPLE TANK
- 340 FLOOR DRAIN SUMP
- 341 EQUIPMENT DRAIN SUMP
- 342 CHEMICAL DRAIN SUMP
- 348 OFF GAS COOLER CONDENSER
- 351 DESICCANT DRYERS
- 352 REGENERATORS
- 353 OFF GAS COOLER
- 354 OFF GAS CHARCOAL ADSORBERS
- 355 OFF GAS CHILLER UNITS
- 358 SOLID RADWASTE STORAGE AREA CRANE
- 359 TRUCK LOADING CRANE
- 360 SOLID RADWASTE STORAGE
- 364 SOLID RADWASTE CONTAINER WASHDOWN
- 366 SOLID RADWASTE CONTAINER CAPPING STATION
- 367 SOLID RADWASTE CONTAINER FILLING STATION
- 371 SOLID WASTE BALER
- 377 EXPANSION TANK
- 378 AIR HANDLING UNIT
- 379 SOLID RADWASTE HOPPER MIXER
- 383 OFF GAS REFRIGERATION EQUIPMENT
- 384 PREFILTER
- 385 AFTERFILTER
- 407 EQUIPMENT REMOVAL PLUG
- 408 CONCENTRATOR WASTE MEASURING TANK
- 409 RADWASTE CENTRIFUGES
- 412 CHEMICAL SOLUTION TANK
- 413 CONDENSATE PRECOAT TANK
- 417 WASTE PRECOAT TANK
- 419 CHEMICAL ADDITION TANK
- 420 WASTE FILTER AID TANK
- 424 FUEL POOL PRECOAT TANK
- 425 RESIN ADDITION TANK
- 432 FUEL POOL HOLD PUMP
- 433 CLEANUP HOLD PUMP
- 435 DECONTAMINATION SOLUTION CONCENTRATOR
- 700 BOTTOM RECYCLE PUMP
- 700 ELEVATOR
- 701 COVERED HATCH
- 702 OPEN HATCH
- 708 SOLID RADWASTE CONTAINER DOLLY RAILS

KEY PLAN

FIGURE C.2-11. Radwaste and Control Building
Floor Plans - Sheet 1

RADWASTE AND CONTROL BUILDING -
CONTROL ROOM TOWER PLAN
AT ELEVATION 160.0 m

RADWASTE AND CONTROL BUILDING -
FLOOR PLAN AT ELEVATION 154.5 m

EQUIPMENT KEY

- 378. AIR HANDLING UNIT
- 386. DECONTAMINATION SOLUTION CONCENTRATOR
- 387. DECONTAMINATION SOLUTION CONCENTRATOR
- HEATING ELEMENT
- 407. EQUIPMENT REMOVAL PLUG
- 438. CONDENSATE FILTER DEMINERALIZER
- 439. FLOOR DRAIN FILTER
- 442. WASTE COLLECTOR FILTER
- 441. FUEL POOL FILTER DEMINERALIZER
- 442. CLEANUP FILTER DEMINERALIZER
- 443. WASTE DEMINERALIZER
- 444. FLOOR DRAIN DEMINERALIZER
- 445. DISTILLATE POLISHING DEMINERALIZER
- 447. DECONTAMINATION SOLUTION CONCENTRATOR
- EVAPORATOR
- 451. AIR HANDLING UNITS
- 453. FILTER UNITS
- 455. HEATING AND VENTILATION UNIT
- 464. MAIN CONTROL BENCHBOARD
- 465. RELAY CABINETS
- 468. DECONTAMINATION STATION
- 469. RESIN ADDITION TANK
- 470. CHILLERS
- 473. FILTER DEMINERALIZER REMOVAL CRANE
- 700. ELEVATOR
- 701. COVERED HATCH
- 702. OPEN HATCH

RADWASTE AND CONTROL BUILDING -
FLOOR PLAN AT ELEVATION 148.4 m

FIGURE C.2-12. Radwaste and Control Building
Floor Plans - Sheet 2

EQUIPMENT KEY

- 300. FLOOR DRAIN COLLECTOR TANK
- 301. WASTE COLLECTOR TANK
- 302. CONDENSATE PHASE SEPARATOR TANK
- 304. CONDENSATE BACKWASH RECEIVING TANK
- 305. SPENT RESIN TANK
- 306. WASTE SLUDGE PHASE SEPARATOR TANK
- 307. CLEANUP PHASE SEPARATOR TANK
- 308. CHEMICAL WASTE TANK
- 309. DISTILLATE TANK
- 310. DETERGENT DRAIN TANK
- 311. DECONTAMINATION SOLUTION CONCENTRATOR WASTE TANK
- 312. WASTE SURGE TANK
- 313. WASTE SAMPLE TANK
- 314. FLOOR DRAIN SAMPLE TANK
- 348. OFF GAS COOLER CONDENSER
- 353. OFF GAS COOLER
- 354. OFF GAS CHARCOAL ADSORBERS
- 355. OFF GAS CHILLER UNITS
- 358. SOLID RADWASTE STORAGE AREA CRANE
- 359. TRUCK LOADING CRANE
- 360. SOLID RADWASTE CONTAINER STORAGE
- 373. SOLID RADWASTE CONTAINER TRANSFER DOLLY
- 378. AIR HANDLING UNIT
- 383. OFF GAS REFRIGERATION EQUIPMENT
- 386. DECONTAMINATION SOLUTION CONCENTRATOR HEATING ELEMENT
- 387. DECONTAMINATION SOLUTION CONCENTRATOR HEATING ELEMENT
- 388. DECONTAMINATION SOLUTION CONCENTRATOR CONDENSATE RETURN TANK
- 401. TRANSFORMER
- 402. BATTERY RACK
- 406. ELECTRICAL SWITCHGEAR
- 407. EQUIPMENT REMOVAL PLUG
- 412. CHEMICAL SOLUTION TANK
- 413. CONDENSATE PRECOAT TANK
- 417. WASTE PRECOAT TANK
- 419. CHEMICAL ADDITION TANK
- 420. WASTE FILTER AID TANK
- 424. FUEL POOL PRECOAT TANK
- 425. RESIN ADDITION TANK
- 432. FUEL POOL HOLD PUMP
- 433. CLEANUP HOLD PUMP
- 435. DECONTAMINATION SOLUTION CONCENTRATOR BOTTOM RECYCLE PUMP
- 438. CONDENSATE FILTER DEMINERALIZER
- 439. FLOOR DRAIN FILTER
- 440. WASTE COLLECTOR FILTER
- 441. FUEL POOL FILTER DEMINERALIZER
- 442. CLEANUP FILTER DEMINERALIZER
- 443. WASTE DEMINERALIZER
- 444. FLOOR DRAIN DEMINERALIZER
- 445. DISTILLATE POLISHING DEMINERALIZER
- 447. DECONTAMINATION SOLUTION CONCENTRATOR EVAPORATOR
- 451. AIR HANDLING UNITS
- 453. FILTER UNITS
- 455. HEATING AND VENTILATION UNIT
- 457. DECONTAMINATION SOLUTION CONCENTRATOR CONDENSER
- 460. EXHAUST FILTER
- 464. MAIN CONTROL BENCHBOARD
- 465. RELAY CABINETS
- 469. RESIN ADDITION TANK
- 470. CHILLER
- 473. FILTER DEMINERALIZER REMOVAL CRANE
- 480. TURBINE GENERATOR BLDG. HVAC EXHAUST PLENUM
- 494. EXHAUST FANS
- 701. COVERED HATCH
- 702. OPEN HATCH

FIGURE C.2-13. Radwaste and Control Building Elevation Sections

Part of the Radwaste and Control Building is a cast-in-place reinforced concrete structure and part is a structural steel framed structure. The portions of the building that are reinforced concrete are arranged to form a completely enclosed box-like structure, and are composed of concrete walls on four exterior sides and of top and bottom concrete slabs. The remaining portions of the building are made up of steel framed floors and exterior walls.

The concrete portion of the building is a diaphragm slab/shear wall-type structure, supported on a foundation mat separated and apart from the mats under each of the other structures in the plant complex. The floor and roof slabs are continuous slab and beam systems monolithically constructed. All exterior concrete walls are covered with metal siding.

The steel framed portions of the building consist of conventionally braced, framed wall systems with vertical bracing in bays along each wall. The girts and roof purlins are covered with insulated metal siding and insulated metal roof decking, respectively. The steel framed floors are fabricated of structural steel beams with poured concrete slabs.

Interior walls of the Radwaste and Control Building are constructed of reinforced concrete, concrete block, or metal-studded drywall.

C.2.4 Diesel Generator Building

The Diesel Generator Building is located south of the Reactor Building and east of the Radwaste and Control Building, as shown in Figure C.1-1.

The Diesel Generator Building is rectangular in shape. It is approximately 48.5 m by 24.4 m in plan and 13.4 m in height from the ground floor at elevation 134.4 m.

The building is a Seismic Category I structure with reinforced concrete diaphragm slab/shear wall-type construction. It is supported on its own foundations, which are continuous wall footings, isolated spread footings for concrete columns, and isolated equipment foundations. The ground floor is a reinforced concrete slab.

The building has four major levels: the ground floor, the second floor, a third level consisting in part of a low roof area and in part of an enclosed floor area, and a fourth level, the high roof level. None of the floor levels are below the final plant grade.

The ground floor level is divided into four rooms separated by reinforced concrete walls. The easterly room, in turn, is divided by concrete walls extending above the ground floor slab into three subcompartments. The walls are monolithically connected to the ground floor slab. A diesel oil fuel tank is buried below each of the three subcompartments. Manholes in the ground floor slab, accessible from within each of the three subcompartments, provide access to the fuel tanks.

Three emergency diesel generators are located on the ground floor, each in a separate room. The second level houses air handling units, and the third level houses exhaust silencers.

C.2.5 Service Building

The Service Building is the last building in the main complex shown in Figure C.1-1 and is directly east of, but physically separated from, the Turbine Generator Building and the Reactor Building. It contains two stories above grade and a partial substructure. It projects ~25 m east of the Turbine Generator Building along 52 m of the east Turbine Generator Building wall and ~18 m east of the Reactor Building along 32 m of the east Reactor Building wall. It is ~10 m high above grade and ~6 m below grade.

Exterior above-grade walls of the Service Building are made of precast concrete panels. Below-grade construction is reinforced concrete. Interior walls in the area east of the Turbine Generator Building are largely steel-studded composition type, with the exception of a central quality assurance records vault that has concrete block walls. A reinforced concrete wall divides this area from the area east of the Reactor Building. All floors are reinforced concrete. The roof is made of insulated metal decking supported by steel trusses.

The Service Building area east of the Turbine Generator Building houses office, meeting, and storage facilities for the plant administrative and

engineering staff above grade, and equipment of the plant makeup water treatment system below grade. The above-grade area east of the Reactor Building contains a high-bay machine shop serviced by an overhead crane. The below-grade area is used for storage.

C.2.6 Cooling Tower Complex

Six circular mechanical-draft cooling towers, each 18.3 m high and 61.0 m in diameter, are located some 200 m south of the main building complex, as shown in Figure C.1-1. In the interior of the cooling tower complex are the Circulating Water Pumphouse and two electrical buildings.

Each cooling tower is constructed basically of precast concrete modules (louvers, circumferential panels, supports, and fan deck) on a cast-in-place concrete basin. Six 149-kW, 9.1-m diameter, multiblade fans, each surrounded by a fiberglass fan cylinder, sit on top of each cooling tower. Drift eliminators and splash bars are made of asbestos cement board.

The Circulating Water Pumphouse has a structural steel and insulated metal sided superstructure and a reinforced concrete substructure, and is rectangular in plan (19 m by 40 m) and in elevation (12 m above grade and 9 m below grade). The electrical buildings are made of structural steel with insulated metal siding and are also rectangular in plan (12 m by 15 m) and in elevation (5 m above grade). The foundation mat is reinforced concrete.

C.2.7 Spray Pond Complex

Two spray ponds and two Standby Service Water Pumphouses form a complex of Seismic Category I structures located about 200 m southeast of the main plant complex, as shown in Figure C.1-1. Each pond is constructed integrally with a pumphouse. Each pair of structures (pond and pumphouse) is adjacently located, but structurally separated.

The spray pond structures are rectangular (73 m by 74 m), reinforced concrete retention basins consisting essentially of a structural slab on soil and four perimetral exterior walls. The walls and floor slab are bounded by Quality Class I high-density backfill. Each pond is 4.6 m deep. The top of the perimetral walls and floor slab are respectively 0.3 m above and 4.3 m

below the final plant grade, which is at elevation 132.3 m in the immediate vicinity. The floor slab is 178 mm thick and is placed on top of a 51-mm unreinforced concrete subgrade leveling slab. The perimetral walls vary in thickness from 0.4 m at the top to 0.6 m at the bottom.

The pumphouses are rectangular in shape and are approximately 9.5 m by 18.3 m in plan, with the roof approximately 9.8 m above grade and the pump chamber foundation approximately 8.5 m below grade. They are cast-in-place reinforced concrete structures and are supported on foundations consisting of spread footings and foundation mats. Each structure has four levels: the foundation slabs at the bottom of the pump chambers at elevation 124.5 m, the electric vault and valve pit level, the main operating floor at grade-level, and the roof. The final grade in this area varies from elevations 132.8 m to 134.3 m.

C.2.8 Makeup Water Pumphouse

The Makeup Water Pumphouse is located on the bank of the river that runs through a corner of the site. The structure is constructed of reinforced concrete. It contains a pump pit substructure 6.7 m by 6.7 m inside in plan and 12.7 m in depth, and a superstructure approximately 23.5 m by 11.0 m inside in plan and 5.2 m in height to the top of the roof slab.

C.2.9 Auxiliary Buildings

This section contains structural information about the other buildings shown in Figure C.1-1, namely the Office Building, the Warehouse, the Guard House, and the Gas Bottle Storage Building.

Office Building

The Office Building is located approximately 100 m southwest of the Radwaste and Control Building. It is a structural steel building with insulated metal siding and concrete slab floor, and is approximately 20 m by 30 m in plan and 5 m high at the roof crown.

Warehouse

The Warehouse is located approximately 100 m south of the Radwaste and Control Building and is similar in construction to the Office Building. It is approximately 30 m by 60 m in plan and 5 m high at the roof crown.

Guard House

The Guard House is located directly east of the Service Building. The building is constructed of reinforced concrete below grade and precast concrete exterior above grade. The superstructure, which is approximately 15 m by 23 m in plan and 5 m in height, houses a central surveillance complex that is surrounded on all sides by 0.3-m thick reinforced concrete walls and slabs. This interior structure is 7.6 m by 10.1 m inside in plan. The substructure is 7.6 m by 23 m in plan and 5 m deep.

Gas Bottle Storage Building

The Gas Bottle Storage Building is about 125 m north of the Turbine Generator Building. It is a precast concrete structure with reinforced concrete foundation and floor slab, and is rectangular in plan (9 m by 8 m) and in elevation (4 m above grade). The floor is approximately 1 m above grade, at the same height as a 4-m by 4-m concrete loading dock adjoining the east side of the building.

C.3 CONTAMINATED EQUIPMENT

Descriptions of the contaminated equipment located in the Reactor Building, the Turbine Generator Building, and the Radwaste and Control Building are presented in this section.

C.3.1 Reactor Building Equipment

The Reactor Building houses the primary reactor system, reactor auxiliary and emergency cooling systems, and all the facilities required for refueling operations. It is assumed that the process equipment listed in Table C.3-1, which is located outside the primary containment vessel, is contaminated since it contacts (either directly or indirectly) the contaminated reactor water, steam, or resultant condensate. Descriptions of the reactor vessel and its internals, the reactor water recirculation pumps, and the spent fuel storage pool follow.

C.3.1.1 Reactor Vessel

The reactor vessel (EKN 1), shown in Figure C.3-1, is located inside the primary containment vessel (see Figure C.2-1). It contains the core and supporting

TABLE C.3-1. Contaminated Equipment: Reactor Building

EKN(a)	System Equipment Piece (Quantity)	Mass, (each) kg	Overall Dimensions, (each) L x W x H, m
<u>Containment Atmosphere Control</u>			
94	Hydrogen Recombiner (2)	5 443	3 4 x 2 7 x 2 3
<u>Condensate (Nuclear Steam)</u>			
16	Reactor Bldg Cond Supply Pump (1)	340	1 2 x 0 5 x 0 6
17	Radiaste Bldg Cond Supply Pump (1)	1 651	1 7 x 0 8 x 0 8
18	Condensate F/D Backwash Pump (1)	1 651	1 7 x 0 8 x 0 8
<u>Control Rod Drive (CRD)</u>			
20	CRD Pump Suction Filter (1)	181	0 6 dia x 1 2
21	CRD Pump (2)	2 586	3 7 x 1 2 x 1 2
46	CRD Module (185)	356	0 5 x 0 5 x 0 7
---(b)	Control Rod Drive (185)(c)	205	0 2 dia x 4 9 (retracted)
---	CRD Removal Platform (1)(c)	2 495	5 9 x 2 5 x 0 5
<u>Equipment Drain (Radioactive)</u>			
19	Equipment Drain Heat Exchanger (1)	680	0 3 dia x 2 4
98	Equipment Drain Sump Pump (1)	651	0 5 dia x 3 1
<u>Floor Drain (Radioactive)</u>			
97	Reactor Bldg Floor Drain Sump Pump (2)	524	0 5 dia x 3 0
97	Reactor Bldg Floor Drain Sump Pump (1)	651	0 5 dia x 3 1
97	Reactor Bldg Floor Drain Sump Pump (1)	651	0 5 dia x 3 1
97	Reactor Bldg Floor Drain Sump Pump (2)	524	0 5 dia x 3 0
<u>Fuel Pool Cooling and Cleanup</u>			
23	Suppression Pool Cleanup Pump (1)	528	1 5 x 0 6 x 0 6
55	Fuel Pool Circulation Pump (2)	528	1 5 x 0 6 x 0 6
56	Fuel Pool Heat Exchanger (2)	2 041	0 6 dia x 6 1
64	Skimmer Surge Tank (2)	3 606	1 8 dia x 7 6
<u>High-Pressure Core Spray (HPCS)</u>			
5	HPCS Pump (1)	27 443	1 5 dia x 4 3
7	HPCS Water Leg Pump (1)	398	1 5 x 0 6 x 0 6
<u>Low-Pressure Core Spray (LPCS)</u>			
4	LPCS Pump (1)	9 662	1 2 dia x 3 7
6	LPCS Water Leg Pump (1)	398	1 5 x 0 6 x 0 6
<u>MSIV Leakage Control (MSLC)</u>			
123	MSLC Exhaust Fan (2)	205	0 6 x 0 6 x 0 6
<u>Reactor Building Closed Cooling (RBCC)</u>			
50	RBCC Water Pump (3)	1 599	1 6 x 0 6 x 0 8
51	RBCC Water Heat Exchanger (3)	7 469	0 9 dia x 10 7
<u>Reactor Core Isolation Cooling (RCIC)</u>			
10	RCIC Pump (1)	6 350	3 1 x 1 2 x 1 2
11	RCIC Water Leg Pump (1)	398	1 5 x 0 6 x 0 6
12	RCIC Vacuum Tank (1)	408	0 6 x 0 9 x 0 6
13	RCIC Condenser Pump (1)	680	0 3 dia x 0 6
14	RCIC Vacuum Pump (1)	454	0 2 dia x 0 6
15	RCIC Barometric Condenser (1)	544	0 2 dia x 2 1
<u>Reactor Refueling</u>			
---	Refueling Platform	24 948	
---	Service Platform	5 216	
<u>Reactor Water Cleanup (RWCU)</u>			
47	RWCU Recirculation Pump (2)	590	1 5 x 0 6 x 0 6
61	RWCU Regenerative Heat Exchanger (3)	4 135	0 8 dia x 6 4
62	RWCU Non-Regenerative Heat Exchanger (2)	4 082	0 8 dia x 7 9
<u>Residual Heat Removal (RHR)</u>			
8	RHR Pump (3)	7 802	1 5 dia x 3 7
9	RHR Water Leg Pump (1)	398	1 5 x 0 6 x 0 6
54	RHR Heat Exchanger (2)	29 225	1 5 dia x 8 9
<u>Traversing In-Core Probe (TIP)</u>			
42	TIP Drive Mechanism (5)	362	1 2 x 0 8 x 0 9

(a)EKN = Equipment Key Number (see Section C.2)
 (b)Indicates no equipment key number is available
 (c)Located inside Primary Containment

FIGURE C.3-1. Reactor Vessel and Internals

structures; steam separators and dryers; coolant recirculation jet pumps; control rod guide tubes; distribution piping for feedwater, core sprays, and liquid control; in-core instrumentation; and other components. The main connections to the vessel include steam lines, coolant recirculation lines, feedwater lines, control rod drive and in-core nuclear instrumentation housings, high- and low-pressure core spray lines, residual heat removal lines, standby liquid control line, jet pump pressure-sensing lines, and water-level instrumentation.

The reactor vessel is a vertical, cylindrical pressure vessel of welded construction designed for a pressure of 8.72 MPa. The vessel is fabricated of SA533 carbon steel and is clad internally with Type 304 stainless steel (except for the top head, nozzles, and nozzle weld zones, which are unclad).

The vessel top head is secured to the reactor vessel by studs and nuts. The nuts are tightened with a stud tensioner. The vessel flanges are sealed with two concentric metal seal-rings.

The design parameters of the reactor vessel are presented in Table C.3-2.

C.3.1.2 Reactor Vessel Internals

Refer to Figure C.3-1 for a general assembly drawing of the important reactor components.

The reactor vessel internals include (exclusive of fuel, control rods, in-core nuclear instrumentation, and pressure boundary appurtenances welded to the vessel) the following components:

Core Shroud. The core shroud is a stainless steel cylindrical assembly that separates the core region from the outer annulus. The upper part of the shroud surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide below. The central portion of the shroud surrounds the fuel region and forms the longest section of the shroud. This section is bounded at the bottom by the core support plate. The lower portion of the shroud, surrounding part of the lower plenum, is welded to the reactor vessel shroud support.

TABLE C.3-2. Reactor Vessel Design Parameters

Parameter of Interest	Design Specification																																																			
Internal Height from Inside Bottom Head to Inside Top Head	22.23 m																																																			
Shell Height to Top of Closure Flange	18.92 m																																																			
CRD Housing Projection Below Inside Bottom Head	3.15 m																																																			
ID of Shell	6.375 m																																																			
Minimum Shell Wall Thickness	171 mm																																																			
Wall SS Cladding Thickness	3.18 mm																																																			
Bottom Head Thickness	203 mm																																																			
Closure Flange Thickness	330 mm																																																			
ID of Top Head Flange	6.223 m																																																			
Top Head Flange Thickness	432 mm																																																			
Top Head Thickness	114 mm																																																			
Number of Top Head Studs	108																																																			
OD of Top Head Studs	152 mm																																																			
Mass of Vessel with Top Head, Studs, Nuts, and Washers	748.5 Mg																																																			
Mass of Top Head	90.7 Mg																																																			
Mass of Vessel with Top Head and Internals	1 034.2 Mg																																																			
<u>Vessel Nozzles (Quantity)</u>	<table><tr><th><u>ID, mm</u></th><th><u>Elevation Above Inside Bottom Head, m</u></th><th><u>Azimuth</u></th></tr><tr><td>Level Instrumentation and Vessel Vent (1)</td><td>76 Top Head</td><td>--- (a)</td></tr><tr><td>Head Spray (1)</td><td>114 Top Head</td><td>---</td></tr><tr><td>Spare Head (1)</td><td>114 Top Head</td><td>---</td></tr><tr><td>Main Steam (4)</td><td>610 16.46</td><td>72°, 108°, 252°, 288°</td></tr><tr><td>Instrument (4)</td><td>38 15.21</td><td>20°, 160°, 200°, 340°</td></tr><tr><td>Instrument (2)</td><td>38 13.13</td><td>10°, 190°</td></tr><tr><td>Feedwater (6)</td><td>305 12.53</td><td>30°, 90°, 150°, 210°, 270°, 330°</td></tr><tr><td>Core Spray (2)</td><td>229 12.17</td><td>120°, 240°</td></tr><tr><td>Residual Heat Removal (3)</td><td>305 9.46</td><td>45°, 135°, 315°</td></tr><tr><td>Instrument (4)</td><td>38 9.30</td><td>20°, 160°, 200°, 340°</td></tr><tr><td>Recirculation Inlet (10)</td><td>279 4.60</td><td>30°, 60°, 90°, 120°, 150°, 210°, 240°, 270°, 300°, 330°</td></tr><tr><td>Recirculation Outlet (2)</td><td>533 4.38</td><td>0°, 180°</td></tr><tr><td>Jet Pump Instr. (2)</td><td>89 3.86</td><td>105°, 285°</td></tr><tr><td>Core Δp/SLC (1)</td><td>64 Bottom Head</td><td>---</td></tr><tr><td>Bottom Head Drain (1)</td><td>38 Bottom Head</td><td>---</td></tr><tr><td>CRD Housing (185)</td><td>127 Bottom Head</td><td>---</td></tr></table>	<u>ID, mm</u>	<u>Elevation Above Inside Bottom Head, m</u>	<u>Azimuth</u>	Level Instrumentation and Vessel Vent (1)	76 Top Head	--- (a)	Head Spray (1)	114 Top Head	---	Spare Head (1)	114 Top Head	---	Main Steam (4)	610 16.46	72°, 108°, 252°, 288°	Instrument (4)	38 15.21	20°, 160°, 200°, 340°	Instrument (2)	38 13.13	10°, 190°	Feedwater (6)	305 12.53	30°, 90°, 150°, 210°, 270°, 330°	Core Spray (2)	229 12.17	120°, 240°	Residual Heat Removal (3)	305 9.46	45°, 135°, 315°	Instrument (4)	38 9.30	20°, 160°, 200°, 340°	Recirculation Inlet (10)	279 4.60	30°, 60°, 90°, 120°, 150°, 210°, 240°, 270°, 300°, 330°	Recirculation Outlet (2)	533 4.38	0°, 180°	Jet Pump Instr. (2)	89 3.86	105°, 285°	Core Δp/SLC (1)	64 Bottom Head	---	Bottom Head Drain (1)	38 Bottom Head	---	CRD Housing (185)	127 Bottom Head	---
<u>ID, mm</u>	<u>Elevation Above Inside Bottom Head, m</u>	<u>Azimuth</u>																																																		
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Bottom Head Drain (1)	38 Bottom Head	---																																																		
CRD Housing (185)	127 Bottom Head	---																																																		

(a)Indicates "not applicable."

Shroud Support Plate. The shroud support plate is a circular stainless steel plate welded to the vessel wall, forming the base of the outer annulus. This support carries the weight of the core shroud, shroud head, peripheral fuel elements, neutron sources, core support plate, top fuel guide, steam separators, and the restraining effect of components attached to the support.

Core Support Plate. The core support plate consists of a circular stainless steel plate containing bored holes and stiffened with a rim and beam structure. The plate provides lateral support and guidance for the control rod guide tubes, in-core flux monitor guide tubes, peripheral fuel support, and neutron sources. The last two items are also supported vertically by the core support plate.

The entire assembly is bolted to a support ledge between the central and lower portions of the core shroud. Alignment pins that engage slots and that bear against the core shroud are used to correctly position the assembly before it is secured.

Top Fuel Guide. The top fuel guide is formed by a series of stainless steel beams fastened to a peripheral rim and joined at right angles to form square openings. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, one fuel assembly. Sockets are provided in the bottom of the beam intersections to anchor the in-core flux monitors and startup neutron sources. The rim of the top fuel guide rests on a ledge between the upper and central portions of the shroud. The top fuel guide has alignment pins that engage and bear against slots in the shroud to correctly position the assembly before it is secured. Lateral restraint is provided by wedge blocks between the top fuel guide and the shroud wall.

Control Rod Guide Tubes. The 185 control rod guide tubes extend from the top of the control rod drive housings up through holes in the core support plate. Besides being the guide for a control rod, each stainless steel tube vertically supports a four-lobed orificed fuel support piece and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the control rod drive housing penetrating and welded to the

stub tube that is welded to the reactor vessel bottom head. A thermal sleeve is inserted into the control rod drive housing from below, and is rotated to lock the control rod guide tube in place. A key is inserted into a locking slot in the bottom of the control rod drive housing to hold the thermal sleeve in position.

Jet Pumps. Two semicircular groups of 10 jet pumps are located in the outer annulus between the core shroud and the reactor vessel wall. Each stainless steel jet pump consists of an inlet riser pipe, driving nozzle, suction inlet, throat or mixing section, and diffuser with attachment braces (see Figure C.3-2). The driving nozzle, suction inlet, and throat are joined together as a removable unit. The diffuser, associated bracing, and riser pipe are permanently installed. The riser pipe is welded to the recirculation inlet nozzle thermal sleeve. The riser brace consists of cantilever beams extending from welded pads on the reactor vessel wall.

The jet pump nozzle entry section is connected to the riser by a metal-to-metal, spherical-to-conical seal joint. Contact is maintained by a hold-down clamp. The throat section is supported laterally by a bracket attached to the riser. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

Shroud Head and Steam Separator Assembly. The shroud head and steam separator assembly consists of a flange and dome onto which is welded an array of standpipes, with an axial-flow steam separator welded to the top of each standpipe. The entire assembly is stainless steel. It mounts on the flange at the top of the shroud cylinder, forming a non-gasketed joint, and is held in place by long shroud head hold-down bolts. These bolts extend above the steam separators for easy access during removal. The entire assembly is positioned on the shroud by vertical guide rods on the inside of the reactor vessel and locating pins on the shroud flange.

Steam Dryer Assembly. The stainless steel steam dryer assembly mounts in the reactor vessel above the shroud head and steam separator assembly. It consists of a series of vertical corrugated dryer vanes housed in perforated plates at the top of the assembly. A skirt extends down the bottom of

FIGURE C.3-2. Reactor Coolant Recirculation Jet Pump Assembly

the dryer assembly housing to the elevation of the steam separator standpipes. The assembly is positioned in the vessel by the same guide rods as the shroud head. The dryer assembly rests on support brackets extending from the vessel wall, and is locked into position by brackets attached to the underside of the reactor vessel top head.

Feedwater Spargers. The feedwater spargers, located in the outer annulus, are stainless steel headers containing flow holes. An individual sparger is welded to each feedwater nozzle thermal sleeve that, in turn, is welded to the safe end of the vessel nozzle. The header is shaped to conform to the inside curvature of the reactor vessel wall. Sparger end brackets are attached to vessel wall brackets to support the spargers.

Core Spray Lines. Two core spray lines, entering the reactor vessel through the two core spray nozzles, divide immediately inside the reactor vessel. The two halves of the first line are routed to opposite sides of the reactor vessel and are supported by clamps attached to the vessel wall. The lines are then routed downward in the outer annulus and pass through the upper shroud immediately below the flange. The ends of the two spargers are supported by brackets. The other core spray line is identical, except that it enters the opposite side of the vessel and the spargers are at a slightly different elevation inside the shroud.

Top Head Cooling Spray Nozzle. The top head cooling spray nozzle is flange-mounted to a mating flange on the top reactor vessel head nozzle.

Differential Pressure and Liquid Control Line. The differential pressure and liquid control line enters the reactor vessel as two concentric pipes at a point below the core shroud. In the lower plenum, the two pipes separate. The inner pipe terminates near the lower shroud with a perforated length below the core support plate. The outer pipe terminates immediately above the core support plate and senses the pressure in the region outside the fuel assemblies.

In-Core Flux Monitor Guide Tubes. These tubes position fixed neutron detectors in the core and provide a path for the traversing in-core probes. The in-core flux monitor guide tubes extend from the top of the in-core flux

monitor housing in the lower plenum to the top of the core support plate. A latticework of clamps, tie bars, and spacers gives lateral support and rigidity to the guide tubes. The bolts and clamps are welded.

Startup Neutron Sources. The startup neutron sources are held in place by spring pressure, and are located between the top of the core support plate and the bottom of the top fuel guide. Each source consists of two irradiated antimony rods within a single beryllium cylinder. Both the antimony and the beryllium are encased in stainless steel tubes.

C.3.1.3 Reactor Water Recirculation Pumps

Two reactor water recirculation pumps are part of two identical stainless steel reactor water recirculation loops located in the drywell external to the reactor vessel, as shown in Figure C.3-3. Each external loop contains one motor-driven recirculation pump, a flow control valve, two motor-operated shutoff valves, and a bypass around the discharge shutoff valve and the flow control valve. Each loop ends in an external manifold from which individual water inlet lines are routed to the jet pump risers within the reactor vessel. Table C.3-3 lists the design characteristics of the reactor water recirculation system.

The reactor water recirculation pump (EKN 33), shown in Figure C.3-4, is a single-suction, single-stage, double-volute, motor-driven pump. The pump drive motor is mounted on a carbon steel motor stand above the pump. Motor torque is transmitted to the pump impeller by a vertically mounted pump shaft connected to the motor shaft by a rigid coupling. The pump suction nozzle is at the bottom of the pump casing, and the discharge nozzle is at the side in the horizontal plane.

The stainless steel pump case is welded to the reactor recirculation piping. A stuffing box is attached to the pump case with case studs and nuts. The upper pump shaft components are installed in the stuffing box, which also contains a cooling jacket and inlet and outlet connection points for the cooling and seal-staging water systems. The impeller is mounted on a taper fit to the pump shaft and is retained axially by the impeller nut and cap-screw.

SIMPLIFIED SCHEMATIC

FIGURE C.3-3. Reactor Water Recirculation System

TABLE C.3-3. Reactor Water Recirculation System Design Parameters

<u>Parameter of Interest</u>	<u>Parameter Specification</u>
<u>External Piping Loops:</u>	
Number of Loops	2
<u>Pipe Sizes (nominal O.D.):</u>	
Pump Suction	0.61 m
Pump Discharge	0.61 m
Discharge Manifold	0.41 m
External Risers	0.30 m
Recirculation Bypass Line	0.25 m
Material	Austenitic 316 SS
<u>Recirculation Pump:</u>	
<u>Dimensions:</u>	
Width	3.25 m
Depth	3.57 m
Height (including motor)	6.17 m
<u>Mass:</u>	
Motor	27 215 kg
Pump and Accessories	16 375 kg
Unit Total	43 590 kg
<u>Motor:</u>	
Type	Type K, Vertical, Solid Shaft, Four- Pole Induction Motor, 1 787 rpm
Power	6 637 kW
Voltage	6 600 Volts AC, Three-phase
Frequency	60 Hz
Power Source	Startup Transformer
<u>Pump:</u>	
Type	Vertical, Double Volute, Single Suction, Single Stage
Rated Flow	179 m ³ /min
Rated Head	254 m
Rated Speed	1 787 rpm
Shaft Seals	Mechanical
Case Material	Austenitic 316 SS
<u>Flow Control Valve:</u>	
Type	Ball
Material	Austenitic 316 SS
Type Actuation	Hydraulic
Diameter	0.61 m
<u>Recirculation Loop Block Valve:</u>	
Number	2
Type	Gate Valve
Actuator	Motor
Material	Austenitic 316 SS
Diameter	0.61 m
<u>Recirculation Loop Bypass Valve:</u>	
Type	Right Angle
Actuator	Motor
Material	Austenitic 316 SS
Diameter	0.25 m

FIGURE C.3-4. Reactor Water Recirculation Pump

Besides being held in place by its placement in the recirculation loop piping, each recirculation pump motor is suspended from the overhead radial beam system by four spring-loaded hangers. Each pump case is semi-rigidly attached to the surrounding drywell structure by three mechanical snubbers.

C.3.1.4 Spent Fuel Storage Pool

The spent fuel storage pool (EKN 90) is located in the Reactor Building, as shown in Figures C.2-3 and C.2-4. The pool, shown in Figure C.3-5, is a Seismic Category I, reinforced concrete structure completely lined with 6.4-mm stainless steel plates. The liner plates are seamwelded together and are anchored to the surrounding concrete by reinforcing members (i.e., channels, I-beams, etc.) welded to the liner plates. The pool contains fuel storage racks with enough storage locations for 2658 fuel assemblies. The pool is filled with water to a depth of approximately 11.5 m (water volume: $\sim 1325 \text{ m}^3$). The water supply is the condensate supply system, with emergency makeup water provided by the standby service water system.

The fuel storage racks consist of cells that are square stainless steel tubes with B_4C plates between them. The tube walls are 3.2 mm thick, and the B_4C plates are 5.3 mm thick. The vertical tubes are welded to horizontal stainless steel grids, at the top and bottom, to form the tubes into racks and to maintain center-to-center spacing between the tubes. A 3.2-mm stainless steel plate is seal welded to the rack sides. The rack is welded to an elevated base plate that, in turn, is supported by a system of welded beams and stiffeners. The racks are welded together into modules held firmly in place by stainless steel turnbuckles attached top and bottom between the modules and the pool wall. The turnbuckles are connected at the pool wall to a steel band imbedded in the concrete wall and seal welded to the pool liner. Different rack sizes are used (12 by 16, 12 by 13, 8 by 13, 7 by 18, and 11 by 16 cell arrays) to take full advantage of the fuel storage space in the pool (see Figure C.3-5).

C.3.2 Turbine Generator Building Equipment

The Turbine Generator Building houses the equipment associated with the power conversion process. Aside from the electricity generating and distribution equipment, the turbine and generator auxiliary systems, and the auxiliary

Spent Fuel Storage Pool Arrangement

Typical Spent Fuel Storage Rack
(12 x 16 Cells)

FIGURE C.3-5. Spent Fuel Storage Facilities

boiler equipment, the equipment located in the Turbine Generator Building comes in contact (either directly or indirectly) with the steam produced in the reactor or with the resultant condensate in preparing the reactor feed-water and pumping it to the reactor. The major pieces of radioactively contaminated equipment in the Turbine Generator Building are the main turbine (EKN 201) and the main condenser (EKN 200), for which detailed descriptions follow. Table C.3-4 lists the rest of the equipment that requires special handling because of radioactive contamination.

C.3.2.1 Main Turbine

The main turbine (EKN 201), shown in Figure C.3-6, is a tandem-compound unit, consisting of one double-flow high-pressure turbine and three double-flow low-pressure turbines. Table C.3-5 presents the physical parameters of the main turbine components.

The high-pressure turbine is a double axial flow element with one control stage followed by seven stages of reaction blading in each end. The supply steam lines enter the high-pressure element through two steam chest assemblies, one located on each side of the turbine. The steam chest inlets are the four main steam lines coming from the reactor, and the outlets are four inlet pipes to the high-pressure turbine casing. Turbine exhaust lines exit each end, four in the base and two in the cover.

Each steam chest is a horizontal, hour-glass-shaped assembly containing two horizontally mounted inlet throttle valves and two vertically mounted outlet governing valves.

The high-pressure cylinder consists of an outer casing of carbon steel, split at the horizontal centerplane to form a base and cover. The base and cover are bolted together by stud bolts. The nozzle chambers are centrally located in the high-pressure unit, with the chamber inlets welded to the casing. The outer casing of the high-pressure turbine is supported by four arms at the top of the base.

The high-pressure turbine rotor is machined from an alloy steel forging. A separate extension shaft is bolted to the governor end of the rotor.

TABLE C.3-4. Contaminated Equipment: Turbine Generator Building

EKN(a)	System Equipment Piece (Quantity)	Mass, (each) kg	Overall Dimensions, (each) L x W x H, m
<u>Air Removal</u>			
215.	Mechanical Vacuum Pump (2)	3 175	0.6 x 0.8 x 0.8
<u>Condensate (Nuclear Steam)</u>			
212.	Condensate Pump (3)	21 909	2.6 dia x 5.9
213.	Condensate Booster Pump (3)	12 020	5.5 x 1.8 x 1.8
216.	Steam Jet Air Ejector (2)	6 622	5.8 x 1.5 x 3.0
217.	Gland Seal Steam Condenser (1)	4 037	0.8 dia x 3.1
228.	Condensate Storage Tank (2)	50 536	13.7 dia x 13.3
253.	Low-Pressure Feedwater Heater (3)	63 050	1.8 dia x 15.5
253.	Low-Pressure Feedwater Heater (3)	51 256	2.1 dia x 14.6
253.	Low-Pressure Feedwater Heater (3)	50 349	2.1 dia x 14.6
253.	Low-Pressure Feedwater Heater (3)	35 380	1.8 dia x 14.0
253.	Low-Pressure Feedwater Heater (2)	68 946	2.4 dia x 12.8
<u>Condenser Off Gas Treatment</u>			
218.	Catalytic Recombiner (2)	454	1.4 dia x 2.1
224.	Off Gas Condenser (1)	898	1.4 dia x 4.7
225.	Off Gas Water Separator (1)	272	0.3 dia x 1.7
<u>Equipment Drain (Radioactive)</u>			
273.	Equipment Drain Sump Pump (4)	586	0.5 dia x 2.4
<u>Floor Drain (Radioactive)</u>			
274.	Floor Drain Sump Pump (4)	484	0.5 dia x 2.4
<u>Heater Drain</u>			
255.	Evaporator Drain Tank (2)	885	0.9 dia x 1.2
256.	Reheater Drain Tank (4)	1 135	0.9 dia x 2.4
256.	Reheater Drain Tank (4)	6 282	1.2 dia x 2.4
263.	Moisture Separator Drain Tank (2)	1 712	1.4 dia x 2.7
<u>Main Steam</u>			
254.	Steam Evaporator (2)	13 476	1.8 dia x 10.7
258.	Turbine Bypass Valve Assembly (1)	5 270	5.3 x 0.9 x 3.3
270.	Moisture Separator Reheater (2)	208 653	3.9 dia x 28.3
<u>Miscellaneous Drain and Vent</u>			
226.	Seal Water Liquid Tank (1)	488	1.2 dia x 2.7
227.	Pumped Drain Tank (1)	655	1.7 dia x 4.6
<u>Reactor Feedwater</u>			
214.	Reactor Feedwater Pump (2)	54 886	9.5 x 4.0 x 3.7
252.	High-Pressure Feedwater Heater (2)	73 482	2.4 dia x 12.5

(a)EKN = Equipment Key Number (see Section C.2).

FIGURE C.3-6. Main Steam Turbine and Generator

TABLE C.3-5. Physical Parameters of the Main Turbine Components

<u>Component</u>	<u>Axial Length, m</u>	<u>Width, m</u>	<u>Height, m</u>	<u>Mass, kg</u>
<u>High-Pressure Turbine</u>				
Cover	5.0	3.8	3.7	58 967
#1 Blade Ring Half (4 used)	0.6	2.4 dia		4 536 (each)
#2 Blade Ring Half (4 used)	0.8	2.5 dia		7 711 (each)
Rotor with Extension Shaft	8.4	2.2 dia		52 163
Base	7.0	3.8	3.7	64 864
<u>Low-Pressure Turbine (3 each)</u>				
Outer Cover	9.4	8.6	3.7	58 287
#1 Inner Cover and Base	1.9	5.5	2.8	25 265 (each)
#2 Inner Cover and Base	3.9	6.2	2.8	45 495 (each)
Blade Ring Half (4 used)	0.5	3.3 dia		2 631 (each)
Rotor	8.8	4.5 dia		88 677
Outer Base	10.7	9.9	3.2	142 882
<u>Steam Chest (2 each)</u>	6.1	1.7	2.1	55 565 (each)

Each low-pressure turbine is a double axial flow element with eight stages of reaction blading in each end. Supply steam lines enter through the cover at the center of each element, and multiple steam extraction lines exit the base. Exhaust openings at each end go downward to the main condenser, which is located directly beneath the three low-pressure turbines.

The low-pressure turbine rotors, consisting of a series of alloy steel discs shrunk on a shaft and keyed in position, are machined from alloy steel forgings.

Each low-pressure turbine has a fabricated outer casing of carbon steel. Each outer cover is removable as a single piece. The casing is supported by

a continuous foot (or skirt) extending around the cylinder base. The foot of each casing rests on a separate seating plate grouted to the foundation.

Flanged, rigid couplings are used to connect the rotors of the high-pressure turbine, the three low-pressure turbines (with two stub shafts between), and the generator. The rotating element thus formed is supported by 10 journal bearings, and is located axially by a thrust bearing mounted at the governor end of the No. 2 low-pressure turbine.

C.3.2.2. Main Condenser

The main condenser (EKN 200) is shown in Figure C.3-7. It is a single-pass, three-bank, divided-waterbox type surface condenser built for service in conjunction with the main turbine under which it is located. It was fabricated in place.

The condenser shell is of welded carbon steel plate. The tube supports and waterboxes are likewise carbon steel. Tube sheets are constructed of a copper-zinc alloy (Admiralty metal), as are the majority (93%) of the approximately 48,600 32-mm O.D. tubes. The peripheral tubes are formed from a copper-nickel alloy. The dry mass of the assembled condenser is about 1570 Mg.

A low-pressure feedwater heater is mounted in each of the three steam inlets of the condenser shell. Steam extraction piping is included in the condenser necks. Rubber belt expansion joints connect the turbine exhausts to the steam inlets of the condenser.

The tube sheets are bolted in place at the ends of each shell. The tubes are expanded into the tube sheets and are held in position by 84 tube supports welded to support lugs and support pipes, which in turn are welded to the shell proper. The inlet, outlet, and intermediate water boxes are bolted to their respective tube plates. The shell sits on six foundation pads, three on each side. It is anchored to the south-central pad by 50-mm diameter stud bolts, and is free to move on the other five pads. The hotwell, located in the bottom of the shell, has a storage capacity of 617 m³.

Numerous penetrations through the condenser shell or into the water boxes accommodate (among other things) turbine bypass steam, exhaust from the

FIGURE C.3-7. Main Condenser

feedwater pump turbines, cascading feedwater heater drains, feedwater heater shell steam, and condensate pump suction. Manways allow physical access to the interior of the condenser and the water boxes.

C.3.3 Radwaste and Control Building Equipment

The Radwaste and Control Building houses equipment associated with the liquid and solid radwaste systems, the condenser off gas treatment system, emergency electrical and control systems, and the main control room. The equipment inside of the control tower on the three highest floor elevations is assumed to be uncontaminated radioactively. Outside of the control tower, most process equipment, with few exceptions, is assumed to be radioactively contaminated. Table C.3-6 lists the contaminated equipment.

C.3.4 Contaminated Piping and Valves

The following tables list the piping and valves in plant systems that handle contaminated contents: Table C.3-7, Reactor Building; Table C.3-8, Primary Containment; Table C.3-9, Turbine Generator Building; Table C.3-10, Radwaste and Control Building; and Table C.3-11, summary of the entire plant. The primary flows in these systems are contaminated to a lesser or greater degree, although the auxiliary and utility flows may not be. However, the data in these tables do not make this distinction; all piping and valves designated by the facility owner as part of a specific system are included.

TABLE C.3-6. Contaminated Equipment: Radwaste and Control Building

EKN(a)	System Equipment Piece (Quantity)	Mass, (each) kg	Overall Dimensions, (each) L x W x H, m
<u>Condensate Filter Demineralizer</u>			
302.	Condensate Phase Separator Tank (2)	3 182	4.6 dia x 5.2
304.	Condensate Backwash Receiver Tank (1)	6 920	3.7 dia x 6 4
320.	Condensate Sludge Discharge Mixing Pump (1)	420	1 5 x 0 6 x 0 5
321.	Condensate Decant Pump (1)	420	1.5 x 0.6 x 0.5
322.	Condensate Backwash Transfer Pump (1)	420	1.5 x 0.6 x 0.5
429.	Condensate Demin. Holding Pump (6)	159	1.1 x 0.4 x 0 5
438.	Condensate Filter Demineralizer (6)	5 307	2.1 dia x 3.2
<u>Condenser Off Gas Treatment</u>			
348.	Cooler Condenser (2)	907	0.6 dia x 2.4
349.	Moisture Separator (2)	1 361	0.3 dia x 1.8
351.	Desiccant Dryer (4)	623	0.9 dia x 1 2
352.	Dryer Regenerator (2)	3 629	3.1 x 2.7 x 1.8
353.	Gas Cooler (2)	2 268	3.7 x 0.5 x 5.5
354.	Charcoal Adsorber (5.7 m ³ charcoal)(8)	4 082	1 2 dia x 6.9
378.	Air Handling Unit (4)	2 620	3.1 x 1.8 x 2 9
384.	Prefilter (2)	1 134	0.9 dia x 3.2
385.	Afterfilter (2)	1 134	0.9 dia x 3.2
<u>Equipment Drain (Radioactive)</u>			
301.	Waste Collector Tank (1)	10 241	4.9 dia x 5.5
305.	Spent Resin Tank (1)	658	1 8 dia x 1 8
312.	Waste Surge Tank (1)	18 304	7.6 dia x 6.1
313.	Waste Sample Tank (2)	6 968	4.5 dia x 4.4
319.	Waste Collector Pump (1)	284	1 4 x 0.5 x 0.8
323.	Waste Surge Pump (1)	284	1.4 x 0.5 x 0.8
324.	Waste Sample Pump (2)	231	1.3 x 0 5 x 0 7
327.	Spent Resin Pump (1)	102	1 1 x 0.4 x 0.5
341.	Equipment Drain Sump Pump (1)	667	0.5 dia x 2.4
431.	Waste Filter Hold Pump (1)	318	1.0 x 0 5 x 0.5
440.	Waste Collector Filter (1)	1 814	1.2 dia x 3 4
443.	Waste Demineralizer (1)	908	1.5 dia x 3 1
<u>Floor Drain (Radioactive)</u>			
300.	Floor Drain Collector Tank (1)	10 241	4 9 dia x 5.5
306.	Waste Sludge Phase Separator Tank (1)	5 497	4.3 dia x 3.4
314.	Floor Drain Sample Tank (1)	6 968	4.5 dia x 4.4
318.	Floor Drain Collector Pump (1)	284	1 4 x 0.5 x 0.8
326.	Floor Drain Sample Pump (1)	231	1.3 x 0.5 x 0.7
328.	Waste Decant Pump (1)	102	1.1 x 0.4 x 0.5
330.	Waste Sludge Discharge Mixing Pump (1)	288	1.4 x 0.5 x 0 5
340.	Floor Drain Sump Pump (3)	646	0.5 dia x 2.4
430.	Floor Drain Filter Hold Pump (1)	318	1 0 x 0.5 x 0 5
439.	Floor Drain Filter (1)	1 814	1.2 dia x 3.4
444.	Floor Drain Demineralizer (1)	908	1 5 dia, x 3.1

(a)EKN = Equipment Key Number (see Section C 2).

TABLE C.3-6. Contaminated Equipment: Radwaste and Control Building (Contd)

EKN(a)	System Equipment Piece (Quantity)	Mass, (each) kg	Overall Dimensions, (each) L x W x H, m
Fuel Pool Cooling and Cleanup			
432.	Fuel Pool Hold Pump (2)	195	1.1 x 0.3 x 0.5
441.	Fuel Pool Filter Demineralizer (2)	1 568	0.9 dia x 4.0
Miscellaneous Waste (Radioactive)			
308.	Chemical Waste Tank (2)	5 030	3.7 dia x 5.5
309.	Distillate Tank (2)	5 030	3.7 dia x 5.5
310.	Detergent Drain Tank (2)	1 836	1.8 dia x 2.4
311.	Decon. Solution Conc. Waste Tank (2)	712	1.7 dia x 1.7
331.	Concentrator Feed Pump (2)	254	1.1 x 0.3 x 0.6
332.	Chemical Waste Pump (2)	479	1.3 x 0.4 x 0.5
333.	Distillate Tank Pump (2)	231	1.2 x 0.3 x 0.4
334.	Detergent Drain Pump (2)	175	1.0 x 0.4 x 0.5
335.	Decon. Solution Conc. Waste Pump (1)	254	1.1 x 0.4 x 0.5
342.	Chemical Drain Sump Pump (1)	667	0.5 dia x 2.4
376.	Detergent Drain Filter (1)	1 134	0.3 dia x 0.9
386.	Decontamination Solution Concentrator (2)	3 409	1.2 dia x 4.6
387.	Decon. Solution Conc. Heating Element (2)	3 147	0.9 dia x 4.6
408.	Concentrator Waste Measuring Tank (1)	386	0.8 dia x 2.4
435.	Decon. Solution Conc. Bottoms Recycle Pump (2)	844	1.7 x 0.6 x 1.1
445.	Distillate Polishing Demineralizer (1)	454	1.5 dia x 3.1
457.	Decon. Solution Conc. Condenser (2)	2 308	0.9 dia x 3.4
Process Waste (Radioactive)			
371.	Solid Waste Hydraulic Baler (1)	908	1.3 x 0.8 x 2.7
373.	Transfer Dolly (1)	1 134	1.8 x 1.8 x 1.8
379.	Hopper Mixer (2)	2 727	2.3 x 1.5 x 2.7
380.	Waste Processing Pump (1)	136	1.0 x 0.3 x 0.4
409.	Centrifuge (2)	454	2.7 x 1.1 x 1.7
Reactor Water Cleanup			
307.	Cleanup Phase Separator Tank (2)	2 041	3.1 dia x 2.7
329.	Cleanup Decant Pump (1)	102	1.1 x 0.4 x 0.4
337.	Cleanup Sludge Discharge Mixing Pump (1)	284	1.2 x 0.5 x 0.6
426.	Cleanup Precoat Tank (1)	227	0.8 dia x 1.2
427.	Cleanup Precoat Pump (1)	454	1.3 x 0.4 x 0.7
433.	Cleanup Hold Pump (2)	535	1.3 x 0.5 x 0.7
442.	Cleanup Filter Demineralizer (2)	3 175	0.9 dia x 2.7

(a)EKN = Equipment Key Number (see Section C.2)

TABLE C.3-7. Contaminated Piping and Valves: Reactor Building

		Nominal Outside Diameter, mm																
Material	System	≤60		73 to 254			305 to 406			457 to 610			660 to 762			Total		
Aluminum	Condenser Off Gas Treatment	---(a)		31/	1 020/	0 ^(b)	---			---			---			31/	1 020/	0
Total		---		31/	1 020/	0	---			---			---			31/	1 020/	0
Carbon Steel	Containment Atmosphere Control	131/	646/N.A. ^(c)	324/	5 076/	16	---			---			---			455/	5 722/	16 ^(d)
	Containment Purge Exhaust	3/	14/N.A.	---			---			59/	6 929/	2	15/	2 755/	2	77/	9 698/	4
	Condensate (Nuclear Steam)	126/	520/N.A.	520/	10 172/	33	42/	3 627/	0	47/	6 357/	0	---			735/	20 676/	33
	Condensate Filter Demineralizer	---		43/	1 205/	0	---			---			---			43/	1 205/	0
	Control Rod Drive	946/	3 455/N.A.	143/	6 731/	8	3/	485/	0	---			---			1 092/	10 671/	8
	Equipment Drain (Radioactive)	3/	9/N.A.	109/	1 219/	7	---			---			---			112/	1 228/	7
	Floor Drain (Radioactive)	10/	51/N.A.	380/	5 455/	6	---			---			---			390/	5 506/	6
	Fuel Pool Cooling and Cleanup	83/	317/N.A.	438/	19 099/	35	---			---			---			521/	19 416/	35
	High-Pressure Core Spray	61/	202/N.A.	17/	735/	9	113/	26 538/	5	25/	3 825/	3	---			216/	31 300/	17
	Low-Pressure Core Spray	156/	660/N.A.	28/	643/	6	79/	7 911/	4	24/	3 679/	1	---			287/	12 893/	11
	Main Steam	4/	7/N.A.	13/	293/	2	---			---			40/	12 095/	4	57/	12 395/	6
	MSIV Leakage Control	1/	1/N.A.	118/	1 937/	0	---			---			---			119/	1 938/	0
	Reactor Building Closed Cooling	184/	445/N.A.	578/	17 423/	26	73/	5 571/	4	41/	4 332/	0	---			876/	27 771/	30
	Reactor Core Isolation Cooling ^(e)	307/	1 043/N.A.	385/	20 154/	19	---			---			---			692/	21 197/	19
	Reactor Feedwater	2/	4/N.A.	14/	493/	1	---			13/ 6 891/ 4			---			29/	7 388/	5
	Reactor Water Cleanup	49/	118/N.A.	368/	10 779/	18	---			---			---			417/	10 897/	18
	Residual Heat Removal	213/	750/N.A.	444/	8 954/	22	195/	17 765/	10	679/	87 563/	29	---			1 531/	115 032/	61
	Standby Gas Treatment	44/	237/N.A.	---			---			64/ 7 584/ 16			---			108/	7 821/	16
Totals		2 323/	8 479/N.A.	3 922/	110 368/	208	505/	61 897/	23	952/	127 160/	55	55/	14 850/	6	7 757/	322 754/	292
Stainless Steel	Condensate (Nuclear Steam)	15/	54/N.A.	67/	575/	0	54/	2 143/	0	---			---			136/	2 772/	0
	Control Rod Drive	6 136/	18 545/N.A.	---			---			---			---			6 136/	18 545/	0
	Fuel Pool Cooling and Cleanup	1/	1/N.A.	109/	291/	7	---			---			---			110/	292/	7
	Miscellaneous Wastes (Radioactive)	17/	74/N.A.	324/	3 685/	3	---			---			---			341/	3 759/	3
Totals		6 169/	18 674/N.A.	500/	4 551/	10	54/	2 143/	0	---			---			6 723/	25 368/	10

(a) Dashed line represents insignificant amount.
(b) The data are presented in the following format: Piping Length, m/Piping Mass, kg/Number of Valves.
(c) N.A. = not available.
(d) The total number of valves for a particular system does not include small-bore valves 51 mm or less in diameter.
(e) Includes a 51-mm diameter drain going to the main condenser (EKN 200) in the Turbine Generator Building.

TABLE C.3-7. Contaminated Piping and Valves: Reactor Building

TABLE C.3-8. Contaminated Piping and Valves: Primary Containment

		Nominal Outside Diameter, mm																	
Material	System	≤ 60		73 to 254			305 to 406			457 to 610			660 to 762			Total			
Carbon Steel	Downcomer Vent	--- (a)		---			---			1 178/	520 546/	0 (b)	252/	95 772/	0	1 430/	616 318/	0	
	Equipment Drain (Radioactive)	6/	19/N.A. (c)	180/	2 658/	2	---			---			---			186/	2 677/	2 (d)	
	Floor Drain (Radioactive)	25/	136/N.A.	48/	602/	0	---			---			---			73/	738/	0	
	Fuel Pool Cooling and Cleanup	11/	185/N.A.	---			---			---			---			11/	185/	0	
	High-Pressure Core Spray	---		1/	50/	0	23/	3 059/	2	---			---			24/	3 109/	2	
	Low-Pressure Core Spray	---		---			25/	2 795/	2	---			---			25/	2 795/	2	
	Main Steam	23/	113/N.A.	424/	47 306/	36	---			---			122/	49 540/	4	569/	96 959/	40	
	Reactor Building Closed Cooling	112/	573/N.A.	334/	8 715/	31	---			---			---			446/	9 288/	31	
	Reactor Core Isolation Cooling	1/	3/N.A.	52/	2 229/	3	---			---			---			53/	2 232/	3	
	Reactor Feedwater	1/	1/N.A.	---			64/	9 923/	0	40/	28 410/	4	---			105/	38 334/	4	
	Reactor Recirculation	21/	78/N.A.	5/	62/	0	---			---			---			26/	140/	0	
	Reactor Water Cleanup	58/	248/N.A.	38/	1 538/	0	---			---			---			96/	1 786/	0	
	Residual Heat Removal	5/	10/N.A.	2/	21/	0	99/	13 983/	10	21/	5 921/	1	---			127/	19 935/	11	
Totals		263/	1 366/N.A.	1 084/	63 181/	72	211/	29 760/	14	1 239/	554 877/	5	374/	145 312/	4	3 171/	794 496/	95	
Stainless Steel	Control Rod Drive	3 840/	10 573/	0	---			---			---			---			3 840/	10 573/	0
	Equipment Drain (Radioactive)	1/	2/N.A.	6/	105/	0	---			---			---			7/	107/	0	
	Fuel Pool Cooling and Cleanup	---		4/	70/	0	---			---			---			4/	70/	0	
	Miscellaneous Wastes (Radioactive)	1/	5/N.A.	66/	1 063/	6	---			---			---			67/	1 068/	6	
	Reactor Recirculation	8/	23/N.A.	34/	2 173/	2	61/	81 391/	0	55/	21 440/	6	---			158/	32 027/	8	
	Residual Heat Removal	---		---			3/	398/	3	---			---			3/	398/	3	
Totals		3 850/	10 603/N.A.	110/	3 411/	8	64/	8 789/	3	55/	21 440/	6	---			4 079/	44 243/	17	

(a) Dashed line represents insignificant amount.

(b) The data are presented in the following format: Piping Length, m/Piping Mass, kg/Number of Valves.

(c) N.A. = not available.

(d) The total number of valves for a particular location does not include small-bore valves 51 mm or less in diameter.

TABLE C.3-8. Contaminated Piping and Valves: Primary Containment

TABLE C.3-9. Contaminated Piping and Valves: Turbine Generator Building

		Nominal Outside Diameter, mm																			
Material	System	≤ 60		73 to 254			305 to 406			457 to 610			660 to 762			914 to 1829			Total		
Aluminum	Air Removal	--- ^(a)		65/	5 621/	6 ^(b)	63/	9 577/	1	---			---			---			128/	15 198/	7 ^(c)
	Condenser Off Gas Treatment	56/	146/N.A. ^(d)	32/	1 046/	4	53/	3 139/	0	---			---			---			141/	4 331/	4
	Heater Vent	353/	1 566/N.A.	736/	15 512/	49	---			---			---			---			1 089/	17 078/	49
	Reactor Feedwater	---		14/	1 270/	1	---			---			---			---			14/	1 270/	1
Totals		409/	1 712/N.A.	847/	23 449/	60	116/	12 716/	1	---			---			---			1 372/	37 877/	61
Carbon Steel	Air Removal	28/	70/N.A.	58/	3 326/	5	70/	5 732/	0	---			---			---			156/	9 128/	5
	Bleed (Extraction) Steam	414/	1 693/N.A.	174/	7 257/	21	361/	36 265/	17	688/	70 102/	33	---			---			1 617/	115 317/	71
	Condensate (Nuclear Steam) ^(e)	784/	2 480/N.A.	488/	8 387/	40	250/	27 498/	12	210/	53 051/	48	53/	19 345/	0	115/	41 438/	3	1 900/	152 317/	71
	Condensate Filter Demineralizer	98/	165/N.A.	48/	1 655/	1	---			---			---			---			146/	1 820/	1
	Equipment Drain (Radioactive)	66/	363/N.A.	30/	516/	0	---			---			---			---			96/	879/	0
	Exhaust Steam	---		---			---			---			---			47/	19 814/	2	47/	19 814/	2
	Floor Drain (Radioactive)	150/	826/N.A.	76/	870/	0	---			---			---			---			226/	1 696/	0
	Heater Drain	823/	3 147/N.A.	977/	50 465/	82	712/	62 868/	31	331/	44 704/	5	---			---			2 843/	161 184/	118
	Heater Vent	362/	1 480/N.A.	321/	15 897/	13	30/	2 316/	0	158/	20 741/	0	---			---			871/	40 434/	13
	Main Steam	70/	327/N.A.	269/	16 359/	13	91/	13 806/	4	241/	64 730/	1	387/	200 742/	0	397/	173 630/	6	1 455/	469 594/	24
	Miscellaneous Drain and Vent	525/	3 575/N.A.	151/	4 780/	14	---			---			---			---			676/	8 355/	14
	Reactor Feedwater	16/	27/N.A.	40/	6 013/	5	133/	28 115/	5	224/	132 993/	9	25/	20 611/	0	---			438/	187 759/	19
Totals		3 336/	14 153/N.A.	2 632/	115 525/	194	1 647/	176 600/	69	1 832/	386 321/	96	465/	240 698/	0	559/	234 882/	11	10 471/	1 168 179/	370
Stainless Steel	Condensate (Nuclear Steam) ^(e)	---		38/	1 474/	0	103/	6 421/	0	---			---			---			141/	7 895/	0
Total		---		38/	1 474/	0	103/	6 421/	0	---			---			---			141/	7 895/	0

(a) Dashed line represents insignificant amount.

(b) The data are presented in the following format: Piping Length, m/Piping Mass, kg/Number of Valves.

(c) The total number of valves for a particular system does not include small-bore valves 51 mm or less in diameter.

(d) N.A. = not available.

(e) Includes piping and valves that connect to the condensate storage tanks (EKN 228) located outside the Turbine Generator Building.

TABLE C.3-9. Contaminated Piping and Valves: Turbine Generator Building

TABLE C.3-10. Contaminated Piping and Valves: Radwaste and Control Building

		Nominal Outside Diameter, mm																				
Material	System	≤ 60		73 to 254			305 to 406			457 to 610			660 to 762			914			Total			
Aluminum	Condensor Off Gas Treatment	19/	288/N.A.(a,b)	487/	17 841/	22	10/	920/	0	--- (c)			---			---			516/	19 049/	22 ^(d)	
Totals		19/	288/N.A.	487/	17 841/	22	10/	920/	0	---			---			---			516/	19 049/	22	
Carbon Steel	Condensate (Nuclear Steam)	1 367/	3 895/N.A.	548/	11 402/	59	329/	28 146/	12	12/	4 584/	3	---			99/	29 410/	0	2 355/	77 437/	74	
	Condensate Filter Demineralization	91/	233/N.A.	358/	11 436/	23	8/	1 005/	0	---			---			---			457/	12 674/	23	
	Condenser Off Gas Treatment	29/	104/N.A.	86/	2 430/	0	1/	70/	0	---			---			---			116/	2 604/	0	
	Equipment Drain (Radioactive)	257/	958/N.A.	470/	8 820/	47	---			---			---			---			727/	9 778/	47	
	Floor Drain (Radioactive)	273/	1 110/N.A.	600/	10 127/	52	---			---			---			---			873/	11 237/	52	
	Fuel Pool Cooling and Cleanup	97/	340/N.A.	299/	10 369/	24	---			---			---			---			396/	10 709/	24	
	Miscellaneous Wastes (Radioactive)	484/	2 193/N.A.	158/	2 192/	7	---			---			---			---			642/	4 385/	7	
	Process Vents (Radioactive)	80/	221/N.A.	141/	3 671/	2	---			---			---			---			221/	3 892/	2	
	Process Waste (Radioactive)	121/	266/N.A.	38/	475/	0	---			---			---			---			159/	741/	0	
	Radwaste Building Closed Cooling	40/	88/N.A.	288/	7 532/	5	---			---			---			---			328/	7 620/	5	
	Reactor Water Cleanup	248/	859/N.A.	351/	7 324/	39	---			---			---			---			599/	8 183/	39	
	Totals		3 087/	10 267/N.A.	3 337/	75 778/	258	338/	29 221/	12	12/	4 584/	3	---			99/	29 410/	0	6 873/	149 260/	273
	Stainless Steel	Condensate (Nuclear Steam)	---		153/	2 050/	0	---			---			---			---			153/	2 050/	0
		Condenser Off Gas Treatment	283/	540/N.A.	---			---			---			---			---			283/	540/	0
		Equipment Drain (Radioactive)	74/	380/N.A.	129/	1 298/	23	---			---			---			---			203/	1 678/	23
Floor Drain (Radioactive)		56/	294/N.A.	98/	1 034/	20	---			---			---			---			154/	1 328/	20	
Miscellaneous Wastes (Radioactive)		707/	3 399/N.A.	637/	5 701/	53	55/	1 756/	0	---			---			---			1 399/	10 856/	53	
Process Waste (Radioactive)		30/	134/N.A.	9/	81/	0	---			---			---			---			39/	215/	0	
Totals		1 150/	4 747/N.A.	1 026/	10 164/	96	55/	1 756/	0	---			---			---			2 231/	16 667/	96	

(a) The data are presented in the following format: Piping Length, m/Piping Mass, kg/Number of Valves.
(b) N.A. = not available.
(c) Dashed line represents insignificant amount.
(d) The total number of valves for a particular system does not include small-bore valves 51 mm or less in diameter.

TABLE C.3-10. Contaminated Piping and Valves: Radwaste and Control Building

TABLE C.3-11. Summary of Contaminated Piping and Valves

		Nominal Outside Diameter, mm																			
Material	Location	≤ 60		73 to 254			305 to 406			457 to 610			660 to 762			914 to 1 829			Total		
Aluminum	Reactor Building	---		31/ 1 020/ 0 ^(b)			---			---			---			---			31/ 1 020/ 0		
	Turbine Generator Building	409/	1 712/N.A. ^(c)	847/	23 449/	60	116/	12 716/	1	---	---	---	---	---	---	---	---	1 372/	37 877/	61 ^(d)	
	Radwaste and Control Building	19/	288/N.A.	487/	17 841/	22	10/	920/	0	---	---	---	---	---	---	---	---	516/	19 049/	22	
	Totals	428/	2 000/N.A.	1 365/	42 310/	82	126/	13 636/	1	---	---	---	---	---	---	---	---	1 919/	57 946/	83	
Carbon Steel	Reactor Building	2 323/	8 479/N.A.	3 922/	110 368/	208	505/	61 897/	23	952/	127 160/	55	55/	14 850/	6	---	7 757/	322 754/	292		
	Primary Containment	263/	1 366/N.A.	1 084/	63 181/	72	211/	29 760/	18	1 239/	554 877/	1	374/	145 312/	4	---	3 171/	794 496/	95		
	Turbine Generator Building	3 336/	14 153/N.A.	2 632/	115 525/	194	1 647/	176 600/	69	1 832/	386 321/	96	465/	240 698/	0	559/	234 882/	11	10 471/1 168 179/	370	
	Radwaste and Control Building	3 087/	10 267/N.A.	3 337/	75 778/	258	338/	29 221/	12	12/	4 584/	3	---	---	---	99/	29 410/	0	6 873/	152 139/	273
Totals		9 009/	34 265/N.A.	10 975/	364 852/	732	2 701/	294 478/	122	4 035/1 072 942/	155	894/	400 860/	10	658/	264 292/	11	28 272/2 434 689/1 030			
Stainless Steel	Reactor Building	6 169/	18 674/N.A.	500/	4 551/	10	54/	2 143/	0	---	---	---	---	---	---	---	6 723/	25 368/	10		
	Primary Containment	3 850/	10 603/N.A.	110/	3 411/	8	64/	8 789/	3	55/	21 440/	6	---	---	---	---	4 079/	44 243/	17		
	Turbine Generator Building	---	---	38/	1 474/	0	103/	6 421/	0	---	---	---	---	---	---	---	141/	7 895/	0		
	Radwaste and Control Building	1 150/	4 747/N.A.	1 026/	10 164/	96	55/	1 756/	0	---	---	---	---	---	---	---	2 231/	16 667/	96		
Totals		11 169/	34 024/N.A.	1 674/	19 600/	114	276/	19 109/	3	55/	21 440/	6	---	---	---	---	13 174/	94 173/	123		

(a) Dashed line represents insignificant amount.

(b) The data are presented in the following format: Piping Length, m/Piping Mass, kg/Number of Valves.

(c) N.A. = not available.

(d) The total number of valves for a particular system does not include small-bore valves 51 mm or less in diameter.

TABLE C.3-11. Summary of Contaminated Piping and Valves in the Reference BWR

C.4 HEATING, VENTILATING AND AIR CONDITIONING SYSTEMS

This section presents information about the various heating, ventilating and air conditioning (HVAC) systems in the Reactor Building and Primary Containment, the Turbine Generator Building, and the Radwaste and Control Building. These systems are designed to guard against the spread of airborne radioactive contamination.

The design and process information in this section has not been converted to metric units, since it is included for illustration only and is not essential to the study results.

C.4.1 HVAC - Reactor Building and Primary Containment

The HVAC systems for the Reactor Building and the Primary Containment are separate and distinct, except for an intertie used as needed for purging the Primary Containment. The following sections describe each of these systems.

C.4.1.1 Reactor Building

The Reactor Building heating and ventilating system is shown in Figure C.4-1. The system is basically a "push-pull" heating and ventilation system, providing once-through air flow with no recirculation, consisting of the following subsystems:

- main supply system
- main exhaust system
- sump vent exhaust filter system
- emergency cooling system
- miscellaneous ventilation systems.

Equipment details are given in Table C.4-1. System descriptions follow.

Main Supply System. The Reactor Building main supply system consists of a heating and ventilation unit (EKN 107), air distribution ductwork, two isolation butterfly valves on the fresh air intake, and associated controls. The heating and ventilation unit consists of an insulated sheet metal housing containing a prefilter, steam coil, evaporative cooling section, and two 100%-capacity (97,350-cfm) vaneaxial fans. The prefilter is of the renewable roll

FIGURE C.4-1. Reactor Building HVAC System

TABLE C.4-1. Reactor Building HVAC System Component Design Data

Reactor Building Supply Heating and Ventilating Unit (EKN 107)

Location	e1. 174.4 m
No. of Units	1
Air Flow, SCFM	97,350
Air Washer Pump HP	10 (two pumps - one operating, one standby)
Fan Type	Vaneaxial, V-Belt Drive
Drive Motor HP	200 (two fans - one operating, one standby)
Heating Coil	
Area, ft ²	115 2
Steam Flow, lb/hr	6,800
Filter	Continuous Roll
Seismic Category	II
Housing Size, LxWxH, m	9.1 x 5.2 x 4.9

Reactor Building Exhaust Fans (EKN 122)

Location	e1. 174.4 m
No. of Fans	2 (one operating, one standby)
Fan Type	Vaneaxial
Drive	Direct (200 HP Motor)
Capacity, each, SCFM	95,000 - 105,000
Seismic Category	II
Size, LxWxH, m	1.4 dia x 1.8

Reactor Building Sump Vent Filter Units (EKN 96)

Location	e1. 174.4 m
No. of Units	2
Type	Package
Capacity, each, SCFM	1000
Fan Type	Centrifugal, Direct Drive (5 HP Motor)
Prefilters	
Type	Replaceable Media
Efficiency	85% Min.
Heater	
Type	Electric Resistance, 3-Stage
Capacity	21 kW Total
HEPA Filters	
Type	High-Efficiency
Media	Glass Media, Water Repellent and Fire Resistant
Efficiency	99.97%
Charcoal Adsorber	
Media	Activated Carbon Type BC727
Efficiency @ 70% RH & 298 K	99.9% Elemental Iodine 98.0% Methyl Iodine
Moisture Separator	Impingement Type
Seismic Category	II
Housing Size, LxWxH, m	6.7 x 0.8 x 1.0

TABLE C.4-1. Reactor Building HVAC System Component Design Data (Contd)

Reactor Building Emergency Fan Coil Units (EKN 100)

Location	RHR Pump Rooms, el. 135.3 m	HPCS Pump Room, el. 135.3 m	LPSC Pump Room, el. 135.3 m	RCIC Pump Room, el. 135.3 m	MCC Rooms, el. 159.1 m
No. of Units	3	1	1	1	2
Air Flow per Unit, ACFM	5,208	15,625	9,575	3,125	5,730
Sensible Cooling Capacity per Unit, Btuh	165,000	500,000	280,000	60,000	71,280
Fan Drive Motor HP	3	10	5	2	3
Cooling Water Supply Service	Standby Service Water	Standby Service Water	Standby Service Water	Standby Service Water	Standby Service Water
Seismic Category	I	I	I	I	I
Housing Size, LxWxH, m	1.1 x 1.4 x 1.7	2.4 x 1.8 x 1.8	1.2 x 1.4 x 2.4	1.2 x 1.5 x 1.1	1.7 x 1.1 x 1.4

Location	D.C. MCC Room, el. 143.6 m	H ₂ Recombiner Rms, el. 174.4 m	RWCU Analyzer Rms, el. 167.0 m
No. of Units	1	2	2
Air Flow per Unit, SCFM	6,500	4,100	3,300
Sensible Cooling Capacity per Unit, Btuh	85,000	53,900	43,000
Fan Drive Motor HP	3	3	2
Cooling Water Supply Source	Standby Service Water	Standby Service Water	Standby Service Water
Seismic Category	I	I	I
Housing Size, LxWxH, m	1.7 x 1.4 x 0.8	1.5 x 0.9 x 1.2	1.5 x 0.9 x 1.2

Miscellaneous Area Ventilation Systems

Location	Steam Tunnel, el. 152.7 m	RWCU Sample Hood, Sample Room 1A, el. 167.0 m	Railroad Airlock, el. 134.4 m
No. of Units	2	1	1
Fan Drive	Direct (2 HP Motor)	Direct (3 HP Motor)	V-Belt (5 HP Motor)
Nominal Air Flow per Unit, SCFM	3,000	850	5,000
Sensible and Latent Cooling Capacity at Nominal Air Flow, Btuh	97.2	---(a)	42,500
Cooling Water Supply Source	Plant Service Water	---	Plant Service Water
Filters	---	Prefilter. Replaceable Viscous Impingement. HEPA Filter: Glass Media, 99.97 Efficient	---
Seismic Category	I	II	II
Housing Size, LxWxH, m	1.2 x 0.9 x 1.5	1.8 x 0.9 x 0.8	2.3 x 1.8 x 1.8

(a) Dashed line means "not applicable."

type, automatically progressed to maintain consistent pressure drop. The steam coil is of the non-freeze type, with automatic face and bypass dampers used for temperature control. The source of the heating steam during decommissioning is the auxiliary boiler (EKN 236). The evaporative cooling section is of the capillary type, with two full-capacity water recirculation pumps. Makeup water to the evaporative cooler is supplied from the plant potable water system.

During normal operation, the main supply system isolation dampers are open, and the system continuously distributes tempered, 100% outdoor air throughout the building. One supply fan is normally operating, with the second fan in standby. The standby fan starts automatically if the operating fan fails.

The Reactor Building main supply system also provides makeup air to the Primary Containment during Primary Containment purge, if needed. During purge, isolation valves direct air from the supply air system into the Primary Containment.

In the event of a Reactor Building isolation signal, the main supply system ventilation unit stops and the two isolation dampers in the outdoor air intake close. The signals that cause Reactor Building isolation are:

- reactor vessel low-water level
- high drywell pressure
- high radiation level in the Reactor Building exhaust air plenum.

Only the last isolation signal has application during decommissioning.

Main Exhaust System. The Reactor Building main exhaust system draws air from all areas with potential radioactive contamination and discharges it to the elevated release point on the north side of the building. There are two 100%-capacity (97,350-cfm) vaneaxial fans (EKN 122) in the system. One fan normally operates, with the second fan in standby.

Approximately half of the Reactor Building exhaust air is drawn from the refueling level of the Reactor Building. Intake ducts to the main exhaust

system are embedded around the periphery of the spent fuel pool, the dryer and separator storage pool, and the reactor well, to remove any radioactive vapors generated in the pools. Volume control dampers in the ducts from each pool control the air flow over each pool.

During Primary Containment purge, the purge air is discharged through the Reactor Building main exhaust system or through the standby gas treatment system. Ducts connect the Primary Containment drywell and suppression pool areas with both systems. The reactor building main exhaust system is normally used for purge, with the standby gas treatment system used only when an unacceptably high level of airborne radioactivity is present inside the Primary Containment.

The aforementioned isolation signals stop and isolate the Reactor Building main exhaust system and start the standby gas treatment system (EKN 92). (See Section C.5.3.)

Sump Vent Exhaust Filter System. The sump vent exhaust filter system (EKN 96) is composed of two full-capacity (1000-cfm) filter units that draw air from the sumps and drain headers and discharge it to the main Reactor Building exhaust system. Each filter unit consists of the following elements in a sheet metal housing:

- moisture separator
- electric blast coil heater
- medium-efficiency particulate prefilter
- high-efficiency particulate (HEPA) filter
- tray-type 2-inch thick charcoal adsorber filter
- centrifugal fan.

Reactor Building Emergency Cooling Systems (RBECS). Each of the 13 rooms housing critical equipment is provided with an individual air handling unit (EKN 100) fully enclosed within the room. Each air handling unit is composed of a direct-driven centrifugal fan and a cooling water coil in a sheet metal housing. Water is supplied to the water coils by the standby service water system. During normal operation all 13 air handling units are in standby.

The units that service the emergency pump rooms start automatically upon actuation of their associated pumps. The units servicing the motor control center (MCC) equipment rooms and the analyzer rooms start automatically upon any one of the aforementioned Reactor Building isolation signals.

All units recirculate the air within the room they serve, removing the heat generated in the room via the water coil.

Miscellaneous Area Ventilation Systems. The following HVAC systems are provided to service local areas in the Reactor Building:

1. The main steam tunnel is serviced by two 100%-capacity, recirculating fan coil units located within the tunnel. Both units are composed of a cooling water coil, supplied by the plant service water system, and a direct-driven centrifugal fan in a sheet metal housing.
2. The reactor water cleanup (RWCU) sample hood is provided with supply air from the Reactor Building supply air system. An electric duct heater is provided in the supply air duct. The exhaust filter unit, which draws air from the hood, is composed of a viscous impingement prefilter, a HEPA filter, and a centrifugal fan in a sheet metal housing. The exhaust fan discharges air into the Reactor Building exhaust system.
3. The railroad airlock, on grade elevation, is heated and cooled by an air handling unit that operates in a recirculation mode. The air handling unit is composed of a prefilter, cooling water coil, and centrifugal fan, all in a sheet metal housing. A 15-kW electric blast coil heater is provided in the discharge duct for heating. Plant service water is supplied to the water coil for cooling.

C.4.1.2 Primary Containment

The Primary Containment cooling system is depicted in Figure C.4-2. Equipment details are listed in Table C.4-2. Cooling of the drywell is provided by five fan coil units (EKN 107) that recirculate containment air through cooling water coils for heat removal. Nine recirculation fans

NOTES

- 1 ALL AIR FLOWS IN CFM
- 2 MOTOR POWER IN HP

KEY

	FILTER
	COOLING COIL
	SUPPLY DUCTWORK
	EXHAUST DUCTWORK
	FAN MOTOR AND MOTOR POWER (TYPICAL)

FIGURE C.4-2. Primary Containment Cooling System

TABLE C.4-2. Primary Containment Cooling System
Component Design Data

Primary Containment Fan Coil Units (EKN 107)

No. of Units	3 (one standby)	2 (one standby)
Nominal Air Flow, each, SCFM	32,000	35,000
Cooling Capacity (Sensible) at Nominal Air Flow, Btuh	1,140,000	1,140,000
Fan Type	Vaneaxial (two per unit)	Vaneaxial (two per unit)
Drive Motor HP	40 and 50, each unit	30 and 75, each unit
Cooling Water Supply Source	Reactor Building Closed Cooling System	Reactor Building Closed Cooling System
Seismic Category	I	I
Housing Size, LxWxH, m	3.4 x 1.8 x 2.1	3.2 x 1.8 x 2.1

Primary Containment Head Area Return Fans (EKN 106)

No. of Fans	2
Fan Type	Vaneaxial
Drive	Direct (7.5 HP Motor)
Capacity, each, ACFM	6000
Seismic Category	I
Size, m	0.6 dia x 0.6

Primary Containment Recirculation Fans (EKN 104,105)

No. of Fans	7
Fan Type	Vaneaxial
Drive	Direct (10 HP Motor)
Capacity, each, ACFM	20,000
Seismic Category	I
Size, m	1.1 dia x 1.1

(EKN 104, 105, 106) are also provided at various locations in the drywell. Each of the five fan coil units consists of two vaneaxial fans (both of which operate at the same time), a cooling water coil supplied from the Reactor Building closed cooling water system, and a prefilter, all in a sheet metal housing.

Three of the drywell cooling units are located low in the drywell, and two are located at a higher level. During normal operation, two of the lower level units and one of the upper level units operate continuously to remove the heat generated in the drywell. The third lower level unit and the second upper level unit are normally in standby.

Each of the three lower level fan coil units are provided with one 18,250-cfm capacity vaneaxial fan that discharges directly into the general drywell volume, and one two-speed, 13,750/18,000-cfm vaneaxial fan that discharges, via ductwork, into the area beneath the reactor vessel and between the reactor vessel and the sacrificial shield wall. Volume dampers in the branch ducts proportion flow to the two areas served.

The two upper level fan coil units are each provided with one 26,000-cfm fan and one 9000-cfm fan. The larger fans discharge directly into the upper volume area, and the smaller fans discharge, via a shared duct header, into the containment head area above the refueling bellows. Two 6000-cfm vaneaxial recirculation fans draw air from the containment head area and discharge it below the refueling bellows. Return air to the upper fan coil units is drawn from immediately below the refueling bellows.

The remaining seven recirculation fans produce an air flow of 20,000 cfm each, and are located at various locations on lower levels in the drywell to increase air circulation in these locations. During normal operation, five of the fans operate.

C.4.2 HVAC - Turbine Generator Building

The heating and ventilating system of the Turbine Generator Building is primarily a "push-pull" heating and ventilating system consisting of the following subsystems:

- main supply system
- main exhaust system
- auxiliary boiler room ventilation system
- transformer vault ventilation system
- sample room air conditioning system.

Equipment details are given in Table C.4-3. The system is depicted in Figure C.4-3. System descriptions follow.

Main Supply System

The Turbine Generator Building supply air system is composed of four supply ventilation units (EKN 276) and associated distribution ductwork. The units are operated in pairs, with one pair discharging into a common supply duct system servicing the west side of the building and the other discharging into a common supply duct system servicing the east side of the building.

Each of the four ventilation units is composed of a prefilter, a steam coil with face and bypass dampers, an evaporative cooling section, and a centrifugal fan, all enclosed in an insulated sheet metal housing. The pre-filters are of the renewable roll type, automatically progressed to maintain consistent pressure drop. The steam coils are of the non-freeze type, with automatic face and bypass dampers used for temperature control. The source of the heating steam during decommissioning is the auxiliary boiler (EKN 236). The evaporative cooling sections are of the capillary air washer type, with two full-capacity water recirculation pumps. Makeup water to the air washers is supplied from the plant potable water system.

The centrifugal fan of each ventilation unit is furnished with automatic inlet vanes that vary the air flow supply to maintain the Turbine Building at a set pressure with respect to outdoors. The nominal rates of air flow through the ventilation units are 74,750 cfm through each of the two west units and 58,450 cfm through each of the two east units. The fan intake ducts of each pair of ventilation units are joined by means of a manual damper, so that a single fan can draw air through both ventilation units in the event of failure of one fan. In this mode of operation, with fan inlet vanes fully open, the fan capacity can be increased approximately 50%, thus minimizing the effect of a fan failure on system cooling capacity.

**TABLE C.4-3. Turbine Generator Building HVAC
System Component Design Data**

Turbine Generator Building Supply Heating and Ventilating Units (EKN 276)

Location	East el. 152.7 m	West el. 152.7 m
No. of Units	2	2
Air Flow, each, SCFM	58,450	74,750
Air Washer Pump HP	5	7.5
Fan Type	Centrifugal, V-Belt Drive	Centrifugal, V-Belt Drive
Drive Motor HP	125	125
No. of Fans	2 (one for each unit)	2 (one for each unit)
Heating Coil Area, ft ²	88.6	112.6
Steam Flow, lb/hr	4,530	5,570
Housing Size, LxWxH, m	11.3 x 4.0 x 4.0 and 14.0 x 4.6 x 4.0	10.7 x 3.5 x 4.9

Turbine Generator Building Exhaust Fans (EKN 494)

Location	Radwaste and Control Building Roof, el. 165.2 m
No. of Fans	4 (one standby)
Fan Type	Centrifugal
Drive	Direct (100 HP Motor)
Capacity, each, ACFM	97,059
Size, m	3.6 dia x 3.1

Auxiliary Boiler Room Air Handling Unit

Location	Auxiliary Boiler Room, el. 134.4 m
No. of Units	1
Drive	V-Belt (15 HP Motor)
Nominal Air Flow, SCFM	15,000
Steam Flow, lb/hr	980
Seismic Category	II
Size, LxWxH, m	3.1 x 3.1 x 1.4

Miscellaneous Exhaust Fans

Location	Fume Hood, Sample Room, el. 134.4 m	Sample Room, el. 134.4 m	Transformer Vaults, el. 134.4 m
No. of Units	1	1	2
Drive	Direct (2 HP Motor)	Direct (3/4 HP Motor)	Direct (2 HP Motor)
Nominal Air Flow, SCFM	1,350	810	9,000 (each)
Seismic Category	II	II	II
Size, m	0.6 dia x 0.6	0.4 dia x 0.5	0.7 dia x 0.8

FIGURE C.4-3. Turbine Generator Building HVAC System

Automatic dampers provided on the intake of each ventilation unit permit the unit to draw 100% outdoor air or 100% recirculation air from the Turbine Building.

Main Exhaust System

The main exhaust system consists of four centrifugal fans (EKN 494) that draw air (~97,000 cfm each) from a central exhaust plenum (EKN 480), all mounted on the roof of the Radwaste and Control Building. Normally, three of the exhaust fans operate continuously, with one fan in standby. Constant air flow through the operating fans is maintained by volume dampers on the fan discharges.

Almost all exhaust air is drawn from the shielded areas of the Turbine Generator Building, where the potential for airborne radioactive contamination is highest, thus inducing flow from the less-contaminated areas through the shielded areas. The air in the exhaust plenum is monitored for radioactive contaminants, which are indicated on a recorder in the main control room.

Motor operated shut-off dampers in the main branches of the exhaust duct system stop exhaust flow on an area-by-area basis. Automatic volume dampers in the exhaust system draw full exhaust flow from the shielded equipment vaults on the lowest level of the Turbine Generator Building when the exhaust system is operating at reduced capacity.

Auxiliary Boiler Room Ventilation System

The auxiliary boiler room, located on the lowest level of the Turbine Generator Building, is ventilated from the Turbine Generator Building supply air system when the boiler is not operating. However, when the boiler is operating and drawing combustion air from the room, additional air (15,000 cfm) is supplied by a separate air handling unit. The air handling unit starts automatically when the boiler is started and draws 100% outdoor air through a weather louver, heats it as required, and discharges it into the room. Part of this air is drawn by the boiler as combustion air, with the balance of the air leaving via relief dampers in the exterior wall of the boiler room.

Transformer Vault Ventilation System

Two transformers are located in adjacent equipment vaults in the lower level of the Turbine Generator Building, and two 9000-cfm tubeaxial fans provide ventilation. The two fans, one operating and the other in standby, exhaust air from the vaults, with makeup air being provided through ventilation openings in the walls. Both fans are powered from the critical diesel generator buses.

Sample Room Air Conditioning

The Turbine Generator Building sample room, located on the lowest level of the building, is provided with a sample hood exhaust filter system and a self-contained air conditioning system. The sample room hood air is supplied by a centrifugal fan that draws air from the corridor outside the sample room and from the room itself. Air is exhausted from the hood by a filter unit composed of a medium-efficiency prefilter, a HEPA filter, and a 1350-cfm exhaust fan, all contained in a sheet metal housing. The filter unit discharges into the main Turbine Building exhaust system.

The sample room air conditioning unit is a 5-Ton, water-cooled unit, with condensing water supplied from the plant service water system. The unit delivers 2400 cfm in a partial recirculation mode, recirculating 2000 cfm of room air and drawing 400 cfm of makeup air from an adjacent corridor.

C.4.3 HVAC - Radwaste and Control Building

The heating, ventilating and air conditioning systems in the Radwaste and Control Building consist of three separate and distinct systems for the following areas:

- control tower
- condenser off gas charcoal adsorber vault
- radwaste area.

Descriptions of these systems follow.

Control Tower

The control tower has two redundant, continuously recirculating HVAC systems with minimal fresh air intake and minimal exhaust. It is assumed that the control tower and its HVAC systems are not radioactively contaminated; therefore, no further description is necessary.

Condenser Off Gas Charcoal Adsorber Vault

The condenser off gas charcoal adsorber vault has a redundant closed-loop refrigeration system and dual air recirculation systems. This system does not operate during decommissioning, and, since it is contaminated, the air circulation equipment is discussed in Section C.3.3.

Radwaste Area

The heating and ventilating system of the radwaste area in the Radwaste and Control Building is shown in Figure C.4-4. The main system is a "push-pull" heating and ventilating system providing once-through air flow with no recirculation. In addition, individual air conditioning systems are provided for all rooms that personnel normally occupy for extended periods of time during plant operation. Equipment details are given in Table C.4-4. System descriptions follow.

Main Supply System. The main radwaste area supply air system consists of a supply ventilation unit (EKN 455) and distribution ductwork. The supply ventilation unit consists of an insulated sheet metal housing containing a prefilter, steam coil, evaporative cooling section, and two 100%-capacity (82,430-cfm) centrifugal fans. The renewable roll-type prefilter is automatically progressed to maintain consistent pressure drop. The non-freeze-type steam coil uses automatic face and bypass dampers for temperature control. The source of the heating steam during decommissioning is the auxiliary boiler (EKN 236). The evaporative cooling section is of the capillary type, with two full-capacity water recirculation pumps. Makeup water to the evaporative cooler is supplied from the plant potable water system.

FIGURE C.4-4. Radwaste Area HVAC System

TABLE C.4-4. Radwaste Area HVAC System Component Design Data

Radwaste Area Supply Heating and Ventilating Unit (EKN 455)

Location	e1. 148.4 m
No. of Units	1
Air Flow, SCFM	82,430 (91,304 max.)
Airwasher Pump HP	7.5 (two pumps - one operating, one standby)
Fan Type	Centrifugal, V-Belt Drive
Drive Motor HP	150 (two fans - one operating, one standby)
Heating Coil Area, ft ²	115.2
Steam Flow, lb/hr	6,980
Filter	Continuous Roll
Seismic Category	II
Housing Size, LxWxH, m	10.1 x 5.5 x 3.8

Radwaste Area Exhaust Filter Units (EKN 453)

Location	e1. 148.4 m
No. of Units	3 (one standby)
Type	Package
Capacity, each, SCFM	42,000
Fan	Centrifugal, Direct Drive (75 HP Motor)
Prefilters	
Type	Replaceable Media
Efficiency	Min. 85
HEPA Filters	
Type	High-Efficiency
Media	Glass Media, Water Repellent and Fire Resistant
Efficiency	99.97
Housing Size, LxWxH, m	4.9 x 5.8 x 4.0

Air Handling Units (EKN 378)

Location	Health Physics, e1. 148.4 m	Radiochem. Lab, e1. 154.5 m	Hot Instr. Shop, e1. 148.4 m	Radwaste Control Room, e1. 148.4 m	Counting Room, e1. 154.5 m
No. of Units	1	1	1	1	1
Drive	V-Belt (1 HP Motor)	V-Belt (5 HP Motor)	V-Belt (2 HP Motor)	V-Belt (5 HP Motor)	V-Belt (1.5 HP Motor)
Nominal Air Flow, SCFM	1,475	6,900	3,000	7,000	2,000
Cooling Capacity (Sensible and Latent) at Nominal Air Flow, MBtuh	57	255	115	204	76
Cooling Water Supply Source	Chilled Water	Chilled Water	Chilled Water	Chilled Water	Chilled Water
Seismic Category	II	II	II	II	II
Housing Size, LxWxH, m	1.7 x 0.9 x 0.8	2.4 x 2.3 x 2.9	1.4 x 1.7 x 1.7	2.0 x 1.7 x 2.4	2.4 x 2 x 0.9

Fume Hood Exhaust Filter Units (EKN 460)

Location	Radiochem. Lab, e1. 148.4 m	Hot Instr. Shop, e1. 148.4 m	Sample Room, e1. 148.4 m	Sample Room, e1. 154.5 m
No. of Units	3	1	1	1
Drive	Direct (3 HP Motor)	Direct (3 HP Motor)	V-Belt (3 HP Motor)	V-Belt (7.5 HP Motor)
Nominal Air Flow, SCFM	1,850 (each)	1,850	1,450	3,700
Prefilter (Identical for each)	Replaceable Viscous Impingement Type			
HEPA Filter (Identical for each)	Glass Media, Water Repellent and Fire Resistant, 99.97 Efficient			
Seismic Category	II	II	II	II
Housing Size, LxWxH, m	2.4 x 1.5 x 1.1	2.0 x 1.1 x 1.1	2.6 x 1.4 x 0.9	2.7 x 1.4 x 1.4

During normal plant operation, the supply unit operates continuously, providing tempered air throughout the building via the supply duct distribution system. In the event of loss of offsite power, the radwaste area supply system is inoperative.

Main Exhaust System. The radwaste area exhaust system is composed of three 50%-capacity (41,015-cfm) exhaust filter units (EKN 453). Each unit consists of a sheet metal housing containing a medium-efficiency cartridge-type prefilter, HEPA filter, and centrifugal fan.

During normal operation, with one unit in standby, two exhaust units draw air from a common exhaust duct system and discharge it above the Radwaste and Control Building roof. All exhaust is drawn from areas with the highest radioactive contamination potential, thus inducing air flow from clean areas into the potentially contaminated zones.

Individual Air Conditioning Systems. The radwaste control room, hot instrument shop, radiochemistry laboratory, sample room, counting room, and health physics office have individual air conditioning systems to maintain personnel comfort. All areas except the radwaste control room are serviced by once-through air conditioning systems. The radwaste control room, with no contamination potential during plant operation, has a partial recirculation air conditioning system.

The air conditioning for each area is provided by air handling units (EKN 378) that draw air from the radwaste area main supply system; pass it through filters, chilled water coils, and electric heaters; and discharge it into the rooms being conditioned. Chilled water is supplied to the coils from the Radwaste Building chilled water system. The conditioned air is exhausted from the rooms through vents to the balance of the building or through hoods that discharge into the main exhaust system.

The exhaust hoods in the radiochemistry laboratory, hot instrument shop, and sample room are provided with local filter units (EKN 460), composed of prefilters, HEPA filters, and fans, to remove radioactive particles from the exhaust air prior to discharge into the exhaust system.

C.5 RADWASTE SYSTEMS

This section describes the systems in the reference BWR that handle the liquid, solid, and gaseous radwastes. Figure C.5-1 shows the process flow of both the liquid and solid radwaste systems.

C.5.1 Liquid Radwaste System

The liquid radwaste system is composed of a group of subsystems designed to collect, treat, and recycle different categories of waste water. These subsystems are designated as the equipment drain (radioactive) subsystem, the floor drain (radioactive) subsystem, and the miscellaneous waste (radioactive) subsystem. A simplified block flow diagram is presented in Figure C.5-2. Equipment design data are listed in Table C.5-1, and equipment performance data are given in Table C.5-2.

All liquid radwaste process streams terminate in a sample or distillate tank. Since the liquid radwaste system is operated on a batch basis, each treated batch can be sampled to ensure that the treatment is effective. If the sample indicates that the process liquid is still above acceptable radioactivity limits or is substandard in purity, the batch is recycled, either through the same treatment or through a subsystem providing a higher degree of treatment. If the sample indicates that the level of radioactivity is within limits for discharge, the processed liquid may be discharged to the river, or, if desired, to the condensate storage tanks (EKN 228).

C.5.1.1 Equipment Drain (Radioactive) Subsystem

Liquid wastes collected in the waste collector tank (EKN 301) or the waste surge tank (EKN 312) come from the following sources:

- drywell equipment drain sump
- Reactor Building equipment drain sump
- Radwaste Building equipment drain sump
- Turbine Generator Building equipment drain sumps (2 each)
- reactor water cleanup demineralizer system
- residual heat removal system
- cleanup phase separators (decant water)
- condensate phase separators (decant water)
- fuel pool seal rupture drains.

FIGURE C.5-2. Liquid Radwaste System Block Flow Diagram

TABLE C.5-1. Liquid Radwaste Equipment Design Data

<u>EKN(a)</u>	<u>Subsystem Equipment Piece (Quantity)</u>	<u>Material of Construction(b)</u>	<u>Capacity, each(c)</u>
<u>Equipment Drain (Radioactive)</u>			
301.	Waste Collector Tank (1)	CS	75.7 m ³
312.	Waste Surge Tank (1)	CS	284 m ³
313.	Waste Sample Tank (2)	CS	75.7 m ³
319.	Waste Collector Pump (1)	CS	0.72 m ³ /min @ 47 m TDH
323.	Waste Surge Pump (1)	CS	0.72 m ³ /min @ 47 m TDH
324.	Waste Sample Pump (2)	CS	0.72 m ³ /min @ 31 m TDH
431.	Waste Filter Hold Pump (1)	CS	0.28 m ³ /min @ 15 m TDH
440.	Waste Collector Filter (1)	CS	17.5-m ² Filter Area
443.	Waste Demineralizer (1)	CS	1.8-m ³ Resin Bed; 0.7 m ³ /min
<u>Floor Drain (Radioactive)</u>			
300.	Floor Drain Collector Tank (1)	CS	75.7 m ³
314.	Floor Drain Sample Tank (1)	CS	75.7 m ³
318.	Floor Drain Collector Pump (1)	CS	0.72 m ³ /min @ 47 m TDH
326.	Floor Drain Sample Pump (1)	CS	0.72 m ³ /min @ 31 m TDH
430.	Floor Drain Filter Hold Pump (1)	CS	0.28 m ³ /min @ 15 m TDH
439.	Floor Drain Filter (1)	CS	17.5-m ² Filter Area
444.	Floor Drain Demineralizer (1)	CS	1.8-m ³ Resin Bed; 0.7 m ³ /min
<u>Miscellaneous Waste (Radioactive)</u>			
308.	Chemical Waste Tank (2)	SS	56.8 m ³
309.	Distillate Tank (2)	SS	56.8 m ³
310.	Detergent Drain Tank (2)	CS	6.1 m ³
331.	Concentrator Feed Pump (2)	SS	0.11 m ³ /min @ 29 m TDH
332.	Chemical Waste Pump (2)	SS	1.1 m ³ /min @ 61 m TDH
333.	Distillate Tank Pump (2)	SS	0.54 m ³ /min @ 70 m TDH
334.	Detergent Drain Pump (2)	SS	0.1 m ³ /min @ 27 m TDH
376.	Detergent Drain Filter (1)	CS	0.1 m ³ /min
386.	Decontamination Solution Concentrator (2)	SS	0.04 m ³ /min
387.	Decon. Solution Conc. Heating Element (2)	SS and CS	Steam Flow: 2 860 kg/hr Solution Flow: 8.7 m ³ /min
435.	Decon. Solution Conc. Bottoms Recycle Pump (2)	SS	8.7 m ³ /min @ 12 m TDH
445.	Distillate Polishing Demineralizer (1)	CS	1.8-m ³ Resin Bed; 0.7 m ³ /min
457.	Decon. Solution Conc. Condenser (2)	SS and CS	Coolant Flow: 2.8 m ³ /min
<u>Auxiliary Equipment</u>			
416.	Waste Precoat Pump (1)	CS	1.2 m ³ /min @ 15 m TDH
417.	Waste Precoat Tank (1)	CS	0.8 m ³
418.	Chemical Addition Pump (2)	Carpenter 20 Steel	0.3 m ³ /hr
419.	Chemical Addition Tank (2)	CS	0.8 m ³
420.	Waste Filter Aid Tank (1)	CS	2.4 m ³
421.	Waste Filter Aid Pump (1)	CS	0 to 0.52 m ³ /hr @ 1.0 MPa
428.	Floor Drain Filter Aid Pump (1)	CS	0 to 0.52 m ³ /hr @ 1.0 MPa
425.	Resin Addition Tank (1)	SS	4.5 m ³

(a)EKN = Equipment Key Number (see Section C.2).

(b)CS stands for carbon steel, and SS for stainless steel.

(c)The design pressure of the equipment is 1.14 MPa, the design temperature is 339 K.

TABLE C.5-2. Liquid Radwaste Equipment
Performance Factors

<u>Equipment</u>	<u>Radioactivity Decontamination Factor</u>
Demineralizers	Soluble 100 Insoluble 50
Filters	Soluble 1 Insoluble 2
Concentrators (Influent/Distillate) ^(a)	1 000

(a)Concentration Factor (Influent/Bottoms, wt% salts) = 0.6/32.

Treatment consists of filtration, demineralization, and transfer to a waste sample tank (EKN 313). After sampling and analysis, the processed liquids may be routed to the blowdown line for discharge to the river or may be reprocessed, as required.

C.5.1.2 Floor Drain (Radioactive) Subsystem

Liquid wastes collect in the floor drain collector tank (EKN 300) from the following sources:

- drywell floor drain sump
- Reactor Building floor drain sumps (4 each)
- Radwaste Building floor drain sumps (3 each)
- Turbine Generator Building floor drain sumps (2 each)
- waste sludge phase separator.

Processing consists of filtration, demineralization, and transfer to the floor drain sample tank (EKN 314) for sampling and analysis. The process liquid can be recycled for additional treatment, as necessary, or, if it meets acceptable limits, can be discharged via the blowdown line to the river.

C.5.1.3 Miscellaneous Waste (Radioactive) Subsystem

Liquid wastes collected in the chemical waste tanks (EKN 308) are from the following sources:

- detergent drain tanks (EKN 310)
- Reactor Building and Turbine Generator Building decontamination drains
- low-purity wastes from either the equipment or floor drain subsystems
- filter demineralizer element chemical cleaning solutions
- battery room drains
- chemical system overflows and tank drains
- laboratory drains.

The high conductivity of these wastes precludes treatment by ion exchange. Because of higher chemical content, they are processed through a decontamination solution concentrator (EKN 386).

The concentrator bottoms are processed through the solid radwaste system, and the distillate is routed to a distillate tank (EKN 309). After analysis, the distillate is either routed through the distillate polishing demineralizer (EKN 445) or recycled through the concentrator to further reduce impurities, or discharged to the river via the blowdown line.

Detergent wastes are collected in the detergent drain tanks (EKN 310). These wastes are primarily laboratory and decontamination solutions that contain detergents. They are filtered through the detergent drain filter (EKN 376) prior to routing to the chemical waste tanks (EKN 308) for further processing.

C.5.2 Solid Radwaste System

The solid radwaste system collects, monitors, processes, packages, and provides temporary storage facilities for the radioactive solid wastes prior to offsite disposal. Figure C.5-3 is a block process flow diagram.

FIGURE C.5-3. Solid Radwaste System Process Diagram

The solid radwaste system processes both wet and dry solid radioactive wastes. Wet waste includes backwash sludges, spent resins, and concentrator bottoms from the demineralizers, the filters, and the concentrators in the liquid radwaste systems. Dry waste includes rags, paper, small equipment parts, solid laboratory wastes, air filters, non-recyclable anti-contamination clothing, etc.

Backwash sludges are collected in phase separators, where they are allowed to settle. The resulting decantates are sent to the liquid radwaste system, and the sludges are transferred, together with the spent resins, to the process waste (radioactive) system. There the residual water is centrifuged off, a solidifying agent (cement and sodium silicate) and the concentrator bottoms are added, and the mixture is packaged in 1.42-m^3 shipping containers for ultimate offsite disposal.^(a) Table C.5-3 lists design data for equipment that handles wet solid radioactive wastes.

Dry solid radioactive wastes are segregated into compressible and non-compressible wastes. Compressible wastes are compacted into 0.21-m^3 steel drums. Noncompressible wastes are packaged in similar drums. This waste is shipped offsite for disposal.

C.5.3 Gaseous Radwaste System

Besides the HVAC systems described in Section C.4, the reference BWR systems that process radioactively contaminated gases are the condenser off gas treatment system and the Reactor Building standby gas treatment system.

C.5.3.1 Condenser Off Gas Treatment System

Since no off gas emanates from the main condenser (EKN 200) during plant shutdown, the condenser off gas treatment system is not needed during decommissioning. Therefore, no system description is presented here. However, since the system is radioactively contaminated, the individual equipment pieces are listed in Section C.3.3.

(a) Although the reference BWR is designed to handle the 1.42-m^3 containers, this study assumes the use of 2.84-m^3 containers for consistency with the pressurized water reactor decommissioning study.⁽³⁾

TABLE C.5-3. Wet Solid Radwaste Equipment Design Data

EKN(a)	Subsystem Equipment Piece (Quantity)	Material of Construction(b)	Capacity, each(c)
<u>Condensate Filter Demineralizer</u>			
302.	Condensate Phase Separator Tank (2)	CS and SS	89.0 m ³
304.	Condensate Backwash Receiver Tank (1)	SS	71.9 m ³
320.	Condensate Sludge Discharge Mixing Pump (1)	SS	1.6 m ³ /min @ 49 m TDH
321.	Condensate Decant Pump (1)	SS	1.7 m ³ /min @ 15 m TDH
322.	Condensate Backwash Transfer Pump (1)	SS	1.7 m ³ /min @ 15 m TDH
<u>Equipment Drain (Radioactive)</u>			
305.	Spent Resin Tank (1)	SS	4.5 m ³
327.	Spent Resin Pump (1)	SS	0.08 m ³ /min @ 32 m TDH
<u>Floor Drain (Radioactive)</u>			
306.	Waste Sludge Phase Separator Tank (1)	CS and SS	49.2 m ³
328.	Waste Decant Pump (1)	SS	0.2 m ³ /min @ 15 m TDH
330.	Waste Sludge Discharge Mixing Pump (1)	SS	0.8 m ³ /min @ 32 m TDH
<u>Miscellaneous Waste (Radioactive)</u>			
311.	Decon. Solution Conc. Waste Tank (2)	SS	2.7 m ³
335.	Decon. Solution Conc. Waste Pump (1)	SS	0.11 m ³ /min @ 21 m TDH
408.	Concentrator Waste Measuring Tank (1)	SS	1.5 m ³
<u>Process Waste (Radioactive)</u>			
379.	Hopper Mixer (2)	SS	1.7 m ³
380.	Waste Processing Pump (1)	SS	1.1 to 3.4 m ³ /hr @ 0.34 MPa
409.	Centrifuge (2)	SS	1.0 m ³ /min continuous input rate, 98% solids removal efficiency
<u>Reactor Water Cleanup</u>			
307.	Cleanup Phase Separator Tank (2)	SS	17.0 m ³
329.	Cleanup Decant Pump (1)	SS	0.2 m ³ /min @ 15 m TDH
337.	Cleanup Sludge Discharge Mixing Pump (1)	SS	0.8 m ³ /min @ 52 m TDH
<u>Auxiliary Equipment</u>			
315.	Polymer Day Tank (1)	CS	1.0 m ³
316.	Catalyst Mixing Tank (2)	Polypropylene	0.4 m ³
317.	Polymer Storage Tank (6)	Aluminum	2.1 m ³
336.	Polymer Transfer Pump (1)	CS	1.7 m ³ /hr @ 0.34 MPa
338.	Polymer Processing Pump (1)	SS	0.11 to 1.1 m ³ /hr @ 0.27 MPa
---(d)	Catalyst Processing Pump (2)	SS	0.01 to 0.23 m ³ /hr @ 0.34 MPa

(a)EKN = Equipment Key Number (see Section C.2).

(b)CS = Carbon Steel; SS = Stainless Steel.

(c)Equipment design pressure is 1.14 MPa, design temperature is 339 K.

(d)No equipment key number is available (see Section C.2).

C.5.3.2 Reactor Building Standby Gas Treatment System

The Reactor Building standby gas treatment system (EKN 92), shown schematically in Figure C.5-4, consists of two fully redundant filter trains mounted in separate welded steel housings located on elevation 174.4 m of the Reactor Building (refer to Figure C.2-3). Each filter train, exclusive of the fans, is nominally 14.1 m by 1.7 m by 2.0 m high and weighs approximately 3630 kg. Each fan weighs about 550 kg. Each filter train is composed of the following components in series:

- moisture separator
- two banks of electric blast coil heaters, one primary and one backup; each powered from separate emergency diesel-generator buses; and each composed of three 6.9-kW stages
- one bank of four fiberglass prefilters
- one bank of high-efficiency particulate air (HEPA) filters
- one 102-mm-deep bank of charcoal adsorber filters (515 kg of potassium iodide-impregnated, coconut-base charcoal)
- a second identical bank of charcoal filters
- a second bank of HEPA filters
- two full-capacity ($372\text{-m}^3/\text{min}$) centrifugal fans, one primary and one backup, each powered from separate emergency diesel-generator buses.

Ductwork and valving on the intake of each filter train are arranged so each unit can draw air either from the Reactor Building in the immediate vicinity of the unit or from the Primary Containment via duct connections to the Primary Containment purge exhaust lines. Ductwork and butterfly valves on the discharge side of each filter train are arranged so that either fan can draw air through the filter train, either discharging it out of the Reactor Building via the Reactor Building vent or recirculating it back into the Reactor Building. The Reactor Building standby gas treatment system operates during a Reactor Building isolation or during a Primary Containment purge (see Section C.4.1.1).

FIGURE C.5-4. Reactor Building Standby Gas Treatment System

C.6 ACRONYMS, ABBREVIATIONS, AND EQUIPMENT KEY NUMBERS

Table C.6-1 lists the acronyms and abbreviations that are used in Appendix C, and Table C.6-2 gives all the equipment key numbers and the corresponding equipment.

TABLE C.6-1. Acronyms and Abbreviations in Appendix C

BLDG	Building
BWR	Boiling Water Reactor
CAC	Containment Atmosphere Control
CL	Clearance
CONC or CONCEN	Concentrator
CNDS or COND	Condensate
CRD	Control Rod Drive
CS	Carbon Steel
DECON	Decontamination
DEMIN	Demineralizer
ECCS	Emergency Core Cooling System
EKN	Equipment Key Number
ELEC	Electrical
F/D	Filter Demineralizer
FPC	Fuel Pooling Cooling and Cleanup
FSAR	Final Safety Analysis Report
HEPA	High-Efficiency Particulate Air/ Absolute (referring to filters)
HPCS	High-Pressure Core Spray
HX or HT XCHGR	Heat Exchanger
HVAC	Heating, Ventilating and Air- Conditioning
LPCS	Low-Pressure Core Spray
MCC	Motor Control Center
MG	Motor-Generator
MSIV	Main Steam Isolation Valve
MSLC	MSIV Leakage Control
RBCC	Reactor Building Closed Cooling
RBECS	Reactor Building Emergency Cooling System
RCIC	Reactor Core Isolation Cooling
RH	Relative Humidity
RHR	Residual Heat Removal
RM	Room
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
SGTS	Standby Gas Treatment System
SLC	Standby Liquid Control
SPLY	Supply
SS	Stainless Steel
SYS	System
TDH	Total Dynamic Head
T-G	Turbine Generator
TIP	Traversing In-Core Probe
WPPSS	Washington Public Power Supply System
WNP-2	WPPSS Nuclear Project No. 2

TABLE C.6-2. Equipment Key Numbers in Appendix C

<u>Reactor Building</u>			
1. Reactor Vessel	85. Spent Fuel Cask Washdown Area	276. Turbine Generator Building	358. Solid Radwaste Storage Area Crane
2. Reactor Pedestal	88. Slot Plug Laydown	Supply Heating and Ventilating Unit	359. Truck Loading Crane
3. Vent Pipe	89. Stud Tensioner Spreader		360. Solid Radwaste Container Storage
4. LPCS Pump	90. Spent Fuel Storage Pool		364. Solid Radwaste Container Washdown, Smear, and Testing Station
5. HPCS Pump	91. RBCC Water Surge Tank	<u>Radwaste and Control Building</u>	366. Solid Radwaste Container Capping Station
6. LPCS Water Leg Pump	92. Standby Gas Filter Unit	300. Floor Drain Collector Tank	367. Solid Radwaste Container Filling Station
7. HPCS Water Leg Pump	93. Exhaust Air Damper	301. Waste Collector Tank	371. Solid Waste Baler
8. RHR Pump	94. Hydrogen Recombiner	302. Condensate Phase Separator Tank	373. Solid Radwaste Container Transfer Dolly
9. RHR Water Leg Pump	95. Exhaust Air Plenum	304. Condensate Backwash Receiving Tank	376. Detergent Drain Filter
10. RCIC Pump	96. Sump Vent Filter Unit	305. Spent Resin Tank	377. Expansion Tank
11. RCIC Water Leg Pump	97. Floor Drain Sump Pump	306. Waste Sludge Phase Separator Tank	378. Air Handling Unit
12. RCIC Vacuum Tank	98. Equipment Drain Sump Pump	307. Cleanup Phase Separator Tank	379. Hopper Mixer
13. RCIC Condenser Pump	100. Fan Coil Unit	308. Chemical Waste Tank	383. Off Gas Refrigeration Equipment
14. RCIC Vacuum Pump	104. Lower Level Primary Containment Recirculation Fan	309. Distillate Tank	384. Prefilter
15. RCIC Barometric Condenser	105. Upper Level Primary Containment Recirculation Fan	310. Detergent Drain Tank	385. Afterfilter
16. Reactor Bldg. Cond. Sply. Pump	106. Primary Containment Head Air Return Fan	311. Decontamination Solution Concentrator Waste Tank	386. Decontamination Solution Concentrator
17. Radwaste Bldg. Cond. Sply. Pump	107. Reactor Building Supply Heating and Ventilating Unit	312. Waste Surge Tank	387. Decontamination Solution Concentrator Heating Element
18. Condensate F/D Backwash Pump	107. Primary Containment Fan Coil Unit	313. Waste Sample Tank	388. Decontamination Solution Concentrator Condensate Return Tank
19. Equipment Drain Heat Exchanger	109. Reactor Building Vent	314. Floor Drain Sample Tank	401. Transformer
20. CRD Pump Suction Filter	111. Bridge Crane	315. Polymer Day Tank	402. Battery Rack
21. CRD Pump	112. Dryer and Separator Storage Pool	316. Catalyst Mixing Tank	406. Switchgear
23. Suppression Pool Cleanup Pump	122. Reactor Building Exhaust Fan	317. Polymer Storage Tank	407. Equipment Removal Plug
27. Control Instrument Air Tank	123. MSLC Exhaust Fan	318. Floor Drain Collector Pump	408. Concentrator Waste Measuring Tank
33. Reactor Water Recirculation Pump	124. Air Dryer	319. Waste Collector Pump	409. Centrifuge
34. Drywell Personnel Lock		320. Condensate Sludge Discharge Mixing Pump	412. Chemical Solution Tank
35. Drywell Equipment Hatch	<u>Turbine Generator Building</u>	321. Condensate Decant Pump	413. Condensate Precoat Tank
39. Air Compressor	200. Main Condenser	322. Condensate Backwash Transfer Pump	416. Waste Precoat Pump
40. Main Steam Tunnel	201. Turbine Generator Unit	323. Waste Surge Pump	417. Waste Precoat Tank
42. TIP Drive Mechanism	202. Main Steam and Feedwater Pipe Chase	324. Waste Sample Pump	418. Chemical Addition Pump
46. CRD Module	212. Condensate Pump	326. Floor Drain Sample Pump	419. Chemical Addition Tank
47. RWCU Pump	213. Condensate Booster Pump	327. Spent Resin Pump	420. Waste Filter Aid Tank
49. Sacrificial Shield Wall	214. Reactor Feedwater Pump	328. Waste Decant Pump	421. Waste Filter Aid Pump
50. RBCC Water Pump	215. Mechanical Vacuum Pump	329. Cleanup Decant Pump	422. Floor Drain Filter Aid Pump
51. RBCC Water Heat Exchanger	216. Steam Jet Air Ejector	330. Waste Sludge Discharge Mixing Pump	424. Fuel Pool Precoat Tank
54. RHR Heat Exchanger	217. Gland Seal Steam Condenser	331. Concentrator Feed Pump	425. Resin Addition Tank
55. Fuel Pool Circulation Pump	218. Catalytic Recombiner	332. Chemical Waste Pump	426. Cleanup Precoat Tank
56. Fuel Pool Heat Exchanger	224. Off Gas Condenser	333. Distillate Tank Pump	427. Cleanup Precoat Pump
58. Standby Liquid Control Storage Tank	225. Off Gas Water Separator	334. Detergent Drain Pump	429. Condensate Demineralizer Holding Pump
60. Standby Liquid Control Pump	226. Seal Water Liquid Tank	335. Decontamination Solution Concentrator Waste Pump	430. Floor Drain Filter Hold Pump
61. RWCU Regenerative Heat Exchanger	227. Pumped Drain Tank	336. Polymer Transfer Pump	431. Waste Filter Hold Pump
62. RWCU Non-regenerative Heat Exchanger	228. Condensate Storage Tank	337. Cleanup Sludge Discharge Mixing Pump	432. Fuel Pool Hold Pump
63. Reactor Well	236. Auxiliary Boiler	338. Polymer Processing Pump	433. Cleanup Hold Pump
64. Skimmer Surge Tank	252. High-Pressure Feedwater Heater	340. Floor Drain Sump Pump	435. Decontamination Solution Concentrator Bottoms Recycle Pump
65. Air Lock	253. Low-Pressure Feedwater Heater	341. Equipment Drain Sump Pump	438. Condensate Filter Demineralizer
66. Steam Dryer Storage	254. Steam Evaporator	342. Chemical Drain Sump Pump	439. Floor Drain Filter
68. Spent Fuel Shipping Cask Storage	255. Evaporator Drain Tank	348. Cooler Condenser	440. Waste Collector Filter
74. Upper and Lower Shield Plug Laydown	256. Reheater Drain Tank	349. Moisture Separator	441. Fuel Pool Filter Demineralizer
75. Reactor Vessel Head Insulation Laydown	258. Turbine Bypass Valve Assembly	351. Desiccant Dryer	442. Cleanup Filter Demineralizer
76. Flux Monitor Shipping Crate	263. Moisture Separator Drain Tank	352. Dryer Regenerator	443. Waste Demineralizer
77. New Fuel Storage Vault	270. Moisture Separator Reheater	353. Gas Cooler	444. Floor Drain Demineralizer
79. Drywell Head Laydown	272. Louver Penthouse	354. Charcoal Adsorber	445. Distillate Polishing Demineralizer
82. Upper Pool Plug Laydown	273. Equipment Drain Sump Pump	355. Off Gas Chiller Unit	
83. Lower Pool Plug Laydown	274. Floor Drain Sump Pump		
84. Reactor Vessel Head Laydown			
			<u>Miscellaneous Equipment</u>
			700. Elevator
			701. Covered Hatch
			702. Open Hatch
			703. Railroad
			708. Solid Radwaste Container Dolly Rails

TABLE C.6-2. Equipment Key Numbers in Appendix C.

REFERENCES

1. WPPSS Nuclear Project No. 2 Final Safety Analysis Report, Washington Public Power Supply System, Richland, WA, February 1977.
2. WPPSS Nuclear Project No. 2 Environmental Report, Operating License Stage, Washington Public Power Supply System, Richland, WA, March 1977.
3. R. I. Smith, G. J. Konzek and W. E. Kennedy, Jr., Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station, NUREG/CR-0130, Appendix-I, U.S. Nuclear Regulatory Commission Report by Pacific Northwest Laboratory, June 1978.*

* Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and the National Technical Information Service, Springfield, Virginia 22161.

APPENDIX D

RADIATION DOSE RATE AND CONCRETE SURFACE CONTAMINATION DATA

The radiation dose rate at a specific work site has an important influence on the time needed to complete each decommissioning task. In addition, the degree of concrete contamination determines how much surface requires removal and how much contaminated rubble requires disposal. Established in this appendix are the dose rates and the concrete surface contamination levels that are assumed to be present in the reference BWR at final reactor shutdown, except for the dose rates resulting from activated material in and surrounding the reactor vessel. These latter dose rates are presented in Section E.1.3 of Appendix E.

D.1 ESTIMATED RADIATION DOSE RATES AT SHUTDOWN

Measured shutdown radiation dose rates were obtained from seven operational BWRs, three dual-unit plants and one single-unit plant. These plants are: Dresden Units 2 and 3 and Quad Cities Units 1 and 2 operated by Commonwealth Edison Company, Peachbottom Units 2 and 3 operated by Philadelphia Electric Company, and Monticello operated by Northern States Power Company. At the time of these measurements, the reactors had operated commercially for 3 to 8 years. Actual data on these dose rates and on their corresponding sources of radiation were provided by health physics personnel at all four plant sites. Composite values presented in this appendix are derived from these data.

D.1.1 Assumptions and Definitions

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

- The plant is in a shutdown/refueling mode, with no fuel movements in progress.
- "General area" refers to the radiation field not emanating specifically from one discrete source or direction in a room or area, although a

specific source may be the sole contributor to the radiation measurement. General area readings are taken at least 1 m from any surface.

- "Contact" means a dose rate at the closest approach to a given surface (a surface dose rate), including the necessary corrections for geometry and source size made in the field by the health physics technician.
- Estimated radiation dose rates in specific areas where measurements are unavailable are obtained from Reference 1. (The estimated dose rates from Reference 1 deal primarily with plant operation, and are used accordingly.)
- Crud traps that may exist at valves or at sharp bends in piping represent potential high dose rate areas that are not explicitly considered.

D.1.2 Presentation of Dose Rate Data

The radiation dose rate data assumed for the reference BWR at final shut-down are presented in Figures D.1-1 and D.1-2 for the Reactor Building (including Primary Containment), in Figures D.1-3 through D.1-5 for the Turbine Generator Building, and in Figures D.1-6 and D.1-7 for the Radwaste and Control Building. The measurement location is identified in Figures D.1-1 through D.1-7 by the equipment key number (see Section C.2 in Appendix C) and the piece of equipment for which the measurement was taken. The other buildings and structures on the reference BWR site (see Section C.1 in Appendix C) are assumed to have no potential for radiation exposure.

Although some of the data from the seven operating BWRs cover a fairly wide range, all of the dose rate data shown in the figures are composite values based on those data, except those obtained from Reference 1. In some instances, the dose rates vary by as much as a factor of 10 or more, depending on the plant operating histories and on the extent and success of prior decontamination programs. As shown in Figure D.1-8 (from Reference 2), and based on current technology only, much higher dose rates might be expected at the

REACTOR BLDG. - FLOOR PLAN AT ELEV. 128.7 m

REACTOR BLDG. - FLOOR PLAN
AT ELEV. 134.4 m AND 135.3 m
(GRADE)

REACTOR BLDG. - FLOOR PLAN AT ELEV. 143.6 m

REACTOR BLDG. - FLOOR PLAN AT ELEV. 152.7 m

EQUIPMENT KEY NUMBER	LOCATION	TYPE OF MEASUREMENT (d)	MEASURED DOSE RATE (R/hr)
2, 3	REACTOR PEDESTAL, VENT PIPES	GENERAL AREA	>.100 (b)
4	LPCS PUMP	CONTACT	.005 - .015
	LPCS PIPING	CONTACT	.010 - .180
5	HPCS PUMP	CONTACT	.002 - .008
	HPCS PIPING	CONTACT	.010 - .250
8	RHR PUMP	CONTACT	.018 - .027
	RHR PUMP PIPING	CONTACT	.035 - .055
	RHR PUMP	GENERAL AREA	.020 - .050
10	RCIC PUMP	CONTACT	.002 - .008
	RCIC PUMP PIPING	CONTACT	.050 - .060
	RCIC PUMP	GENERAL AREA	.005 - .010
17	RADWASTE BUILDING CONDENSATE SUPPLY PUMP	CONTACT	.018 - .027
18	CONDENSATE FILTER-DEMINERALIZER BACKWASH PUMP	CONTACT	.020 - .030
20	CONTROL ROD DRIVE PUMP FILTER	CONTACT	.080
21	CONTROL ROD DRIVE PUMP	CONTACT	.001
	CONTROL ROD DRIVE PUMP	GENERAL AREA	≤.015 (b)
	CONTROL ROD DRIVE PUMP PIPING	CONTACT	.015
27	CONTROL INSTRUMENT AIR TANK	GENERAL AREA	≤.0025 (b)
33	REACTOR WATER RECIRCULATION PUMP	CONTACT	.100 - .370
	PIPING AT BOTTOM OF PUMP	CONTACT	.035 - .060
	RETURN PIPING TO REACTOR	CONTACT	.080 - .090
	JET PUMP RISERS	CONTACT	.100 - .460
34	DRYWELL PERSONNEL LOCK	GENERAL AREA	.040 - .100
35	DRYWELL EQUIPMENT HATCH	GENERAL AREA	.100
	STEAM RELIEF VALVE FROM MAIN STEAM LINE (IN DRYWELL)	CONTACT	.040 - .043
	STEAM LINE AT RELIEF VALVE	CONTACT	.033 - .040
	MAIN STEAM LINE	CONTACT	.035 - .070
	DRYWELL	GENERAL AREA	.035 - .550
39	AIR COMPRESSOR	GENERAL AREA	<.005 (b)
40	MAIN STEAM TUNNEL	GENERAL AREA	.090 - .210
	MAIN STEAM PIPING AND VALVES	CONTACT	.005 - .040
42	TIP DRIVE MECHANISMS	CONTACT	.015
65	AIR LOCK	GENERAL AREA	.002 - .010
97	FLOOR DRAIN SUMP	CONTACT	.020 - .045
98	EQUIPMENT DRAIN SUMP	CONTACT	.005
100	FAN COIL UNIT	GENERAL AREA	.010
102	TIP ROOM	GENERAL AREA	.030 - .050
	TIP PROBE SHIELDS IN TIP ROOM	CONTACT	.050 - 2.0
107	HEATING AND VENTILATING UNIT	GENERAL AREA	>.100 (b)
124	AIR DRYER	GENERAL AREA	<.0025 (b)
700	ELEVATOR	GENERAL AREA	<.0025 (b)
701	COVERED HATCH	GENERAL AREA	<.0025 (b)
702	OPEN HATCH	GENERAL AREA	<.0025 (b)
703	RAILROAD TRACK	CONTACT	.004 - .040

(a) General Area refers to the radiation field in a room or area, not specifically from one discrete source or direction, although a specific source may be the sole contributor to the radiation measurement.
Contact means the closest approach to a surface (a surface dose rate) including the necessary geometry and source size corrections done in the field by the health physicist.
(b) Based on information in Reference 1, primarily associated with plant operation.

FIGURE D.1-1. Radiation Dose Rates
in the Reactor Building - Sheet 1

REACTOR BLDG. - FLOOR PLAN AT ELEV. 159.1 m

REACTOR BLDG. - FLOOR PLAN AT ELEV. 167.0 m

REACTOR BLDG. - FLOOR PLAN AT ELEV. 174.4 m

REACTOR BLDG. - FLOOR PLAN AT ELEV. 185.0 m

EQUIPMENT KEY NUMBER	LOCATION	TYPE OF MEASUREMENT (a)	MEASURED DOSE RATE (R/hr)
1	REACTOR VESSEL (NEAR THE FEED- WATER NOZZLES)	CONTACT	.700 - 3.0
2	REACTOR PEDESTAL	GENERAL AREA	>.100 (b)
46	CONTROL ROD DRIVE MODULES	CONTACT	.002 - .045
47	RWCU PUMPS	CONTACT	.100 - 12.0
	RWCU PIPING	CONTACT	.070 - .220
	RWCU DRAIN	CONTACT	.160 - .210
49	SACRIFICIAL SHIELD WALL	CONTACT	>.100 (b)
50	RBCC WATER PUMP	CONTACT	.005 - .040
	RBCC WATER PIPING	CONTACT	.010 - .080
	RBCC WATER PUMP	GENERAL AREA	.006 - .030
51	RBCC WATER HEAT EXCHANGER	CONTACT	.020 - .050
54	RHR HEAT EXCHANGER	CONTACT	.050 - .250
	RHR HEAT EXCHANGER PIPING	CONTACT	.800
	RHR HEAT EXCHANGER	GENERAL AREA	.160 - .600
55, 56	FUEL POOL COOLING EQUIPMENT	GENERAL AREA	>.100 (b)
58, 60	STANDBY LIQUID CONTROL EQUIPMENT	GENERAL AREA	<.0025 (b)
61	REGENERATIVE HEAT EXCHANGER	CONTACT	.300 - 10.0
	REGENERATIVE HEAT EXCHANGER PIPING	CONTACT	.200 - .500
	REGENERATIVE HEAT EXCHANGER	GENERAL AREA	.020 - .750
62	NON-REGENERATIVE HEAT EXCHANGER	CONTACT	.320 - 10.0
	NON-REGENERATIVE HEAT EXCHANGER PIPING	CONTACT	.400
	NON-REGENERATIVE HEAT EXCHANGER	GENERAL AREA	.050 - .750
63	REACTOR WELL POOL CAVITY	CONTACT	.015 - .060
66	STEAM DRYER	GENERAL AREA	>.100 (b)
67	SHROUD HEAT & STEAM SEPARATOR	GENERAL AREA	>.100 (b)
68	SPENT FUEL SHIPPING CASK STORAGE	GENERAL AREA	>.100 (b)
N/A(c)	REFUELING FLOOR	CONTACT	.002 - .500
74	UPPER AND LOWER SHIELD PLUG LAYDOWN	GENERAL AREA	<.0025 (b)
77	NEW FUEL STORAGE VAULT	GENERAL AREA	<.015 (b)
75, 76, 79, 82, 83, 84, 85, 88, 89	LAYDOWN AREAS FOR VARIOUS REACTOR PARTS	GENERAL AREA	<.0025 (b)
90	SPENT FUEL STORAGE POOL	GENERAL AREA	.002 - .020
91	RBCC WATER SURGE TANK	GENERAL AREA	.006 - .030
92	STANDBY GAS FILTER UNIT	GENERAL AREA	<.0025 (b)
93	DAMPERS (AIR OPERATORS)	GENERAL AREA	<.0025 (b)
94	HYDROGEN RECOMBINER	GENERAL AREA	<.015 (b)
95	EXHAUST AIR PLENUM	GENERAL AREA	<.015 (b)
96	SUMP VENT FILTER UNITS	GENERAL AREA	<.015 (b)
107	HEATING & VENTILATING UNIT	GENERAL AREA	<.015(b)
109	RELEASE STACK OPENING	GENERAL AREA	<.001 (b)
112	DRYER & SEPARATOR STORAGE POOL	GENERAL AREA	.002 - .020
700	ELEVATOR	GENERAL AREA	<.0025 (b)
701	COVERED HATCH	GENERAL AREA	<.0025 (b)
702	OPEN HATCH	GENERAL AREA	<.0025 (b)

- (a) General Area refers to the radiation field in a room or area, not specifically from one discrete source or direction, although a specific source may be the sole contributor to the radiation measurement.
Contact means the closest approach to a surface (a surface dose rate) including the necessary geometry and source size corrections done in the field by the health physicist.
(b) Based on information in Reference 1; primarily associated with plant operation.
(c) Indicates "not available."

FIGURE D.1-2. Radiation Dose Rates in the Reactor Building - Sheet 2

ELEVATION 134.4 m
(GRADE)

EQUIPMENT KEY NUMBER	LOCATION	TYPE OF MEASUREMENT (a)	MEASURED DOSE RATE (R/hr)
200	MAIN CONDENSER	CONTACT	.002
212	CONDENSATE PUMP PIPING	CONTACT	.002
212	CONDENSATE PUMP	GENERAL AREA	.002
213	CONDENSATE BOOSTER PUMP PIPING	CONTACT	.002
213	CONDENSATE BOOSTER PUMPS	GENERAL AREA	.002
214	REACTOR FEEDWATER PUMPS	CONTACT	.002
215	MECHANICAL VACUUM PUMP PIPING	CONTACT	.002 - .005
215	MECHANICAL VACUUM PUMPS	GENERAL AREA	.001 - .002
216	STEAM JET AIR EJECTOR CONDENSER	CONTACT	.002 - .030
217	GLAND SEAL STEAM CONDENSER	CONTACT	.001
218	CATALYTIC RECOMBINERS	CONTACT	.002 - .005
218	CATALYTIC RECOMBINERS PIPING	CONTACT	.002 - .015
226	SEAL WATER LIQUID TANK	GENERAL AREA	.002
227	PUMPED DRAIN TANK	GENERAL AREA	.002
228	CONDENSATE STORAGE TANKS	GENERAL AREA	.001
273	EQUIPMENT DRAIN SUMP	GENERAL AREA	> .100 (b)
274	FLOOR DRAIN SUMP	GENERAL AREA	> .100 (b)
700	ELEVATOR	GENERAL AREA	≤ .0025 (b)
703	RAILROAD TRACK	GENERAL AREA	≤ .0025 (b)

- (a) General Area refers to the radiation field in a room or area, not specifically from one discrete source or direction, although a specific source may be the sole contributor to the radiation measurement.
Contact means the closest approach to a surface (a surface dose rate) including the necessary geometry and source size corrections done in the field by the health physicist.
- (b) Based on the information in Reference 1; primarily associated with plant operation.

FIGURE D.1-3. Radiation Dose Rates in the Turbine Generator Building - Elevation 134.4 m

ELEVATION 143.6 m

EQUIPMENT KEY NUMBER	LOCATION	TYPE OF MEASUREMENT (a)	MEASURED DOSE RATE (R/hr)
200	MAIN CONDENSER	CONTACT	.002
224, 225	OFF-GAS CONDENSER, MOISTURE SEPARATOR	GENERAL AREA	.003 - .020
228	CONDENSATE STORAGE TANK	CONTACT	.001
252	HIGH-PRESSURE FEEDWATER HEATERS AND PIPING	CONTACT	.002 - .025
253	LOW-PRESSURE FEEDWATER HEATERS, PIPING	CONTACT	.002 - .015
254	STEAM EVAPORATORS	CONTACT	.002 - .010
255	EVAPORATOR DRAIN TANKS	CONTACT	.002 - .006
256	REHEATER DRAIN TANKS	CONTACT	.003
258	TURBINE BYPASS VALVE ASSEMBLY	CONTACT	.002
263	MOISTURE SEPARATOR DRAIN TANK	GENERAL AREA	.002
700	ELEVATOR	GENERAL AREA	<.0025 (b)
701	COVERED HATCH	GENERAL AREA	<.0025 (b)
702	OPEN HATCH	GENERAL AREA	<.0025 (b)

- (a) General Area refers to the radiation field in a room or area, not specifically from one discrete source or direction, although a specific source may be the sole contributor to the radiation measurement.
Contact means the closest approach to a surface (a surface dose rate) including the necessary geometry and source size corrections done in the field by the health physicist.
- (b) Based on the information in Reference 1; primarily associated with plant operation.

FIGURE D.1-4. Radiation Dose Rates in the Turbine General Building - Elevation 143.6 m

ELEVATION 152.7 m

EQUIPMENT KEY NUMBER	LOCATION	TYPE OF MEASUREMENT ^(a)	MEASURED DOSE RATE (R/hr)
201	TURBINE	GENERAL AREA	.001 - .002
202	MAIN STEAM AND FEEDWATER PIPE CHASE	GENERAL AREA	.002
253	LOW-PRESSURE FEEDWATER HEATERS	CONTACT	.002 - .015
270	MOISTURE SEPARATOR REHEATER	GENERAL AREA	.002
276	HEATING AND VENTILATING UNIT	GENERAL AREA	.001
700	ELEVATOR	GENERAL AREA	≤ .0025 (b)
701	COVERED HATCH	GENERAL AREA	≤ .0025 (b)
702	OPEN HATCH	GENERAL AREA	≤ .0025 (b)

- (a) General Area refers to the radiation field in a room or area, not specifically from one discrete source or direction, although a specific source may be the sole contributor to the radiation measured. Contact means the closest approach to a surface (a surface dose rate) including the necessary geometry and source size corrections done in the field by the health physicist.
- (b) Based on the information in Reference 1, primarily associated with plant operation

FIGURE D.1-5. Radiation Dose Rates in the Turbine Generator Building - Elevation 152.7 m

RADWASTE AND CONTROL BUILDING -
FLOOR PLAN AT ELEVATION 142.3 m

EQUIPMENT KEY NUMBER	LOCATION	TYPE OF MEASUREMENT ^(a)	MEASURED DOSE RATE (R/hr)
300	FLOOR DRAIN COLLECTOR TANK	CONTACT	.150 - 5.000
300	FLOOR DRAIN COLLECTOR TANK	GENERAL AREA	.050 - 1.600
301	WASTE COLLECTOR TANK	CONTACT	.110 - 1.200
302	CONDENSATE PHASE SEPARATOR TANKS	CONTACT	1.500
302	CONDENSATE PHASE SEPARATOR TANKS	GENERAL AREA	.200 - 1.000
304	CONDENSATE BACKWASH RECEIVING TANK	GENERAL AREA	.300
305	SPENT RESIN TANK	CONTACT	1.000 - 1.500
305	SPENT RESIN TANK	GENERAL AREA	.005 - .150
306	WASTE SLUDGE PHASE SEPARATOR TANK	GENERAL AREA	.005 - .450
306	WASTE SLUDGE PHASE SEPARATOR TANK PIPING	CONTACT	.500
307	CLEANUP PHASE SEPARATOR TANKS	GENERAL AREA	.025 - .075
308	CHEMICAL WASTE TANKS	GENERAL AREA	.240
309	DISTILLATE TANKS	GENERAL AREA	.150
310	DETERGENT DRAIN TANK	GENERAL AREA	.160
311	DECONTAMINATION SOLUTION CONCENTRATOR	GENERAL AREA	.500 - .620
312	WASTE SURGE TANK	CONTACT	.050 - .200
312	WASTE SURGE TANK	GENERAL AREA	.005 - .022
313	WASTE SAMPLE TANK, PIPING	CONTACT	.030 - .080
313	WASTE SAMPLE TANKS	GENERAL AREA	.050 - .080
314	FLOOR DRAIN SAMPLE TANK	CONTACT	.014
318	FLOOR DRAIN COLLECTOR PUMP (PUMP GALLERY)	GENERAL AREA	.080
319	WASTE COLLECTOR PUMP (PUMP GALLERY)	GENERAL AREA	.090
327	SPENT RESIN PUMP (PUMP GALLERY)	CONTACT	.090
330	WASTE SLUDGE DISCHARGE PUMP (PUMP GALLERY)	GENERAL AREA	.020 - .150
340	FLOOR DRAIN SUMP	CONTACT	.200
341	EQUIPMENT DRAIN SUMP	CONTACT	1.500
342	CHEMICAL DRAIN SUMP	CONTACT	.220
348	OFF GAS COOLER CONDENSERS	CONTACT	.150
351	DESICCANT DRYERS	CONTACT	.220
352	DRYER REGENERATORS	CONTACT	.130
353	OFF GAS COOLER	CONTACT	.170
354	OFF GAS CHARCOAL ADSORBERS	CONTACT	.100 (b)
355	OFF GAS CHILLER UNITS	GENERAL AREA	<.015 (b)
358	SOLID RADWASTE STORAGE AREA CRANF	GENERAL AREA	>.100 (b)
359	TRUCK LOADING CRANE	GENERAL AREA	<.0025 (b)
360	SOLID RADWASTE STORAGE	GENERAL AREA	.010 - .160

EQUIPMENT KEY NUMBER	LOCATION	TYPE OF MEASUREMENT ^(a)	MEASURED DOSE RATE (R/hr)
364	CONTAINER WASHDOWN SMEAR AND TESTING STATION	CONTACT	≥.100 (b)
366	SOLID RADWASTE CONTAINER CAPPING STATION	GENERAL AREA	≥.100 (b)
367	CONTAINER FILLING STATION	GENERAL AREA	>.100 (b)
371	SOLID RADWASTE BALER	GENERAL AREA	<.0025 (b)
377	EXPANSION TANK	GENERAL AREA	<.015 (b)
378	AIR HANDLING UNITS	GENERAL AREA	<.015 (b)
379	SOLID RADWASTE HOPPER MIXER (BOTTOM)	CONTACT	.500 - 3.500
379	SOLID RADWASTE HOPPER MIXER (SIDES)	CONTACT	.300 - .380
379	SOLID RADWASTE HOPPER MIXER	GENERAL AREA	.120 - 1.000
383	OFF GAS REFRIGERATION EQUIPMENT	GENERAL AREA	<.0025 (b)
384	PREFILTER	GENERAL AREA	≥.100 (b)
385	AFTERFILTER	GENERAL AREA	≥.100 (b)
393	CONTAINER FILLING STATION	GENERAL AREA	<.100 (b)
407	EQUIPMENT REMOVAL PLUG	GENERAL AREA	<.0025 (b)
408	CONCENTRATOR WASTE MEASURING TANK	GENERAL AREA	≥.100 (b)
409	RADWASTE CENTRIFUGES	CONTACT	.020 - .070
409	RADWASTE CENTRIFUGE PIPING	CONTACT	1.500
409	RADWASTE CENTRIFUGE ROOM	GENERAL AREA	.010 - .060
412	CHEMICAL SOLUTION TANK	GENERAL AREA	<.0025 (b)
413	CONDENSATE PRECOAT TANK	GENERAL AREA	<.0025 (b)
417	WASTE PRECOAT TANK	GENERAL AREA	<.0025 (b)
419	CHEMICAL ADDITION TANK	GENERAL AREA	<.0025 (b)
420	WASTE FILTER AID TANK	GENERAL AREA	<.0025 (b)
424	FUEL POOL PRECOAT TANK	GENERAL AREA	<.0025 (b)
425	RESIN TANK	GENERAL AREA	<.0025 (b)
432	FUEL POOL HOLD PUMP	GENERAL AREA	<.100(b)
433	CLEANUP HOLD PUMP	GENERAL AREA	<.100(b)
435	DECONTAMINATION SOLUTION CONCENTRATOR	GENERAL AREA	≥.100 (b)
700	BOTTOMS RECYCLE PUMP ELEVATOR	GENERAL AREA	<.0025 (b)
701	COVERED HATCH	GENERAL AREA	<.0025 (b)
702	OPEN HATCH	GENERAL AREA	<.0025 (b)
705	SOLID RADWASTE CONTAINER DOLLY RACK	GENERAL AREA	≥.100 (b)

(a) General Area refers to the radiation field in a room or area, not specifically from one discrete source or direction, although a specific source may be the sole contributor to the radiation measurement.
Contact means the closest approach to a surface (a surface dose rate) including the necessary geometry and source size corrections done in the field by the health physicist.
(b) Based on the information in Reference 1; primarily associated with plant operation.

RADWASTE AND CONTROL BUILDING -
N. W. MEZZANINE PLAN AT ELEVATION 137.8 m

RADWASTE AND CONTROL BUILDING -
FLOOR PLAN AT ELEVATION 133.2 m

FIGURE D.1-6. Radiation Dose Rates
in the Radwaste and Control Building -
Sheet 1

RADWASTE AND CONTROL BUILDING -
CONTROL ROOM TOWER PLAN
AT ELEVATION 160.0 m

RADWASTE AND CONTROL BUILDING -
FLOOR PLAN AT ELEVATION 154.5 m

RADWASTE AND CONTROL BUILDING -
FLOOR PLAN AT ELEVATION 148.4 m

EQUIPMENT KEY NUMBER	LOCATION	TYPE OF MEASUREMENT ^(a)	MEASURED DOSE RATE (R/hr)
378	AIR HANDLING UNIT	GENERAL AREA	<.0025 (b)
386	DECONTAMINATION SOLUTION CONCENTRATOR	GENERAL AREA	>.100 (b)
387	DECONTAMINATION SOLUTION CONCENTRATOR HEATING ELEMENT	GENERAL AREA	>.100 (b)
407	EQUIPMENT REMOVAL PLUG	GENERAL AREA	<.0025 (b)
438	CONDENSATE FILTER DEMINERALIZER	GENERAL AREA	>.100 (b)
439	FLOOR DRAIN FILTER	GENERAL AREA	>.100 (b)
440	WASTE COLLECTOR FILTER	GENERAL AREA	>.100 (b)
441	FUEL POOL FILTER DEMINERALIZER	GENERAL AREA	>.100 (b)
442	CLEANUP FILTER DEMINERALIZER	GENERAL AREA	>.100 (b)
443	WASTE DEMINERALIZER PUMPS	CONTACT	.016 - .040
443	WASTE DEMINERALIZER PIPING	CONTACT	.020 - 6.000
443	WASTE DEMINERALIZER	GENERAL AREA	.010 - .100
444	FLOOR DRAIN DEMINERALIZER	GENERAL AREA	>.100 (b)
445	DISTILLATE POLISHER DEMINERALIZER	GENERAL AREA	>.100 (b)
447	DECONTAMINATION SOLUTION CONCENTRATOR EVAPORATOR	CONTACT	.200 - .300
451	AIR HANDLING UNITS (FOR CONTROL ROOM)	GENERAL AREA	<.0025 (b)
453	RADWASTE AIR FILTER UNITS	CONTACT	.160
453	RADWASTE AIR FILTER UNITS	GENERAL AREA	.030
455	HEATING AND VENTILATION UNIT	GENERAL AREA	<.015 (b)
464	MAIN CONTROL BENCHBOARD	GENERAL AREA	<.001 (b)
465	RELAY CABINETS	GENERAL AREA	<.001 (b)
468	DECONTAMINATION STATION	GENERAL AREA	<.0025 (b)
469	RESIN ADDITION TANK	CONTACT	.002
470	CHILLERS	CONTACT	.001
473	FILTER DEMINERALIZER REMOVAL CRANE	GENERAL AREA	<.0025 (b)
700	ELEVATOR	GENERAL AREA	<.0025 (b)
701	COVERED HATCH	GENERAL AREA	<.0025 (b)
702	OPEN HATCH	GENERAL AREA	<.0025 (b)

(a) General Area refers to the radiation field in a room or area, not specifically from one discrete source or direction, although a specific source may be the sole contributor to the radiation measurement.
Contact means the closest approach to a surface (a surface dose-rate) including the necessary geometry and source size corrections done in the field by the health physicist.
(b) Based on the information in Reference 1; primarily associated with plant operation.

FIGURE D.1-7. Radiation Dose Rates
in the Radwaste and
Control Building -
Sheet 2

FIGURE D.1-8. Shutdown Radiation Levels on BWR Recirculation Pump Suction Piping (from Reference 2)

end of the reference BWR's operating life. However, there are a number of factors that tend to keep radiation dose rates within manageable bounds:

- During the operating lifetime of the plant, only relatively mild decontamination agents are used to periodically reduce dose rates. After final shutdown, harsher (more effective) decontamination agents and longer decontamination periods may be used, since protection of valve seats, valve operators, and various critical piping and equipment is no longer required.
- It is extremely important to minimize outage time during the plant's operating lifetime because of the cost of replacement power. Hence, because of a lack of available time, decontamination of some portions of the plant may not be as effective as desired. After final shutdown, if

maximum decontamination is not achieved and high radiation readings are still being encountered, the time is available to repeat decontamination procedures or to try other decontamination agents. Such a combination of decontamination approaches is often extremely effective, thus greatly reducing dose rates.

- Current decontamination studies are focused on improved techniques for operating reactors.⁽³⁾ However, it is reasonable to expect future studies to be focused on the more severe measures appropriate for decontamination during decommissioning.
- In addition to improved chemical decontamination agents, various types of mechanical decontaminating techniques are being developed and improved, such as high-pressure water-jet systems, electrolytic systems, and sonic vibrational systems.

In summation, the nuclear power industry's knowledge of decontamination methods is increasing rapidly. Therefore, it is assumed that the radiation dose rates shown in Figures D.1-1 through D.1-7 reasonably represent the dose rates that workers will encounter in performing various tasks in each of the designated areas of the reference BWR at the time of decommissioning.

D.1.3 Application of Radiation Dose Rate Data

The entire project required to complete a particular decommissioning mode is broken down into individual tasks to be performed in each specific area, and the man-hours of radiation exposure are then estimated for each task. In applying the dose rate data presented in this appendix, adjustments are made for ALARA considerations, for decontamination during decommissioning, for special shielding methods, and for other appropriate judgment considerations, as described in Appendices H, I, J, and K. The adjusted dose rate for each specific work site is then multiplied by the estimated man-hours of exposure for each task. After correcting for radioactive decay, the individual task exposures are totaled for the entire decommissioning effort. Radioactive decay of ^{60}Co is the basis in this study for calculating the decay factor for each task. Additional information on the makeup of the surface contamination affecting radiation dose rates in the plant is contained in Section E.2.2 of Appendix E.

D.2 ESTIMATED CONCRETE SURFACE CONTAMINATION AT SHUTDOWN

Measured concrete surface contamination levels were obtained from the same four BWR sites: Desden, Quad Cities, Peachbottom, and Monticello. Study assumptions relating to concrete surface contamination, together with composite values from the measured levels at the four sites and estimated concrete rubble volumes for the reference BWR, are presented in the following subsections.

D.2.1 Assumptions

For this study, the following assumptions are made concerning concrete surface contamination at shutdown:

- The plant is in a shutdown/refueling mode, with no fuel movements in progress.
- "Low-level" or "high-level" surface contamination in a particular area is based on the plant process carried out in, and the accessibility and/or occupancy of, that area during plant operation. Contamination at specific locations within a given area may vary significantly from that in the general area.
- The percentage of the total area that is assumed contaminated is based on the measured data and on the plant process carried out in that location.
- Most contaminated concrete surfaces in the plant are floors; other contaminated surfaces (i.e., walls or curbs) are specifically noted.
- The spalling depth for reinforced concrete surfaces is 0.051 m. (This represents the minimum depth of concrete cover on reinforcing bar at the reference BWR and is presumed sufficient to remove all measurable contamination.)
- Contaminated concrete block walls or concrete curbs are completely broken up and removed as contaminated rubble.
- A contaminated floor area made up of one or more removable concrete plugs is completely removed as contaminated material (i.e., no spalling or fracturing takes place).

D.2.2 Concrete Surface Contamination Data

The data on concrete surface contamination are presented in Figures D.2-1 and D.2-2 for the Reactor Building (including Primary Containment), in Figures D.2-3 through D.2-5 for the Turbine Generator Building, and in Figures D.2-6 and D.2-7 for the Radwaste and Control Building. The equipment key number (see Section C.2 in Appendix C) of an associated piece of equipment is used to identify the location of a particular contaminated area; it is shown as the "Associated Equipment Key Number" in Figures D.2-1 through D.2-7. No concrete surface contamination is assumed in the other buildings and structures at the reference BWR site.

As are the dose rate data, the concrete surface contamination data are presented as composite values of the measured levels, although the data from the operating BWRs sometimes exhibits a fairly wide range. The data vary by a factor of 10 or more between plants, depending on the plant operating histories and on the extent and success of prior decontamination efforts.

New epoxy resin treatments for concrete surfaces are making decontamination and surface renewal easier and quicker. Also, equipment for decontamination is being invented or improved. In addition, new maintenance procedures that reduce the spread of surface contamination are constantly being devised. As a result of these considerations, the concrete surface contamination data and the contaminated concrete areas reported in Figures D.2-1 through D.2-7 are assumed to be reasonably representative of those anticipated to be present at the end of the reference BWR's operating life.

D.2.3 Application of Concrete Surface Contamination Data

The measured concrete surface contamination data are not specifically applied to any analysis in this study, except to help determine the "assumed contamination area" in a specific location and to determine, in those instances where plant process does not dictate, specific areas of concrete surface that will probably need to be decontaminated during decommissioning. (In this study, contaminated concrete surfaces are assumed to be decontaminated to acceptable levels only by removing the surface either with a concrete spaller

or with other physical-removal means.) The volume of concrete rubble or material that requires removal for surface decontamination is calculated as the product of 1) the total surface area associated with the particular piece of equipment, 2) the "percent of area assumed contaminated," and 3) the thickness of the material to be removed.

REACTOR BLDG - FLOOR PLAN AT ELEV. 128.7m

REACTOR BLDG - FLOOR PLAN
AT ELEV. 134.4m AND 135.3m
(GRADE)

ASSOCIATED EQUIPMENT KEY NUMBER	LOCATION	MEASURED CONTAMINATION LEVEL (a) (cpm/100 cm ²)	SURFACE AREA (b) (m ²)	PERCENT OF AREA ASSUMED CONTAMINATED	RUBBLE VOLUME (c) (m ³)
2, 3	SUPPRESSION CHAMBER	0.3 - 2.5 k (d)	2459.0 (e)	100	125.4 (1)
4	LOW PRESSURE CORE SPRAY PUMP ROOM	0.2 - 5 k	67.6	20	0.7
5	HIGH PRESSURE CORE SPRAY PUMP ROOM	0.2 - 22 k	109.6	20	1.1
8	RESIDUAL HEAT REMOVAL PUMP ROOMS	0.4 - 100 k	381.2	20	3.9
10	REACTOR CORE ISOLATION COOLING PUMP ROOM	0.5 - 5.5 k	73.4	20	0.7
17, 18	RADWASTE BUILDING CONDENSATE PUMP AND FILTER BACKWASH PUMP RM	N/A (f)	162.8	20	1.7
21	CONTROL ROD DRIVE PUMP ROOM	0.2 - 12 k	107.0	20	1.1
33	REACTOR WATER RECIRCULATION PUMP AREA (DRYWELL FLOOR)	2 - 2000 k	418.0	100	21.3 (1)
N/A (f)	UNDER VESSEL AREA, ELEV. 152.7 m	4 - 150 k	179.4 (e)	100	9.1 (1)
34	DRYWELL PERSONNEL LOCK ROOM	0.2 - 30 k	25.5	100	1.3
35	DRYWELL EQUIPMENT HATCH ROOM	0.4 - 2 k	68.4	100	3.5
40	MAIN STEAM TUNNEL	0.1 - 12.5 k	295.0	100	15.0
N/A	MAIN STEAM TUNNEL CEILING PLUG	0.1 - 12.5 k	26.8	100	40.8 (h)
42	TRAVERSING IN-CORE PROBE DRIVE AREA	1.8 - 9.5 k	9.1	20	0.1
N/A	DS (DECONTAMINATION STATION), ELEV. 152.7 m	N/A	19.6	100	1.2 (h)
N/A	CRD REPAIR ROOM, ELEV. 152.7 m	0.6 - 35 k	68.7	10	0.4
N/A	PIPE SPACES, ELEV. 152.7 m	N/A	19.2	100	1.0
97	FLOOR DRAIN SUMPS	N/A	73.6	100	3.7
98	EQUIPMENT DRAIN SUMP	N/A	18.2	100	0.9
102	TRAVERSING IN-CORE PROBE (TIP) ROOM	4 - 80 k	40.4	20	0.4
700	ELEVATOR AREA	-- (g)	-- (g)	-- (g)	-- (g)
701	COVERED HATCH AREA	--	--	--	--
702	OPEN HATCH AREA	--	--	--	--
703	RAILROAD AIRLOCK	0.2 - 0.4 k	266.4	15 (j)	2.0
Total					235.3

- (a) Composite of measurements taken during maintenance outages at operating BWRs
 (b) Floor area unless otherwise indicated.
 (c) Based on a contamination thickness of 0.051 m
 (d) 0.3 - 2.5 k stands for 300 to 2,500 cpm/100 cm² (typical).
 (e) Includes other surface areas beside the floor area
 (f) Indicates "not available."
 (g) Shown for reference only; negligible contamination assumed.
 (h) Includes additional volume due to removal and shipment of entire ceiling plug or concrete curb
 (i) Rubble volumes from within Primary Containment
 (j) Area beneath spent fuel cask loading hatch (EKN 702).

REACTOR BLDG - FLOOR PLAN AT ELEV. 143.6m

REACTOR BLDG - FLOOR PLAN AT ELEV. 152.7m

FIGURE D.2-1. Concrete Surface Contamination in the Reactor Building - Sheet 1

REACTOR BLDG. - FLOOR PLAN AT ELEV. 159.1m

REACTOR BLDG. - FLOOR PLAN AT ELEV. 167.0m

REACTOR BLDG. - FLOOR PLAN AT ELEV. 174.4m

REACTOR BLDG. - FLOOR PLAN AT ELEV. 185.0m

ASSOCIATED EQUIPMENT KEY NUMBER	LOCATION	MEASURED CONTAMINATION LEVEL (a) (cpm/100 cm ²)	SURFACE AREA (b) (m ²)	PERCENT OF AREA ASSUMED CONTAMINATED	RUBBLE VOLUME (c) (m ³)
46	CONTROL ROD DRIVE MODULE AREAS	0.2 - 20 k(d)	318.6	50	8.1
47	REACTOR WATER CLEANUP PUMP ROOMS	1 - >500 k	169.1(e)	100	8.6
N/A(f)	PIPE SPACES, ELEV. 159.1 m	N/A(f)	72.0	100	3.7
50, 51	REACTOR BUILDING CLOSED COOLING EQUIPMENT AREA	0.5 k	117.1	10	0.6
54	PIPE SPACES, ELEV. 167.0 m	1.5 - 10 k	64.4	100	3.3
	ELEV. 167.0 m	0.4 - 300 k	107.0	50	2.7
	ELEV. 174.4 m	2.5 - 300 k	75.9	50	2.0
55, 56	FUEL POOL PUMP AND HX ROOM	0.4 - 18 k	75.8	50	2.0
58, 60	STANDBY LIQUID CONTROL EQUIPMENT AREA	--(g)	--(f)	--(f)	--(f)
61, 62	REACTOR WATER CLEANUP REGENERATIVE AND NON-REGENERATIVE HX ROOM	Floor 7 - 77 k	147.2	50	3.8
		Walls 4 - 8 k	327.0	10	1.7
63	REACTOR WELL CAVITY	N/A	272.8(e)	100	13.9
85	REACTOR VESSEL HEAD LAYDOWN AND SPENT FUEL CASK WASHDOWN AREA	15 - 32 k	69.1	100	4.5(h)
N/A	REFUELING FLOOR, ELEV. 185.0 m	0.2 - 15 k	1 115.4	10	5.7
90	SPENT FUEL POOL	N/A	691.6(e)	100	35.3
92	STANDBY GAS FILTER UNIT AREA	0.1 - 40 k	202.2	20	2.1
94	HYDROGEN RECOMBINER AREAS	N/A	76.2	20	0.4
96	SUMP VENT FILTER AREA	N/A	47.6	20	0.5
107	HEATING AND VENTILATING UNIT AREA	--(g)	--	--	--
112	DRYER AND SEPARATOR STORAGE POOL	N/A	511.7(e)	100	26.1
700	ELEVATOR AREA	--(g)	--(g)	--(g)	--(g)
701	COVERED HATCH AREA	--	--	--	--
702	OPEN HATCH AREA	--	--	--	--
TOTAL					125.0

- (a) Composite of measurements taken during maintenance outages at operating BWRs.
 (b) Floor area unless otherwise indicated.
 (c) Based on a contamination thickness of 0.051 m.
 (d) 0.2 - 20 k stands for 200 to 20,000 cpm/100 cm² (typical).
 (e) Includes other surface areas besides the floor area.
 (f) Indicates "not available."
 (g) Shown for reference only, negligible contamination assumed.
 (h) Includes additional volume due to removal and shipment of entire concrete curb.

FIGURE D.2-2. Concrete Surface Contamination in the Reactor Building - Sheet 2

ELEVATION 134.4 m
(GRADE)

ASSOCIATED EQUIPMENT KEY NUMBER	LOCATION	MEASURED CONTAMINATION LEVEL (a) (cpm/100 cm ²)	SURFACE AREA (b) (m ²)	PERCENT OF AREA ASSUMED CONTAMINATED	RUBBLE VOLUME (c) (m ³)
200	MAIN CONDENSER AREA	0.2 - 2.5 k ^(d)	1 664.8 ^(e)	10	8.5
212, 217	CONDENSATE PUMP AND GLAND SEAL	0.2 - 25 k	217.0	10	1.1
213	STEAM CONDENSER AREA	--(g)	--(g)	--(g)	--(g)
214	CONDENSATE BOOSTER PUMP	--(g)	--(g)	--(g)	--(g)
214, N/A ^(f)	REACTOR FEEDWATER PUMP ROOMS	0.5 - 9 k	1 369.0 ^(e)	50	34.9
	DS (DECONTAMINATION STATION)	N/A	9.5	100	0.8 ^(h)
215	ELEV. 134.4 m				
215	MECHANICAL VACUUM PUMP AREA	0.3 - 4.3 k	79.0	50	2.0
216	STEAM JET AIR EJECTOR ROOMS	0.1 - 100 k	121.0	50	3.1
218	CATALYTIC RECOMBINER ROOM	0.2 - 20 k	356.9 ^(e)	50	9.1
226, 227	GLAND SEAL AND PUMPED DRAIN TANKS AREA	N/A	18.6	100	1.3 ^(h)
228	CONDENSATE STORAGE TANK ENCLOSURE	N/A	662.4	5	1.7
273	EQUIPMENT DRAIN SUMPS	20 k	25.1 ^(e)	100	1.3
274	FLOOR DRAIN SUMPS	20 k	25.1 ^(e)	100	1.3
N/A	SAMPLE ROOM, ELEV. 134.4 m	N/A	30.8	20	1.9 ^(h)
700	ELEVATOR AREA	--(g)	--	--	--
703	RAILROAD TRACK AREA	N/A	--	--	--
TOTAL					67.0

- (a) Composite of measurements taken during maintenance outages at operating BWRs.
 (b) Floor area unless otherwise indicated.
 (c) Based on a contamination thickness of 0.051 m.
 (d) 0.2 - 2.5 k stands for 200 to 2,500 cpm/100 cm² (typical).
 (e) Includes other surface areas besides the floor area.
 (f) Indicates "not available."
 (g) Shown for reference only; negligible contamination assumed.
 (h) Includes additional volume due to removal and shipment of entire concrete curbs or concrete block walls.

FIGURE D.2-3. Concrete Surface Contamination in the Turbine Generator Building - Elevation 134.4 m

ELEVATION 143.6 m

ASSOCIATED EQUIPMENT KEY NUMBER	LOCATION	MEASURED CONTAMINATION LEVEL (cpm/100 cm ²)	SURFACE AREA ^(b) (m ²)	PERCENT OF AREA ASSUMED CONTAMINATED	RUBBLE VOLUME ^(c) (m ³)
200	MAIN CONDENSER AREA	0.2 - 2.1 k ^(d)	391.2	20	4.0
252	HIGH-PRESSURE FEEDWATER HEATER AREA	0.2 k	324.8	20	3.3
253	LOW-PRESSURE FEEDWATER HEATER AREA	0.2 - 2.1 k	763.2	20	7.8
254, 255	STEAM EVAPORATORS AND EVAPORATOR DRAIN TANK AREA	N/A ^(f)	319.4	20	3.3
256, 263	REHEATER DRAIN TANKS AND MOISTURE SEPARATOR AREA	N/A	908.7 ^(e)	10	4.6
258	TURBINE BY-PASS VALVE ASSEMBLY AREA	30 - 100 k	436.6	20	4.4
700	ELEVATOR	-- ^(g)	-- ^(g)	-- ^(g)	-- ^(g)
701	COVERED HATCH	--	--	--	--
702	OPEN HATCH	--	--	--	--
TOTAL					27.4

- (a) Composite of measurements taken during maintenance outages at operating BWRs.
 (b) Floor area unless otherwise indicated.
 (c) Based on a contamination thickness of 0.051 m.
 (d) 0.2 - 2.1 k stands for 200 to 2,100 cpm/100 cm² (typical).
 (e) Includes surface areas from a below-elevation pipe chase.
 (f) Indicates "not available."
 (g) Shown for reference only; negligible contamination assumed.

FIGURE D.2-4. Concrete Surface Contamination in the Turbine Generator Building - Elevation 134.4 m

ASSOCIATED EQUIPMENT KEY NUMBER	LOCATION	MEASURED CONTAMINATION LEVEL (a) (cpm/100 cm ²)	SURFACE AREA (b) (m ²)	PERCENT OF AREA ASSUMED CONTAMINATED	RUBBLE VOLUME (c) (m ³)
201	TURBINE AREA	0.1 - 0.4 k (d)	652.2	10	3.3
253	LOW-PRESSURE FEEDWATER HEATER AREA	N/A (e)	10 184.0	10	4.8
270	MOISTURE SEPARATOR REHEATER AREA	0.1 - 1.0 k	645.9	10	3.3
202	MAIN STEAM AND FEEDWATER PIPE CHASE	-- (f)	-- (f)	-- (f)	-- (f)
276	HEATING AND VENTILATION UNIT	--	--	--	--
700	ELEVATOR AREA	--	--	--	--
701	COVERED HATCH AREA	--	--	--	--
702	OPEN HATCH AREA	--	--	--	--
TOTAL					11.4

- (a) Composite of measurements taken during maintenance outages at operating BWRs.
 (b) Floor area unless otherwise indicated.
 (c) Based on a contamination thickness of 0.051 m.
 (d) 0.1 - 0.4 k stands for 100 to 400 cpm/100 cm² (typical).
 (e) Indicates "not available".
 (f) Shown for reference only, negligible contamination assumed.

FIGURE D.2-5. Concrete Surface Contamination in the Turbine Generator Building - Elevation 152.7 m

RADWASTE AND CONTROL BUILDING -
FLOOR PLAN AT ELEVATION 142.3m

RADWASTE AND CONTROL BUILDING -
N. W. MEZZANINE PLAN AT ELEVATION 137.8m

ASSOCIATED EQUIPMENT KEY NUMBER	LOCATION	MEASURED CONTAMINATION LEVEL (a) (cpm/100 cm ²)	SURFACE AREA (b) (m ²)	PERCENT OF AREA ASSUMED CONTAMINATED	RUBBLE VOLUME (c) (m ³)
300	FLOOR DRAIN COLLECTOR TANK AREA	4 - 250 k (d)	92.0	50	2.4
301	WASTE COLLECTOR TANK AREA	4 - 250 k	92.0	50	2.4
302	CONDENSATE PHASE SEPARATOR TANK AREA	4 - 250 k	106.4(e)	100	6.3(f)
304	CONDENSATE BACKWASH RECEIVING TANK ARE	4 - 250 k	68.5(e)	100	4.1(f)
N/A(g)	PUMP GALLERY, ELEV 133.2 m	4 - 250 k	195.7	50	5.0
305	SPENT RESIN TANK AREA	4 - 250 k	24.4(e)	100	2.4(f)
306	WASTE SLUDGE PHASE SEPARATOR TANK AREA	4 - 250 k	62.0(e)	100	4.7(f)
307	CLEANUP PHASE SEPARATOR TANK AREA	1.1 - 130 k	63.7(e)	100	4.7(f)
308, 309, 310	CHEMICAL WASTE, DISTILLATE AND DETERGENT DRAIN TANKS AREA	0.1 - 120 k	149.0	50	3.8
311	DECONTAMINATION SOLUTION CONCENTRATOR WASTE TANK ROOMS	N/A(g)	33.7	100	1.7
312, 313	WASTE SURGE TANK AND WASTE SAMPLE TANK AREA	0.2 - 8 k	21.9	10	2.2
314	FLOOR DRAIN SAMPLE TANK AREA	0.2 - 8 k	6.3	10	0.6
340	FLOOR DRAIN SUMPS	N/A	26.4	100	1.5
341	EQUIPMENT DRAIN SUMP	N/A	8.8	100	0.5
342	CHEMICAL DRAIN SUMP	N/A	6.9	100	0.4
348	OFF GAS COOLER CONDENSER ROOM	N/A	46.8	50	1.2
351, 352	DESICCANT DRYER AND REGENERATOR ROOMS	N/A	71.0(e)	100	5.6(f)
353, 354	OFF GAS COOLER AND OFF GAS CHARCOAL ADSORBER ROOM	N/A	11.8	100	5.7
N/A	VALVE ROOM, ELEV 133.2 m	N/A	33.8	50	0.9
355	OFF GAS CHILLER UNITS AREA	--(h)	--(h)	--(h)	--(h)
360	SOLID RADWASTE STORAGE AREA	0.5 - 150 k	390.6(e)	100	19.9

RADWASTE AND CONTROL BUILDING -
FLOOR PLAN AT ELEVATION 133.2m

ASSOCIATED EQUIPMENT KEY NUMBER	LOCATION	MEASURED CONTAMINATION LEVEL (a) (cpm/100 cm ²)	SURFACE AREA (b) (m ²)	PERCENT OF AREA ASSUMED CONTAMINATED	RUBBLE VOLUME (c) (m ³)
364, 366, 367	SOLID RADWASTE CONTAINER WASHDOWN, CAPPING, AND FILLING STATION ROOM	1.5 - 15 k	108.7	100	5.5
371	SOLID WASTE BALER AREA	0.4	16.0	50	0.4
N/A	DECONTAMINATION AREA, ELEV 133.2 m	1.5 - 20 k	53.9	100	3.3(f)
378	AIR HANDLING UNIT ROOM	N/A	45.0	100	2.3
379	SOLID RADWASTE HOPPER MIXER ROOM	0.6 - 90 k	73.6	100	3.7
384, 385	PREFILTER AND AFTERFILTER ROOMS	N/A	60.4	100	3.1
407	EQUIPMENT REMOVAL PLUG AREA	--	--	--	--
408	CONCENTRATOR WASTE MEASURING TANK ROOM	80 k	22.3	100	1.1
409	RADWASTE CENTRIFUGE ROOMS	2.4 - 300 k	172.6(e)	100	8.8
N/A	CONTAMINATED TOOL ROOM, ELEV 142.3 m	N/A	193.1(e)	50	13.1(f)
412 to 425	CHEMICAL MAKEUP ROOM	--	--	--	--
432	CLEANUP HOLD PUMP AREAS, VALVE AND PUMP ROOMS, ELEV 142.3 m	2.8 - 10 k	205.3	50	5.6(f)
433	FUEL POOL HOLD PUMP ROOMS	2.5 - 200 k	159.4	50	4.1
435	DECONTAMINATION SOLUTION CONCENTRATOR BOTTOMS RECYCLE PUMP ROOMS	N/A	27.9(e)	100	14.6(f)
700	ELEVATOR AREA	--	--	--	--
701	COVERED HATCH AREA	--	--	--	--
702	OPEN HATCH AREA	--	--	--	--
TOTAL					141.6

- (a) Composite of measurements taken during maintenance outages at operating BWRs
(b) Floor area unless otherwise indicated
(c) Based on a contamination thickness of 0.051 m
(d) 4 - 250 k stands for 4,000 to 250,000 cpm/100 cm² (typical)
(e) Includes other surface areas besides the floor area
(f) Includes additional volume due to removal and shipment of entire concrete block wall(s) or concrete curb
(g) Indicates "not available"
(h) Shown for reference only, negligible contamination assumed

FIGURE D.2-6. Concrete Surface Contamination in the Radwaste and Control Building - Sheet 1

RADWASTE AND CONTROL BUILDING -
CONTROL ROOM TOWER PLAN
AT ELEVATION 160.0 m

RADWASTE AND CONTROL BUILDING -
FLOOR PLAN AT ELEVATION 154.5 m

RADWASTE AND CONTROL BUILDING -
FLOOR PLAN AT ELEVATION 148.4 m

ASSOCIATED EQUIPMENT KEY NUMBER	LOCATION	MEASURED CONTAMINATION LEVEL (a) (cpm/100 cm ²)	SURFACE AREA (b) (m ²)	PERCENT OF AREA ASSUMED CONTAMINATED	RUBBLE VOLUME (c) (m ³)
386, 387	DECONTAMINATION SOLUTION CONCENTRATOR AND HEATING ELEMENT ROOMS, ELEV 145.4 m AND ELEV 148.4 m	N/A (d)	43.7	100	2.2
407	EQUIPMENT REMOVAL PLUGS AND FILTER DEMINERALIZER REMOVAL ROOM, ELEV 154.5 m	0.2 - 6.2 k(e)	606.1	20	21.1(f)
438	CONDENSATE FILTER DEMINERALIZER ROOMS	N/A	110.6	100	5.6
439, 444	FLOOR DRAIN FILTER AND DEMINERALIZER ROOMS	N/A	31.6	100	1.6
440, 443	WASTE COLLECTOR FILTER AND WASTE DEMINERALIZER ROOMS	N/A	31.6	100	1.6
441	FUEL POOL FILTER DEMINERALIZER ROOMS	N/A	31.6	100	1.6
442	CLEANUP FILTER DEMINERALIZER ROOMS	N/A	31.6	100	1.6
445	DISTILLATE POLISHING DEMINERALIZER ROOM	N/A	15.8	100	0.8
468	DECONTAMINATION STATION	N/A	7.4(g)	100	0.7(f)
N/A	HOT MACHINE SHOP, ELEV 148.4 m	<0.1	137.5	50	16.7(f)
N/A	HOT STORAGE, ELEV 148.4 m	0.1 - 50 k	19.0	100	1.0
N/A	RADIOCHEMISTRY LABORATORY, ELEV 148.4 m	N/A	93.2	20	0.9
N/A	SAMPLING AREA, ELEV 148.4 m	N/A	81.6	20	0.8
N/A	HOT INSTRUMENT SHOP, ELEV 148.4 m	N/A	110.3(g)	100	5.6
700	ELEVATOR AREA	--(h)	--(h)	--(h)	--(h)
701	COVERED HATCH AREA	--	--	--	--
702	OPEN HATCH AREA	--	--	--	--

TOTAL

61.8

- (a) Composite of measurements taken during maintenance outages at operating BWRs
 (b) Floor area unless otherwise indicated
 (c) Based on a contamination thickness of 0.051 m
 (d) Indicates "not available"
 (e) 0.2 - 6.2 k stands for 200 to 6,200 cpm/100 cm² (typical)
 (f) Includes additional volume due to removal and shipment of entire equipment removal plugs, concrete block walls, or concrete curbs
 (g) Includes other surface areas besides the floor area

FIGURE D.2-7. Concrete Surface Contamination in the Radwaste and Control Building - Sheet 2

REFERENCES

1. WPPSS Nuclear Project No. 2 Final Safety Analysis Report, Washington Public Power Supply System, Chapter 12.3, Richland, WA, March 1977.
2. S. G. Sawochka, N. P. Jacob and W. L. Pearl, Primary System Shutdown Radiation Levels at Nuclear Power Generation Stations, EPRI-404-2, Figure 3, December 1975.
3. R. L. Dillion et al., Literature Review of Dilute Chemical Decontamination Processes for Water-Cooled Nuclear Reactors, EPRI NP-1033, Prepared by Battelle, Pacific Northwest Laboratories for Electric Power Research Institute, March 1979.

APPENDIX E

RADIONUCLIDE INVENTORIES

The estimated radionuclide inventories at the shutdown, defueled, reference BWR plant are presented in this appendix, together with supporting information and details of the calculational methods for estimating these inventories. The radionuclide inventories at the time of final reactor shutdown (excluding the irradiated spent fuel) are of two types: 1) neutron-activated components in and surrounding the reactor core, and 2) surface contamination from fission products and activated corrosion products deposited inside certain piping and equipment systems, on some structural surfaces, and on the site.

It should be recognized that the data in this appendix are calculated estimates unique to the reference BWR and are not directly applicable to any other specific BWR.

E.1 NEUTRON-ACTIVATED COMPONENTS

The interaction of the neutrons produced in the reactor fuel with the structural materials in and around the reactor vessel radioactivates the elements in these structural materials. Detailed in this section are methods and results of, and supporting information for, the calculations that estimate the inventories and associated dose rates of the neutron-activated radionuclides in these materials.

E.1.1 Radionuclide Inventories in Neutron-Activated Structural Materials

The types and quantities of radionuclides in the activated reactor component materials at the end of the operating life of the reference BWR are derived using the methodology presented in Reference 1. The calculational models employ a homogenized zone of fuel, coolant, and control rods. The fuel in the models has an average exposure of 12,000 MWD/MTU (the midpoint in the equilibrium BWR fuel-burnup cycle). The radionuclide inventories and the methods used to generate them are discussed below.

E.1.1.1 Reference Radionuclide Inventories 1, 2, and 3

Three basic types of materials are used in and around the reactor vessel: stainless steel (type 304), carbon steel (type SA533), and concrete. Reference radionuclide inventories for these materials at final reactor shutdown are presented as follows: Table E.1-1 for stainless steel (reference radionuclide inventory 1), Table E.1-2 for carbon steel (reference radionuclide inventory 2), and Table E.1-3 for concrete (reference radionuclide inventory 3).^(a) These inventories are calculated based on the material densities of the reference BWR, the thermal neutron flux distribution at the axial midplane of the fuel zone, and 30 effective full-power years (EFPY) of operation. They represent maximum values for these materials.

These reference radionuclide inventories are used for several purposes in this study. In this section they are used to calculate the total quantities of radioactivity in the structural components in and around the reactor vessel. This information is used in determining disposal requirements and costs whenever removal of any or all of these components is required by a particular decommissioning mode. In Appendix N these inventories are used in assessing public safety of decommissioning activities.

E.1.1.2 Calculational Methods

Reference 1 describes a series of steps to calculate the inventories of neutron-activated radionuclides reported above. First, the material densities and neutron cross sections of the materials in the homogenized fuel zone and in the structures surrounding the reactor core are calculated. These data are then used to generate neutron flux distributions in the fuel zone and the surrounding components. Finally, the radionuclide inventories in the major components surrounding the fuel zone in the radial direction are calculated using the neutron flux values and the material densities. More details follow.

(a)Reference radionuclide inventory 3 for neutron-activated concrete also contains radionuclides associated with the carbon-steel structural reinforcing beams in the sacrificial shield.

**TABLE E.1-1. Reference Radionuclide Inventory 1,
Neutron-Activated Stainless Steel(a)**

Radionuclide	Radioactivity Concentration at Shutdown (Ci/m ³)	Fractional Radioactivity at Decay Times of:				
		Shutdown	10 Years	30 Years	50 Years	100 Years
³ H	4.65 x 10 ⁻⁴	1.63 x 10 ⁻¹⁰	--- (b)	---	---	---
¹⁰ Be	2.63 x 10 ⁻⁶	---	---	---	---	---
¹⁴ C	1.05 x 10 ²	3.68 x 10 ⁻⁵	3.65 x 10 ⁻⁵	3.65 x 10 ⁻⁵	3.65 x 10 ⁻⁵	3.64 x 10 ⁻⁵
³² P	1.11 x 10 ²	3.89 x 10 ⁻⁵	---	---	---	---
³³ P	6.65 x 10 ²	2.33 x 10 ⁻⁴	---	---	---	---
³⁵ S	5.52 x 10 ¹	1.94 x 10 ⁻⁵	---	---	---	---
³⁶ Cl	2.69 x 10 ⁻⁴	---	---	---	---	---
⁵¹ Cr	1.45 x 10 ⁶	5.09 x 10 ⁻¹	---	---	---	---
⁵⁴ Mn	8.50 x 10 ³	2.98 x 10 ⁻³	7.02 x 10 ⁻⁷	---	---	---
⁵⁵ Fe	9.22 x 10 ⁵	3.24 x 10 ⁻¹	2.25 x 10 ⁻²	1.09 x 10 ⁻⁴	8.63 x 10 ⁻⁷	---
⁵⁹ Fe	2.74 x 10 ⁴	9.61 x 10 ⁻³	---	---	---	---
⁵⁸ Co	2.10 x 10 ⁴	7.37 x 10 ⁻³	---	---	---	---
⁶⁰ Co	3.36 x 10 ⁵	1.18 x 10 ⁻¹	3.11 x 10 ⁻²	2.26 x 10 ⁻³	1.64 x 10 ⁻⁴	2.20 x 10 ⁻⁷
⁵⁹ Ni	6.36 x 10 ²	2.23 x 10 ⁻⁴	2.23 x 10 ⁻⁴	2.23 x 10 ⁻⁴	2.23 x 10 ⁻⁴	2.23 x 10 ⁻⁴
⁶³ Ni	8.75 x 10 ⁴	3.07 x 10 ⁻²	2.85 x 10 ⁻²	2.45 x 10 ⁻²	2.17 x 10 ⁻²	1.45 x 10 ⁻²
⁶⁵ Zn	3.23 x 10 ¹	1.13 x 10 ⁻⁵	3.68 x 10 ⁻¹⁰	---	---	---
⁹³ Zr	8.15 x 10 ⁻⁶	---	---	---	---	---
⁹⁵ Zr	1.41 x 10 ⁻²	4.95 x 10 ⁻⁹	---	---	---	---
^{93m} Nb	1.35 x 10 ⁻¹	4.74 x 10 ⁻⁸	6.74 x 10 ⁻⁸	9.19 x 10 ⁻⁸	9.30 x 10 ⁻⁸	9.61 x 10 ⁻⁸
⁹⁴ Nb	1.50 x 10 ⁰	5.26 x 10 ⁻⁷	5.26 x 10 ⁻⁷	5.26 x 10 ⁻⁷	5.26 x 10 ⁻⁷	5.26 x 10 ⁻⁷
⁹⁵ Nb	1.20 x 10 ²	4.21 x 10 ⁻⁵	---	---	---	---
⁹³ Mo	3.26 x 10 ⁻¹	1.14 x 10 ⁻⁷	1.14 x 10 ⁻⁷	1.14 x 10 ⁻⁷	1.14 x 10 ⁻⁷	1.13 x 10 ⁻⁷
⁹⁹ Tc	3.18 x 10 ⁻²	1.12 x 10 ⁻⁸	1.12 x 10 ⁻⁸	1.12 x 10 ⁻⁸	1.12 x 10 ⁻⁸	1.12 x 10 ⁻⁸
^{108m} Ag	7.36 x 10 ⁻²	2.58 x 10 ⁻⁸	2.45 x 10 ⁻⁸	2.20 x 10 ⁻⁸	1.98 x 10 ⁻⁸	1.52 x 10 ⁻⁸
¹⁰⁸ Ag	8.67 x 10 ¹	3.04 x 10 ⁻⁵	2.45 x 10 ⁻⁸	2.20 x 10 ⁻⁸	1.98 x 10 ⁻⁸	1.52 x 10 ⁻⁸
^{109m} Ag	3.61 x 10 ⁰	1.27 x 10 ⁻⁶	4.49 x 10 ⁻⁹	---	---	---
¹⁰⁹ Cd	3.42 x 10 ⁰	1.20 x 10 ⁻⁶	4.49 x 10 ⁻⁹	---	---	---
^{110m} Ag	8.04 x 10 ⁰	2.82 x 10 ⁻⁶	1.27 x 10 ⁻¹⁰	---	---	---
¹¹⁰ Ag	2.02 x 10 ²	7.09 x 10 ⁻⁵	---	---	---	---
¹⁵¹ Sm	2.12 x 10 ⁻²	7.44 x 10 ⁻⁹	6.88 x 10 ⁻⁹	5.86 x 10 ⁻⁹	5.13 x 10 ⁻⁹	3.36 x 10 ⁻⁹
¹⁵² Eu	1.12 x 10 ⁻³	3.93 x 10 ⁻¹⁰	2.20 x 10 ⁻¹⁰	---	---	---
¹⁵⁴ Eu	3.12 x 10 ⁰	1.09 x 10 ⁻⁶	7.12 x 10 ⁻⁷	2.99 x 10 ⁻⁷	1.59 x 10 ⁻⁸	1.44 x 10 ⁻⁸
¹⁶⁰ Tb	9.48 x 10 ⁻⁴	3.33 x 10 ⁻¹⁰	---	---	---	---
^{166m} Ho	7.84 x 10 ⁻⁴	2.75 x 10 ⁻¹⁰	2.73 x 10 ⁻¹⁰	2.71 x 10 ⁻¹⁰	2.67 x 10 ⁻¹⁰	2.60 x 10 ⁻¹⁰
Totals	2.85 x 10⁶	1.00	8.25 x 10⁻²	2.71 x 10⁻²	2.22 x 10⁻²	1.47 x 10⁻²

(a) Calculated at the inner surface of the 304 stainless steel core shroud, at the axial midplane of the fuel zone, for 30 EFY of operation.

(b) Indicates a value of less than 1.00 x 10⁻¹⁰.

**TABLE E.1-2. Reference Radioactivity Inventory 2,
Neutron-Activated Carbon Steel(a)**

Radionuclide	Radioactivity Concentration at Shutdown (Ci/m ³)	Fractional Radioactivity at Decay Times of:				
		Shutdown	10 Years	30 Years	50 Years	100 Years
¹⁴ C	6.77 x 10 ⁻³	1.84 x 10 ⁻⁵	1.84 x 10 ⁻⁵	1.84 x 10 ⁻⁵	1.83 x 10 ⁻⁵	1.82 x 10 ⁻⁵
³² P	9.20 x 10 ⁻¹	2.51 x 10 ⁻³	---(b)	---	---	---
³⁵ S	1.76 x 10 ⁻²	4.80 x 10 ⁻⁵	---	---	---	---
⁵¹ Cr	1.75 x 10 ⁰	4.77 x 10 ⁻³	---	---	---	---
⁵⁴ Mn	1.06 x 10 ¹	2.89 x 10 ⁻²	6.84 x 10 ⁻⁶	---	---	---
⁵⁵ Fe	3.35 x 10 ²	9.13 x 10 ⁻¹	6.35 x 10 ⁻²	3.08 x 10 ⁻⁴	2.43 x 10 ⁻⁶	---
⁵⁹ Fe	9.44 x 10 ⁰	2.57 x 10 ⁻²	---	---	---	---
⁵⁸ Co	1.49 x 10 ⁰	4.06 x 10 ⁻³	---	---	---	---
⁶⁰ Co	6.49 x 10 ⁰	1.77 x 10 ⁻²	4.74 x 10 ⁻³	3.41 x 10 ⁻⁴	2.46 x 10 ⁻⁵	3.35 x 10 ⁻⁸
⁵⁹ Ni	1.46 x 10 ⁻²	3.98 x 10 ⁻⁵	3.98 x 10 ⁻⁵	3.98 x 10 ⁻⁵	3.98 x 10 ⁻⁵	3.98 x 10 ⁻⁵
⁶³ Ni	1.73 x 10 ⁰	4.71 x 10 ⁻³	4.39 x 10 ⁻³	3.76 x 10 ⁻³	3.33 x 10 ⁻³	2.23 x 10 ⁻³
⁶⁵ Zn	1.88 x 10 ⁻⁶	5.12 x 10 ⁻⁹	---	---	---	---
⁸⁹ Sr	5.24 x 10 ⁻¹³	1.43 x 10 ⁻¹⁵	---	---	---	---
⁹¹ Y	2.25 x 10 ⁻¹³	6.13 x 10 ⁻¹⁶	---	---	---	---
⁹⁵ Zr	1.62 x 10 ⁻⁴	4.41 x 10 ⁻⁷	---	---	---	---
^{93m} Nb	3.90 x 10 ⁻⁴	1.06 x 10 ⁻⁶	1.51 x 10 ⁻⁶	2.10 x 10 ⁻⁶	2.12 x 10 ⁻⁶	2.16 x 10 ⁻⁶
⁹⁴ Nb	8.30 x 10 ⁻⁷	2.26 x 10 ⁻⁹	2.26 x 10 ⁻⁹	2.26 x 10 ⁻⁹	2.25 x 10 ⁻⁹	2.25 x 10 ⁻⁹
⁹⁵ Nb	3.76 x 10 ⁻⁴	1.02 x 10 ⁻⁶	---	---	---	---
⁹³ Mo	9.39 x 10 ⁻⁴	2.56 x 10 ⁻⁶	2.56 x 10 ⁻⁶	2.54 x 10 ⁻⁶	2.54 x 10 ⁻⁶	2.54 x 10 ⁻⁶
⁹⁹ Tc	2.64 x 10 ⁻⁴	7.19 x 10 ⁻⁷	7.19 x 10 ⁻⁷	7.19 x 10 ⁻⁷	7.19 x 10 ⁻⁷	7.19 x 10 ⁻⁷
Totals	3.67 x 10²	1.00	7.27 x 10⁻²	4.47 x 10⁻³	3.42 x 10⁻³	2.29 x 10⁻³

(a) Calculated at the inner surface of the SA533 carbon steel reactor vessel, at the axial midplane of the fuel zone, for 30 EFY of operation.

(b) Indicates a value of less than 1.00 x 10⁻¹⁰.

TABLE E.1-3. Reference Radionuclide Inventory 3,
Neutron-Activated Concrete(a)

Radionuclide	Radioactivity Concentration at Shutdown (Ci/m ³)	Fractional Radioactivity at Decay Times of:				
		Shutdown	10 Years	30 Years	50 Years	100 Years
³ H	2.58 x 10 ⁻⁶	6.72 x 10 ⁻⁷	3.83 x 10 ⁻⁷	1.23 x 10 ⁻⁷	4.04 x 10 ⁻⁸	2.40 x 10 ⁻⁹
¹⁴ C	1.36 x 10 ⁻⁴	3.54 x 10 ⁻⁵	3.52 x 10 ⁻⁵	3.52 x 10 ⁻⁵	3.50 x 10 ⁻⁵	3.49 x 10 ⁻⁵
³² P	4.35 x 10 ⁻²	1.13 x 10 ⁻²	---(b)	---	---	---
³⁵ S	3.25 x 10 ⁻³	8.46 x 10 ⁻⁴	---	---	---	---
³⁶ Cl	1.21 x 10 ⁻⁶	3.15 x 10 ⁻⁷	3.15 x 10 ⁻⁷	3.15 x 10 ⁻⁷	3.15 x 10 ⁻⁷	3.15 x 10 ⁻⁷
³⁷ Ar	3.08 x 10 ⁻²	8.02 x 10 ⁻³	---	---	---	---
³⁹ Ar	5.69 x 10 ⁻³	1.48 x 10 ⁻³	1.45 x 10 ⁻³	1.37 x 10 ⁻³	1.30 x 10 ⁻³	1.15 x 10 ⁻³
⁴⁰ K	3.67 x 10 ⁻⁵	9.55 x 10 ⁻⁶	9.55 x 10 ⁻⁶	9.56 x 10 ⁻⁶	9.56 x 10 ⁻⁶	9.56 x 10 ⁻⁶
⁴¹ Ca	7.90 x 10 ⁻⁴	2.06 x 10 ⁻⁴	2.06 x 10 ⁻⁴	2.06 x 10 ⁻⁴	2.06 x 10 ⁻⁴	2.06 x 10 ⁻⁴
⁴⁵ Ca	3.91 x 10 ⁻¹	1.02 x 10 ⁻¹	2.22 x 10 ⁻⁸	---	---	---
⁴⁶ Sc	2.10 x 10 ⁻⁶	5.47 x 10 ⁻⁷	---	---	---	---
⁵¹ Cr(c)	1.11 x 10 ⁻²	2.89 x 10 ⁻³	---	---	---	---
⁵⁴ Mn(c)	9.60 x 10 ⁻³	2.50 x 10 ⁻³	5.89 x 10 ⁻⁷	---	---	---
⁵⁵ Fe(c)	3.15 x 10 ⁰	8.20 x 10 ⁻¹	5.68 x 10 ⁻²	2.76 x 10 ⁻⁴	2.18 x 10 ⁻⁶	---
⁵⁹ Fe(c)	1.01 x 10 ⁻¹	2.63 x 10 ⁻²	---	---	---	---
⁵⁸ Co(c)	3.00 x 10 ⁻³	7.81 x 10 ⁻⁴	---	---	---	---
⁶⁰ Co(c)	6.45 x 10 ⁻²	1.68 x 10 ⁻²	4.51 x 10 ⁻³	3.23 x 10 ⁻⁴	2.34 x 10 ⁻⁵	3.18 x 10 ⁻⁸
⁵⁹ Ni(c)	1.24 x 10 ⁻⁴	3.23 x 10 ⁻⁵	3.23 x 10 ⁻⁵	3.23 x 10 ⁻⁵	3.23 x 10 ⁻⁵	3.23 x 10 ⁻⁵
⁶³ Ni(c)	1.47 x 10 ⁻²	3.83 x 10 ⁻³	3.54 x 10 ⁻⁹	3.05 x 10 ⁻³	2.71 x 10 ⁻³	1.80 x 10 ⁻³
^{93m} Nb	3.05 x 10 ⁻⁶	7.94 x 10 ⁻⁷	1.13 x 10 ⁻⁶	1.56 x 10 ⁻⁶	1.58 x 10 ⁻⁶	1.61 x 10 ⁻⁶
⁹³ Mo	7.36 x 10 ⁻⁶	1.92 x 10 ⁻⁶	1.91 x 10 ⁻⁶	1.91 x 10 ⁻⁶	1.90 x 10 ⁻⁶	1.90 x 10 ⁻⁶
⁹⁹ Tc	4.08 x 10 ⁻⁶	1.06 x 10 ⁻⁶	1.06 x 10 ⁻⁶	1.06 x 10 ⁻⁶	1.06 x 10 ⁻⁶	1.06 x 10 ⁻⁶
^{108m} Ag	9.73 x 10 ⁻⁸	2.53 x 10 ⁻⁸	2.40 x 10 ⁻⁸	2.16 x 10 ⁻⁸	1.94 x 10 ⁻⁸	1.49 x 10 ⁻⁸
¹⁰⁸ Ag	1.24 x 10 ⁻⁴	3.23 x 10 ⁻⁵	2.40 x 10 ⁻⁸	2.16 x 10 ⁻⁸	1.94 x 10 ⁻⁸	1.49 x 10 ⁻⁸
^{109m} Ag	2.56 x 10 ⁻¹⁰	6.67 x 10 ⁻¹¹	---	---	---	---
¹⁰⁹ Cd	2.17 x 10 ⁻¹⁰	5.65 x 10 ⁻¹¹	---	---	---	---
^{110m} Ag	6.83 x 10 ⁻⁵	1.78 x 10 ⁻⁵	8.05 x 10 ⁻¹⁰	---	---	---
¹¹⁰ Ag	1.74 x 10 ⁻³	4.53 x 10 ⁻⁴	---	---	---	---
¹⁵¹ Sm	1.30 x 10 ⁻³	3.39 x 10 ⁻⁴	3.13 x 10 ⁻⁴	2.66 x 10 ⁻⁴	2.34 x 10 ⁻⁴	1.53 x 10 ⁻⁴
¹⁵² Eu	1.00 x 10 ⁻²	2.60 x 10 ⁻³	1.54 x 10 ⁻³	5.51 x 10 ⁻⁴	1.96 x 10 ⁻⁴	1.47 x 10 ⁻⁵
¹⁵⁴ Eu	1.31 x 10 ⁻³	3.41 x 10 ⁻⁴	1.46 x 10 ⁻⁴	2.70 x 10 ⁻⁵	4.98 x 10 ⁻⁶	7.27 x 10 ⁻⁸
^{166m} Ho	3.91 x 10 ⁻⁶	1.02 x 10 ⁻⁶	1.01 x 10 ⁻⁶	1.00 x 10 ⁻⁶	9.91 x 10 ⁻⁷	9.63 x 10 ⁻⁷
Totals	3.84 x 10 ⁰	1.00	6.50 x 10 ⁻²	6.15 x 10 ⁻³	4.76 x 10 ⁻³	3.41 x 10 ⁻³

(a) Calculated at the inner surface of the concrete portion of the sacrificial shield, at the axial mid-plane of the fuel zone, for 30 EFPY of operation.

(b) Indicates a value of less than 1.00 x 10⁻¹⁰.

(c) Due largely to structural steel in the sacrificial shield.

Material Densities and Cross Sections. The material densities (isotopic compositions) in the homogenized, exposed fuel zone are calculated using the ORIGEN code.⁽²⁾ Value of thermal and epithermal cross sections for the fuel-zone materials are generated by the BRT and EGGNIT codes.^(3,4) Data for BRT and EGGNIT come from the ENDF/B-IV files.⁽⁵⁾ For these particular calculations, the ENDF/B-IV data are consolidated into 30 energy groups by the ET0G code.⁽⁶⁾ Four sets of cross sections for the homogenized, exposed fuel zone are calculated:

1. zero percent coolant voiding with control rods inserted
2. zero percent coolant voiding with control rods withdrawn
3. seventy percent coolant voiding with control rods inserted
4. seventy percent coolant voiding with control rods withdrawn.

The material densities used in calculating the neutron flux distributions in the fuel zone and the surrounding structural components are presented in Table E.1-4.

Neutron Flux Distributions. The distributions of neutron flux throughout the fuel zone and the surrounding components are calculated using the ANISN code.⁽⁷⁾ The radial model is based on the material distributions at the axial midplane of the fuel zone and is shown in Figure E.1-1. The axial model is along the axial centerline of the fuel zone and is presented in Figure E.1-2.

The radial neutron flux distribution is calculated using each aforementioned fuel-zone cross section set to determine which set produces the highest flux in the region surrounding the fuel zone. As might be expected, calculations using fuel-zone cross section set 4 produces the highest flux in this region. The radial thermal neutron flux distribution calculated using set 4 is shown in Figure E.1-3.

The axial neutron flux distribution is calculated using a combination of the fuel-zone cross section sets to model the control rod positions and the steam voiding distribution (see Figure E.1-2). The calculated axial thermal neutron flux distribution is presented in Figure E.1-4.

TABLE E.1-4. Material Densities for Calculating Neutron Flux Distributions

Material Densities in Homogenized Fuel Zone (atoms/cm ³)(a)				
Isotope	0% Coolant Void, Control Rods in	0% Coolant Void, Control Rods out	70% Coolant Void, Control Rods in	70% Coolant Void, Control Rods out
²³⁵ U	6.449 x 10 ¹⁹	6.449 x 10 ¹⁹	6.449 x 10 ¹⁹	6.449 x 10 ¹⁹
²³⁶ U	1.363 x 10 ¹⁹	1.363 x 10 ¹⁹	1.363 x 10 ¹⁹	1.363 x 10 ¹⁹
²³⁸ U	5.205 x 10 ²¹	5.205 x 10 ²¹	5.205 x 10 ²¹	5.205 x 10 ²¹
²³⁹ Pu	2.113 x 10 ¹⁹	2.113 x 10 ¹⁹	2.113 x 10 ¹⁹	2.113 x 10 ¹⁹
²⁴⁰ Pu	7.246 x 10 ¹⁸	7.246 x 10 ¹⁸	7.246 x 10 ¹⁸	7.246 x 10 ¹⁸
²⁴¹ Pu	2.674 x 10 ¹⁸	2.674 x 10 ¹⁸	2.674 x 10 ¹⁸	2.674 x 10 ¹⁸
²⁴² Pu	5.808 x 10 ¹⁷	5.808 x 10 ¹⁷	5.808 x 10 ¹⁷	5.808 x 10 ¹⁷
¹³⁵ Xe	6.325 x 10 ¹⁴	6.325 x 10 ¹⁴	6.325 x 10 ¹⁴	6.325 x 10 ¹⁴
Zr	7.488 x 10 ²²	6.654 x 10 ²¹	7.488 x 10 ²²	6.654 x 10 ²¹
¹⁰ B	1.674 x 10 ²⁰	0.0	1.674 x 10 ²⁰	0.0
O	2.530 x 10 ²²	2.557 x 10 ²²	1.517 x 10 ²²	1.525 x 10 ²²
H	2.894 x 10 ²²	2.948 x 10 ²²	8.683 x 10 ²¹	8.844 x 10 ²¹

Material Densities in Structural Materials Surrounding the Fuel Zone (wt%)(b)			
Element	304 Stainless Steel	SA533 Carbon Steel	Concrete(b)
Cr	19.0	---(c)	---
Fe	67.9	99.3	3.4
Ni	10.0	0.5	0.02
C	---	0.2	---
Ca	---	---	7.2
Si	---	---	27.6
K	---	---	1.7
Al	---	---	4.0
Na	---	---	1.5
O	---	---	43.7
H	---	---	0.5

(a) Fuel exposed to 12,000 MWD/MTU.

(b) Includes carbon steel at the material density of the reinforcing steel in the concrete region of the sacrificial shield.

(c) Indicates none of that particular element in that particular structural material.

REGION	RADIAL DISTANCE TO INNER BOUNDARY, m
FUEL ZONE	0.0
VOID	2.377
WATER	2.579
WATER	2.630
JET PUMPS AND WATER	2.959
WATER	3.086
VESSEL CLADDING	3.210
VESSEL WALL	3.215
VOID	3.378
SACRIFICIAL SHIELD INNER SHELL	3.937
CONCRETE AND STRUCTURAL STEEL	3.957
OUTER SHELL	4.566
OUTER BOUNDARY	4.614

FIGURE E.1-1. BWR Model for Radial Neutron Flux Distribution Calculation

AXIAL COMPONENT		AXIAL LOCATION OF OUTER BOUNDARY (m)	MATERIAL VOLUME FRACTIONS					
			STAINLESS STEEL	ZIRCONIUM	VOID	BORON CARBIDE	WATER	STEAM
STEAM SEPARATOR RISERS		5.233	0.040	0.000	0.000	0.000	0.624	0.336
SHROUD HEAD PLATE		2.972	0.933	0.000	0.000	0.000	0.020	0.047
STEAM DOME		2.921	0.000	0.000	0.000	0.000	0.300	0.700
		2.499						
FUEL ELEMENT HANDLE		2.398	0.014	0.000	0.000	0.000	0.296	0.690
TOP FUEL GUIDE		2.398	0.069	0.048	0.000	0.000	0.265	0.618
TIE PLATE		2.285	0.359	0.160	0.000	0.000	0.144	0.337
END PIN		2.263	0.079	0.160	0.000	0.000	0.228	0.533
END CAP		2.242	0.069	0.370	0.000	0.000	0.168	0.393
		2.235						
PLENUM AND SPRING			0.079	0.120	0.240	0.000	0.168	0.393
		2.032						
GAS PLENUM			0.010	0.120	0.240	0.000	0.190	0.440
		1.829						
			0.000	0.120	0.000	0.000	0.195	0.435
UPPER FUEL ZONE		1.130	0.000	0.120	0.000	0.000	0.239	0.391
		0.560						
			0.000	0.120	0.000	0.000	0.311	0.319
(AXIAL MIDPLANE)		0.000						
			0.000	0.120	0.000	0.0004	0.4199	0.2097
LOWER FUEL ZONE		-0.750	0.000	0.120	0.000	0.0006	0.5413	0.0881
		-1.250						
			0.000	0.120	0.000	0.0014	0.6223	0.0063
END CAP		-1.829	0.000	0.370	0.000	0.0078	0.6222	0.0000
		-1.848						
TIE PLATE		-1.860	0.350	0.200	0.000	0.0078	0.4422	0.0000
FUEL SUPPORT PIECE			0.250	0.000	0.000	0.0078	0.7422	0.0000
		-2.057						
CORE SUPPORT PLATE			0.580	0.000	0.000	0.0078	0.4122	0.0000
		-2.108						
FUEL ELEMENT FLOW ORIFICE			0.109	0.000	0.000	0.0078	0.8834	0.0000
		-2.210						
CONTROL RODS AND GUIDE TUBES			0.064	0.000	0.000	0.0073	0.9287	0.0000
		-5.486						
		-6.121	0.071	0.000	0.000	0.0000	0.9290	0.0000

FIGURE E.1-2. BWR Model for Axial Neutron Flux Distribution Calculation

FIGURE E.1-3. Radial Thermal Neutron Flux Distribution at Core Axial Midplane

FIGURE E.1-4. Axial Thermal Neutron Flux Distribution

Radionuclide Inventories. The neutron flux values at the components of interest surrounding the fuel zone are used in ORIGEN production-depletion calculations to obtain the radionuclide inventories in these various reactor components after 30 EFPY of reactor operation. The material compositions used in these calculations are shown in Table E.1-5.

The calculated buildups of selected radionuclides in the core shroud are illustrated in Figure E.1-5, with the concentration of each radionuclide normalized to unity at 30 EFPY. As might be expected, the shorter-lived radionuclides, such as ^{55}Fe and ^{60}Co , reach an equilibrium concentration after about 23 years, while the concentrations of the long-lived radionuclides, such as ^{59}Ni and ^{94}Nb , increase almost linearly with increased irradiation time.

For perspective, the Elk River Reactor had operated for about 2.5 EFPY when it was dismantled. Based on the calculations for this study, the levels of the long-lived radionuclides at Elk River were less than 10% of those that will be present in the reference BWR after 30 EFPY. The shorter-lived radionuclides like ^{55}Fe and ^{60}Co probably reached about 50% and 30%, respectively, of their 30-EFPY values at Elk River.

The calculated radioactivity concentrations in the core shroud, the reactor pressure vessel, and in the sacrificial shield at the time of final reactor shutdown are the bases for the reference radionuclide inventories presented in Section E.1.1.1. Along with the principal radionuclides from activation of major material constituents, the activation products from a number of elements that are present only in trace amounts ($\leq$ a few hundred parts per million by weight) are taken into consideration. Some of these activated trace elements have long half-lives and can contribute to the external dose rate after about 100 years of decay, as is the case of ^{94}Nb .

E.1.2 Estimated Total Radioactivity in Neutron-Activated Components

The information presented in Section E.1.1 is used to estimate the total radioactivity in the reactor vessel and in the structural components in and around it. Only those components immediately surrounding the fuel zone are considered since the neutron flux levels decrease rapidly outside of that

TABLE E.1-5. Material Compositions Used in Radionuclide Inventory Calculations

Element	Isotope	Initial Isotope Abundance	Nominal Elemental Weight Percent in		
			Internals 304 Stainless	Reactor Vessel SA533 Steel	Sacrificial Shield Concrete
H	¹ H ² H	99.985 0.015	---(a)	---	0.53
C	¹² C ¹³ C	98.89 1.11	0.08	0.25	0.03
N	¹⁴ N ¹⁵ N	99.64 0.36	0.04	0.006	0.0013
O	¹⁶ O ¹⁷ O ¹⁸ O	99.756 0.039 0.205	---	---	43.7
Na	²³ Na	100.0	---	---	1.5
Mg	²⁴ Mg ²⁵ Mg ²⁶ Mg	78.00 10.0 11.01	---	---	0.23
Al	²⁷ Al	100.0	---	---	4.0
Si	²⁸ Si ²⁹ Si ³⁰ Si	92.2 4.7 3.1	1.00	0.04	27.6
P	³¹ P	100.0	0.0005	0.035	0.0045
S	³² S ³³ S ³⁴ S ³⁶ S	95.0 0.75 4.2 0.015	0.03	0.04	0.12
K	³⁹ K ⁴⁰ K ⁴¹ K	93.3 0.012 6.7	---	---	1.7
Ca	⁴⁰ Ca ⁴² Ca ⁴³ Ca ⁴⁴ Ca ⁴⁶ Ca ⁴⁸ Ca	96.94 0.65 0.14 2.08 0.003 0.19	---	---	7.2
Cr	⁵⁰ Cr ⁵² Cr ⁵³ Cr ⁵⁴ Cr	4.35 83.79 9.5 2.36	17.0	0.52	0.064
Fe	⁵⁴ Fe ⁵⁶ Fe ⁵⁷ Fe ⁵⁸ Fe	5.8 91.7 2.14 0.31	68.5	94.43	13.4 (includes reinforcing steel)
Mn	⁵⁵ Mn	100.0	2.0	2.85	0.19
Ni	⁵⁸ Ni ⁶⁰ Ni ⁶¹ Ni ⁶⁴ Ni	68.3 26.1 1.1 0.9	10.0	0.73	0.094
Co	⁵⁹ Co	100.0	0.05-0.15	0.006-0.012	0.00073
Cu	⁶³ Cu ⁶⁵ Cu	69.2 30.8	0.15	0.11	0.014
Mo	⁹² Mo ⁹⁴ Mo ⁹⁵ Mo ⁹⁶ Mo ⁹⁷ Mo ⁹⁸ Mo ¹⁰⁰ Mo	14.8 9.1 15.9 16.7 9.5 24.4 9.6	0.05	0.64	0.082
Nb	⁹³ Nb	100.0	0.016	---	---
Ag	¹⁰⁷ Ag ¹⁰⁹ Ag	51.8 48.2	0.00004	---	0.00008
Cs	¹³³ Cs	100.0	---	0.01	---
Sm	¹⁴⁴ Sm ¹⁴⁷ Sm ¹⁴⁸ Sm ¹⁴⁹ Sm ¹⁵⁰ Sm ¹⁵² Sm ¹⁵⁴ Sm	3.1 15.1 11.3 13.9 7.4 26.7 22.6	0.000001	---	0.0008
Eu	¹⁵¹ Eu ¹⁵³ Eu	47.9 52.1	0.0001	---	0.000008
Ho	¹⁶⁵ Ho	100.0	0.000001	---	0.00008

(a) Indicates none included in calculations

FIGURE E.1-5. Calculated Buildup of Selected Activation Products in the Core Shroud as a Function of Time at Full Power

zone. Several steps are required to compute the total radioactivity in each of the activated components. First, a graphical integration is performed on the component's longitudinal neutron flux distribution to obtain an average flux level over the activated length of the component. (For each component that radially surrounds the fuel zone, the longitudinal neutron flux distribution is assumed to be similar to the fuel-zone axial-centerline distribution shown in Figure E.1-4. For the axially distributed components, such as the top fuel guide and the core support plate, the longitudinal flux distribution

is assumed to be similar to the radial-midplane flux distribution shown in Figure E.1-3.) This average flux level is then divided by the peak flux level of the distribution to obtain a multiplication factor that, when multiplied with the point radioactivity concentration given in Section E.1 (Table E.1-1 for stainless steel components, Table E.1-2 for carbon steel components, and Table E.1-3 for concrete components), gives a longitudinally averaged radioactivity concentration for the particular component. In like manner, a similar factor is found for the lateral flux distribution through that component. This factor is multiplied with the longitudinally averaged concentration to obtain a volume-averaged radioactivity concentration. Multiplying this average radioactivity concentration by the activated volume of the particular component gives the total radioactivity in that component.

Table E.1-6 lists the volume-averaged radioactivity concentration, the volume, and the estimated radioactivity calculated for each structural component that is assumed to contain a significant amount of neutron-activated radioactivity. The radioactivity totals in the reactor vessel and its internal components range from about 0.5 Ci in a single control rod guide tube to about 6.3 million Ci in the core shroud. The total radioactivity in the sacrificial

TABLE E.1-6. Estimated Total Radioactivity in Neutron-Activated Structural Components

Component (quantity)	Volume-Averaged Radioactivity Concentration (Ci/m ³)	Estimated Activated Volume (m ³)	Radioactivity per Component (Ci)	Estimated Total Radioactivity (Ci)
Core Shroud (1)	1.68×10^6	3.75	6.30×10^6	6.30×10^6
Jet Pump Assembly (10)	2.62×10^4	0.076	2.00×10^3	2.00×10^4
Reactor Vessel (1)				
Cladding	1.07×10^3	0.428	4.58×10^2	2.16×10^3
Shell Wall	1.12×10^2	15.26	1.70×10^3	
Sacrificial Shield (1)				
Inner Shell	4.99×10^1	2.19	1.03×10^2	1.66×10^2
Reinforced Concrete Region	4.73×10^{-2}	73.30	3.47×10^0	
Outer Shell	8.66×10^0	6.22	5.39×10^1	
Steam Separator Assembly (1)				
Shroud Head Plate	1.03×10^4	0.841	8.65×10^3	9.60×10^3
Steam Separator Risers	2.53×10^3	0.376	9.52×10^2	
Top Fuel Guide (1)	9.71×10^4	0.310	3.01×10^4	3.01×10^4
Orificed Fuel Support (193)	1.01×10^3	0.0036	3.63×10^0	7.01×10^2
Core Support Plate (1)	2.56×10^2	2.54	6.50×10^2	6.50×10^2
Incore Instrument Strings (55)	7.67×10^5	0.00026	1.99×10^2	1.10×10^4
Control Rod (185)	5.11×10^5	0.0019	9.61×10^2	1.78×10^5
Control Rod Guide Tube (185)	2.16×10^2	0.0024	5.12×10^{-1}	9.47×10^1
Total				6.55×10^6

shield is calculated to be about 166 Ci. The total inventory in all neutron-activated components is about 6.6 million Ci. The activated portion of the core shroud represents approximately 96% of this total inventory.

E.1.3 Radiation Dose Rates from Neutron-Activated Components

Two steps are necessary to compute the radiation dose rate at the surface of an activated structure after the level of radioactivity has been determined. First, the radiation emissions from the decaying radionuclides are characterized in terms of energy and probability of decay. Then, the transport of the radiation through the bulk structural material and the resultant dose rate exterior to the material surface are computed. These steps are discussed in more detail in the following subsections.

E.1.3.1 Radioactive Decay Characteristics of Selected Radionuclides

Many of the radionuclides present in the neutron-activated structures at the shutdown BWR have high specific radioactivity levels but, because of their particular decay half-lives or decay processes or both, do not contribute significantly to the radiation dose rate during decommissioning. Other radionuclides have lower specific radioactivity levels but produce large contributions to an external dose rate because they emit high-energy gamma rays.

The following paragraphs present the characteristics (i.e., production process, radioactive decay mode, and radioactive emissions) of selected radionuclides that represent some concern during decommissioning.

Carbon-14 ($^{14}_6\text{C}$). Carbon-14 is produced by two processes: 1) by multiple neutron capture in $^{12}_6\text{C}$ and $^{13}_6\text{C}$, and 2) by neutron capture-proton emission in $^{14}_7\text{N}$. The second reaction is the predominant one. Both carbon and nitrogen are present in trace quantities in stainless and carbon steels. Carbon-14 has a radioactive decay half-life of about 5,750 years and decays by beta emission with a maximum energy of 156 keV. The contribution of $^{14}_6\text{C}$ 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 environment where it could enter biological internal-exposure pathways.

Iron-55 ($^{55}_{26}\text{Fe}$). Iron-55 is produced principally by thermal neutron capture in 5.8%-abundant ^{54}Fe in iron, a constituent in stainless steel and carbon steel. The neutron capture cross section is about 3 barns.^(a) Iron-55 has a decay half-life of 2.7 years and decays by orbital electron capture (EC) by the nucleus to form $^{55}_{25}\text{Mn}$.

Each EC event is accompanied by the emission of either x-rays or Auger electrons associated with ^{55}Mn stabilization to the ground energy state, with energies less than 7 keV. As is typical with such emissions from nuclides in this atomic number range, most (~74% in this case) of the x-rays are not emitted, and Auger electrons of the same energy appear instead. Additional information on these emissions can be found in References 8 and 9. Suffice it to say that these emissions do not contain the energy (i.e., >68 keV) required to penetrate the outer dead layers of skin and, thus, do not contribute to a skin dose.

Each EC decay event is accompanied by a continuous spectrum of gamma rays from the associated inner bremsstrahlung (IB) process. The maximum energy of this spectrum is 220 keV. For computing radiation dose in this study, the IB spectrum of ^{55}Fe decay is divided into four energy ranges, with a specific yield fraction per decay event for each energy range as follows:

- 0.01 to 0.07 MeV, 1.37×10^{-5}
- 0.07 to 0.12 MeV, 8.37×10^{-6}
- 0.12 to 0.21 MeV, 4.77×10^{-6}
- 0.21 to 0.04 MeV, 8.5×10^{-9} .

Further details concerning the IB gamma spectrum from ^{55}Fe decay are given in References 10 and 11.

Cobalt-60 ($^{60}_{27}\text{Co}$). Cobalt-60 is produced by thermal neutron capture in 100%-abundant ^{59}Co , a trace constituent in all three structural materials. The neutron capture cross section is about 37 barns. Cobalt-60 has a decay half-life of 5.27 years and decays by beta emission to excited energy states of $^{60}_{28}\text{Ni}$. Over 99% of the decay events are to an excited energy state at

(a) 1 barn is equivalent to a cross-sectional area of 10^{-22} mm^2 .

2.505 MeV, which goes to the ground energy state, with a sequential emission of two gamma rays of 1.173 MeV and 1.332 MeV.⁽⁹⁾ These are the major contributors to the dose rate from ^{60}Co . The beta spectrum (maximum-energy 0.32 MeV) contributes a minor fraction.

Nickel-59 ($^{59}_{28}\text{Ni}$). Nickel-59 is primarily produced by thermal neutron capture in 68.3%-abundant ^{58}Ni in nickel, a constituent in stainless steel. The neutron capture cross section is about 5 barns. Nickel-59 has a decay half-life of approximately 80,000 years and decays by the EC process to $^{59}_{27}\text{Co}$.

Again, x-rays or Auger electrons are emitted during stabilization to the ground energy state. In the case of ^{59}Co , the energies are less than 8 keV, and approximately 33% of the emissions are x-rays and the rest are Auger electrons. Additional information is found in References 8 and 9. These emissions do not contribute to the external radiation dose rate.

The contribution of the ^{59}Ni decay process to external dose rate comes from the EC-associated IB process. A continuous spectrum of gamma rays is emitted, with the maximum energy being 1.07 MeV. For computing radiation dose in this study, the IB spectrum of ^{59}Ni decay is divided into seven energy ranges, with a specific yield fraction per decay event for each energy range, as follows:

- 0.01 to 0.07 MeV, 2.23×10^{-4}
- 0.07 to 0.12 MeV, 6.25×10^{-5}
- 0.12 to 0.21 MeV, 1.32×10^{-4}
- 0.21 to 0.4 MeV, 3.21×10^{-4}
- 0.4 to 0.6 MeV, 3.31×10^{-4}
- 0.6 to 0.9 MeV, 3.24×10^{-4}
- 0.9 to 1.2 MeV, 3.01×10^{-5} .

More details concerning the IB gamma spectrum from ^{59}Ni decay can be found in References 11 through 15.

Nickel-63 ($^{63}_{28}\text{Ni}$). Nickel-63 is produced by thermal neutron capture in 3.6%-abundant ^{62}Ni in nickel, a constituent in stainless steel. The neutron capture cross section is about 15 barns. Nickel-63 has a decay half-life of

about 100 years and decays by beta emission to $^{63}_{29}\text{Cu}$. The beta emissions have maximum energies of about 66 keV, which is insufficient to penetrate the outer surface of the skin. No gamma rays are emitted. Therefore, ^{63}Ni does not contribute to the external dose rate.⁽¹⁶⁾ Additional information is available in Reference 17.

Molybdenum-93 ($^{93}_{42}\text{Mo}$). Molybdenum-93 is formed by thermal and resonance neutron capture in 14.8%-abundant ^{92}Mo in molybdenum, a minor constituent in both stainless steel and carbon steel. The thermal neutron capture cross section is about 0.045 barns, and the resonance neutron capture integral is about 0.52 barns. Molybdenum-93 can be formed in an excited energy state at 2.45 MeV, which decays by gamma emission to the ground state with a 6.85-hour half-life. The ground state of ^{93}Mo decays by EC, with a half-life of about 3,500 years. Approximately 96% of the EC events proceed to an excited state of $^{93}_{41}\text{Nb}$ with an excitation energy of 30.4 keV. These EC events are accompanied by an IB gamma spectrum with a maximum energy of 368 keV. The yield and spectral shape of the IB gamma emissions are not amenable to definitive theoretical calculation and have not been measured experimentally. The IB yield is on the order of 3.2×10^{-4} gammas per EC event.

The 30.4-keV state in ^{93}Nb decays with a half-life of 13.6 years. Unless chemically separated from ^{93}Mo , ^{95}Nb has an apparent decay half-life similar to ^{93}Mo at long times ($\gg 13.6$ yr) after neutron irradiation. The 30.4-keV decay gamma rays are internally absorbed, and none have been observed experimentally. The radiation products per decay of the 30.4-keV state are primarily 10.4-keV Auger electrons and 16.6-keV niobium x-rays.

Niobium-94 ($^{94}_{41}\text{Nb}$). Niobium-94 is formed by thermal neutron capture in 100%-abundant ^{93}Nb , a trace constituent in stainless steel. The thermal neutron capture cross section is about 1.1 barns. Neutron capture leads either to the ground state of ^{94}Nb or to an isomeric state that is 41 keV above the ground state. The isomeric state decays with a half-life of 6.26 minutes. Most (99.5%) of these decays go to the ground energy state. Thus, for long-time radiation dose rate considerations, all neutron captures lead to the ground energy state. Niobium-94 has a decay half-life of 20,300 years and

decays by beta emission, with a maximum beta energy of 471 keV. All of the decays go to a single excited energy state of ^{94}Mo at 1.574 MeV excitation. All of the decays from this state go through an excited energy state at 871 keV. Thus, each ^{94}Nb decay yields two gamma rays with energies of 703 and 871 keV, respectively; these dominate the dose rate from ^{94}Nb .

Silver-108m ($^{108\text{m}}_{47}\text{Ag}$). Silver-108m is formed by thermal and resonance neutron capture in 51.8%-abundant ^{107}Ag in silver, a trace constituent in both stainless steel and concrete. The thermal neutron capture cross section is about 2 barns, and the resonance neutron capture integral is about 5 barns. Silver-108m has a decay half-life of 127 years and decays both by EC and by isomeric transition (IT).

The majority (~91%) of the decays are by EC to an excited energy state of $^{108}_{48}\text{Pd}$ that results in three cascade gamma rays with energies of 723 keV, 614 keV, and 434 keV, respectively. In addition, x-rays and conversion electrons with energies less than 25 keV are produced by this decay process.

The rest (~9%) of the decays are by cascade IT gamma rays of 30.4 keV and 79.2 keV to the ground energy state of ^{108}Ag , which is also radioactive. The 30.4-keV gamma ray is absorbed internally, and the 79.2 keV gamma ray is emitted externally only in about 7% of the total $^{108\text{m}}\text{Ag}$ decay events.

Silver-108 has a decay half-life of 2.37 minutes and decays by beta emission, primarily to the ground energy state of $^{108}_{48}\text{Cd}$. However, about 2% of the decays go to the first excited energy state of ^{108}Cd , which emits a 632-keV gamma ray in the stabilization process. The average beta energy per ^{108}Ag decay is 610 keV.

The cascading gamma rays from the EC decay process and the beta and gamma emissions from ^{108}Ag dominate the dose rate from $^{108\text{m}}\text{Ag}$ decay.

Europium-152 ($^{152}_{63}\text{Eu}$). Europium-152 is formed by thermal and resonance neutron capture in 47.9%-abundant ^{151}Eu in europium, a trace constituent in both stainless steel and concrete. The thermal neutron capture cross section is 5,900 barns, and the resonance neutron capture integral is about 2,100 barns. Europium-152 has a decay half-life of 13.3 years. It decays

about 72% of the time by EC or positron emission (0.05% of the time) to many excited energy states of $^{152}_{62}\text{Sm}$, and the rest of the time by beta emission to several excited energy states of $^{152}_{64}\text{Gd}$. The result of each of these decays is the emission of cascade gamma rays in the stabilization process of each of the resultant nuclides. Inner bremsstrahlung gamma rays are negligible compared to the cascade gamma rays, but do occur since some EC events go to the ground energy state of ^{152}Sm .

The decay process results in conversion electrons with a maximum energy of 294 keV and in x-rays with energies up to 50 keV. The beta emissions have an average energy per decay of 125 keV and a maximum energy of 1.475 MeV. The resultant cascade gamma rays, of which a total of 125 discrete energies are listed in the files of Reference 5, have an average energy per decay of 1.16 MeV and a maximum energy of 1.41 MeV.

Europium-154 ($^{154}_{63}\text{Eu}$). Europium-154 is formed by thermal and resonance neutron capture in 52.1%-abundant ^{153}Eu in europium, a trace constituent in both stainless steel and concrete. The thermal neutron capture cross section is 310 barns, and the resonance neutron capture integral is 1,610 barns. Europium-154 has a decay half-life of 8.6 years and decays by beta emission to excited energy states of $^{154}_{64}\text{Gd}$. Again, many cascade gamma rays are produced in the stabilization process.

The process results in conversion electrons having a maximum energy of 198 keV and in x-rays with energies up to 50 keV. The maximum beta energy is 1.85 MeV, and the average energy per decay is 280 keV. The resultant cascade gamma rays, of which 139 are listed in the files of Reference 5, have a maximum energy of 1.90 MeV and an average energy per decay of 1.23 MeV.

E.1.3.2 Radiation Dose Rates

The radiation dose rates at 10 mm away from the inner and outer surfaces of the core shroud, reactor vessel, and sacrificial shield are calculated using the ANISN and PUSHLD codes.^(7,18) ANISN calculates the radiative transport and the resultant dose rate from the monoenergetic gamma rays and the inner bremsstrahlung gamma spectra, and PUSHLD calculates the transport dose rate of the low-energy x-rays. The results for a particular radioisotope are summed to obtain its total contribution to the dose rate.

To check the effect of code differences on the result, PUSHLD is also used to make calculations for comparison with the ANISN results.

The computed dose rates at the time of final reactor shutdown from the radionuclides discussed in Section E.1.3.1 are given in Table E.1-7. Only those radionuclides that contribute significantly to the dose rate, either in the near term or after a long decay time, are included.

TABLE E.1-7. Calculated Radiation Dose Rates from Selected Neutron-Activated Reference BWR Components(a)

Component	Calculated Dose Rate from Selected Radionuclides (R/hr)						
	⁶⁰ Co (gamma)	⁹⁴ Nb (gamma)	⁵⁵ Fe (IB, gamma)(b)	⁵⁹ Ni (IB, gamma)(b)	¹⁰⁸ Ag (gamma)	¹⁵² Eu (gamma)	¹⁵⁴ Eu (gamma)
Core Shroud							
Inner Surface	1.2 x 10 ⁵	7 x 10 ⁻¹	8 x 10 ⁻²	7 x 10 ⁻²	2 x 10 ⁻²	2 x 10 ⁻⁴	5 x 10 ⁻¹
Outer Surface	3.3 x 10 ⁴	2 x 10 ⁻¹	1 x 10 ⁻²	2 x 10 ⁻²	---(c)	---	---
Reactor Vessel							
Inner Surface	1.4 x 10 ¹	1 x 10 ⁻⁴	3 x 10 ⁻⁵	2 x 10 ⁻⁵	---	---	---
Outer Surface	1.4 x 10 ⁻¹	4 x 10 ⁻⁹	1 x 10 ⁻⁶	7 x 10 ⁻⁸	---	---	---
Sacrificial Shield							
Inner Surface	3.2 x 10 ⁻¹	---	3 x 10 ⁻⁶	2 x 10 ⁻⁷	8 x 10 ⁻⁸	7 x 10 ⁻³	7 x 10 ⁻⁵
Outer Surface	9 x 10 ⁻⁴	---	1 x 10 ⁻⁸	1 x 10 ⁻⁹	---	---	---

(a) Calculated at a distance of 10 mm from the surface of the activated component, at the axial midplane of the fuel zone, and at final reactor shutdown.

(b) IB means "inner bremsstrahlung."

(c) Indicates the quantity was not calculated.

A comparison of the time-dependent radioactivity concentrations and dose rates of some selected radionuclides is shown in Figure E.1-6. The data are those calculated for the core shroud. It can be seen that, while the decay rate of ⁶³Ni controls the reduction of the total radioactivity present after the first 10 years, the decay rate of ⁶⁰Co controls the reduction of the radiation dose rate for the first 70 to 80 years. After that time, the dose rate is increasingly dominated by ⁹⁴Nb. The 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 E.1-6. Time Dependence of Radioactivity and Dose Rate in the Neutron-Activated Core Shroud

E.2 SURFACE CONTAMINATION

Activated corrosion products from structural materials in contact with the reactor water and fission products from leaking fuel both contribute to the radionuclide mixtures and levels of surface contamination. This section contains the methods and results of calculations that estimate the radionuclide inventories in internally contaminated piping and equipment systems, and also on external surfaces both inside the reference BWR facility and on the surrounding site.

E.2.1 Radionuclide Inventories in Internally Contaminated Piping and Equipment

A thin layer of radioactive contamination is deposited on the internal surfaces of piping and equipment in the reference BWR during its normal operating lifetime. The piping and equipment systems involved are described in Appendix C. The composition and amount of radioactivity found on internal surfaces at plant shutdown are dependent on such reactor parameters as: 1) structural material composition, 2) reactor size, design, and operating history, and 3) reactor fuel conditions.⁽¹⁹⁾ The internal surface contamination is characterized by the mixture of activated corrosion products and fission products found in the reactor water, steam, and resultant condensate.

In the following sections, the reference radionuclide inventory for piping and internal surfaces is established, and the dose rate versus deposition level relationships for various designs of piping and equipment are determined using a shielding code analysis and a unit deposition level. Inventories of total deposited radioactivity in the reference BWR are calculated using the dose rate/deposition relationships and measured dose rate data from Appendix D.

E.2.1.1 Reference Radionuclide Inventory 4

Specific alloys used in the structural components of the reactor coolant system play a major role in the composition of the internal surface contamination. Because of the different alloys used, differences in the activated corrosion products between a BWR and a PWR are noted.⁽²⁰⁾ Cobalt-58 is only a minor source of radiation at a BWR, while at a PWR it is a significant contributor to the shutdown radiation levels.⁽²¹⁾ This is because more ^{58}Ni , present in materials such as Inconel-600, is found in the PWR. Through neutron capture, the ^{58}Ni is transmuted into ^{58}Co . Cobalt-60 is found in greater quantity at a BWR than at a PWR because of the greater abundance of its parent (^{59}Co) in BWR structural materials. It is of concern because of its large formation cross section, its energetic decay, and its relatively long decay half-life.⁽²¹⁾ Depending on the type of condenser tubes and condensate polishing system used, ^{65}Zn can also be an isotope of concern at a BWR.

Fission products from leaking reactor fuel also contribute to the internal surface contamination. Their concentrations are directly related to

the number of leaking fuel elements in the reactor core and thus will change during plant operation.

It is not within the scope of this study to fully investigate the complex mechanisms that influence the deposition of activated corrosion products and fission products on the internal surfaces of BWR equipment and piping. After a literature review on the subject (References 19 and 21 through 25), the radionuclide composition on internal surfaces of the reference BWR at final shutdown is assumed to be that shown in Table E.2-1 (reference radionuclide inventory 4). This inventory is based on a BWR sludge sample analysis given in Reference 23, and contains representative levels of both activated corrosion products and fission products present in the reactor water systems. Reference radionuclide inventory 4 also compares closely with the mixture of radionuclides in the turbine of an operating BWR.⁽²⁵⁾ No further attempt to characterize the composition of internal surface contamination is made.

TABLE E.2-1. Reference Radionuclide Inventory 4,
BWR Internal Surface Contamination^(a)

Radionuclide	Fractional Radioactivity at Decay Times of:				
	Shutdown	10 Years	30 Years	50 Years	100 Years
⁵¹ Cr	2.1×10^{-2}	--- ^(b)	---	---	---
⁵⁴ Mn	3.9×10^{-1}	1.3×10^{-4}	---	---	---
⁵⁹ Fe	2.5×10^{-2}	---	---	---	---
⁵⁸ Co	9.3×10^{-3}	---	---	---	---
⁶⁰ Co	4.7×10^{-1}	1.3×10^{-1}	9.1×10^{-3}	6.6×10^{-4}	9.2×10^{-7}
⁶⁵ Zn	6.1×10^{-3}	1.5×10^{-7}	---	---	---
⁹⁵ Zr	4.0×10^{-3}	---	---	---	---
⁹⁵ Nb	4.0×10^{-3}	---	---	---	---
¹⁰³ Ru	2.3×10^{-3}	---	---	---	---
¹⁰⁶ Ru	2.8×10^{-3}	2.7×10^{-6}	---	---	---
¹³⁴ Cs	1.9×10^{-2}	---	---	---	---
¹³⁷ Cs	3.4×10^{-2}	2.7×10^{-2}	1.7×10^{-2}	1.1×10^{-2}	3.4×10^{-3}
¹⁴¹ Ce	3.0×10^{-3}	---	---	---	---
¹⁴⁴ Ce	8.1×10^{-3}	8.8×10^{-7}	---	---	---
Totals	1.0	1.6×10^{-1}	2.6×10^{-2}	1.2×10^{-2}	3.4×10^{-3}

(a)Based on a BWR sludge sample analysis given in Reference 23.

(b)Indicates a value less than 1.0×10^{-10} .

About 93% of the radioactivity mixture at shutdown, shown in Table E.2-1, is from activated corrosion products, with 47% of this total being ^{60}Co . Radionuclide characterization data for these activated corrosion products are listed in Table E.2-2. Fission products contribute only about 7% of the total internal surface radioactivity at shutdown.

TABLE E.2-2. Activated Corrosion Product Data

Radionuclide	Fractional Abundance on Surfaces at Final Shutdown	Primary Source Material	Production Reaction	Half-Life (days)	Gamma Emissions	
					Energy (MeV)	Number per Disintegration
^{51}Cr	2.1×10^{-2}	Stainless Steel	$^{50}\text{Cr} (n,\gamma)$	2.8×10^1	3.2×10^{-1}	2.0×10^{-2}
^{54}Mn	3.9×10^{-1}	Carbon Steel	$^{54}\text{Fe} (n,p)$	3.1×10^2	8.4×10^{-1}	1.0×10^0
^{59}Fe	2.5×10^{-2}	Carbon Steel	$^{58}\text{Fe} (n,\gamma)$	4.5×10^1	1.4×10^{-1}	1.0×10^{-2}
					1.9×10^{-1}	3.0×10^{-2}
					1.1×10^0	5.6×10^{-1}
					1.3×10^0	4.4×10^{-1}
^{58}Co	9.3×10^{-3}	Stainless Steel	$^{58}\text{Ni} (n,p)$	7.2×10^1	5.1×10^{-1}	1.0×10^{-2}
					8.1×10^{-1}	9.9×10^{-1}
					8.6×10^{-1}	1.4×10^{-2}
					1.7×10^0	6.2×10^{-3}
^{60}Co	4.7×10^{-1}	Carbon Steel	$^{59}\text{Co} (n,\gamma)$	1.9×10^3	1.2×10^0	1.0×10^0
					1.3×10^0	1.0×10^0
^{65}Zn	6.1×10^{-3}	Admiralty Brass	$^{64}\text{Zn} (n,\gamma)$	2.4×10^2	1.1×10^0	4.9×10^{-1}
<u>Fission Products</u>	<u>6.9×10^{-2}</u>	Leaking Fuel				
Total	1.0					

Reduction of the dose rate with time because of radioactive decay of the mixture is shown in Figure E.2-1. The total dose rate is normalized to unity at the time of plant shutdown. The fractional dose rates are based on the shutdown radionuclide mixture in radionuclide inventory 4. The activated corrosion product ^{60}Co controls the dose rate of the mixture until about 50 years after final shutdown, at which time the fission product ^{137}Cs begins to dominate.

FIGURE E.2-1. Radioactive Decay of Deposited Internal Surface Contamination

E.2.1.2 Dose Rates from Internally Contaminated Piping

Dose rates from internally deposited radionuclide contamination in the reference BWR piping systems are estimated using the ISOSHL D computer program. (26,27) This program is used with a unit deposition level of radionuclide inventory 4 to determine the relationship between the deposition level and the contact dose rate on the external pipe surfaces. ISOSHL D uses a point-kernel integration technique to evaluate exposure rates with a given source/detector geometry. Photon energy flux is calculated for 25 energy groups, with average energies ranging from 15 keV to 3.0 MeV. Twelve source/detector geometries are available. Piping is modeled using cylindrical geometry.

Typical reference BWR pipe sizes and characteristic data are listed in Table E.2-3. The resulting calculated contact dose rates on external pipe surfaces as a function of the internal surface deposition levels are shown in

TABLE E.2-3. Typical Reference-BWR Piping Characteristics

Nominal Outside Diameter (mm)	Approximate Schedule	Nominal Inside Diameter (mm)	Nominal Wall Thickness (mm)
60	40	52.5	3.9
168	60	149	9.5
356	60	325	15.1
559	40	526	16.3
610	120	518	46.0
610	160	491	59.5
711	40	670	20.4
914	20	889	12.7

Figure E.2-2. For these calculations, the modeled pipes are assumed to be filled with water, so that a comparison with dose rates measured at operating BWR power stations can be made.

FIGURE E.2-2. Calculated Contact Dose Rates for BWR Piping as a Function of Deposited Internal Surface Radioactivity

When the pipes are drained during decommissioning, the contact dose rates increase, since the water in the pipes is no longer a shield. For a large (914-mm outside diameter) thin-walled (schedule 20) pipe, the increase in dose rate caused by draining the pipe is calculated using the ISOSHL D program for a unit internal surface deposition. The results, shown in Figure E.2-3, indicate a rapid divergence between the calculated air-filled and water-filled contact dose rates as a function of increasing internal surface deposition. An increase in dose rate by a factor of about 3 for this size of air-filled pipe is found for the range of measured dose rates presented in Appendix D. Since smaller pipes act like line sources, the increase in dose rate caused by draining small pipes is much less.

The wall thickness of a pipe is a controlling factor in the dose rate versus deposition relationship for a given pipe outside diameter. Thus, the wall thickness as well as the pipe size is important in estimating the dose rate from internally contaminated pipe. A comparison of the calculated dose rate versus deposition relationships for schedule 120 and schedule 160 pipe with a 610-mm outside diameter is shown in Figure E.2-4.

FIGURE E.2-3. Calculated Contact Dose Rates for Air-Filled and Water-Filled Schedule 20, 914-mm-OD BWR Pipe as a Function of Deposited Internal Surface Radioactivity

FIGURE E.2-4. Calculated Contact Dose Rates from Schedule 120 and Schedule 160, 610-mm-OD BWR Pipe as a Function of Deposited Internal Surface Radioactivity

The calculated relationship between dose rate, pipe size, and various distances in air from pipe surfaces is shown in Figure E.2-5. This figure graphically indicates that the dose rate from a small-diameter pipe quickly decreases as the distance in air increases and that the dose rate from a large-diameter pipe is less affected by increased distance in air. Thus, dose rates from small-diameter pipes behave as those from line sources, and dose rates from large-diameter pipes behave as those from finite slabs.

E.2.1.3 Estimated Total Radioactivity in Internally Contaminated Piping

Internal surface contamination levels in BWR piping are dependent on the radionuclide concentrations in the reactor water, steam, and condensate. The deposited amounts of reference radionuclide inventory 4 inside piping at the reference BWR at plant shutdown are calculated using the following assumptions:

FIGURE E.2-5. Relative Dose Rate as a Function of Distance in Air from Pipe Outer Surface

- Piping is divided into three categories: reactor water, steam, and condensate. Uniform deposition in all piping of each category is assumed.
- The deposition levels for reactor water and condensate are found using representative measured dose rates in Appendix D and the calculated dose rate/deposition relationships for a specific pipe size and schedule shown in Figure E.2-2.
- Since steam is less dense than liquid water, deposition levels for steam piping are calculated using the dose rate/deposition relationship for a large (914-mm-OD) pipe filled with air, shown in Figure E.2-3, and measured dose rate data from a large steam-filled pipe.
- Average pipe outside diameters are used in calculating the internal surface area of each range of pipe sizes listed in Section C.3.4 of Appendix C.

A summary of the measured dose rate data and resultant deposition levels used to characterize reactor water, steam, and condensate piping systems, is shown in Table E.2-4.

TABLE E.2-4. Piping Contact Dose Rate/Internal Radioactivity Deposition Summary

<u>Pipe Category</u>	<u>Nominal Outside Diameter (mm)</u>	<u>Wall Thickness (mm)</u>	<u>Measured Contact Dose Rate (mR/hr)(a)</u>	<u>Estimated Deposition Level (Ci/m²)(b)</u>
Reactor Water	610	59.5	700	1.1
Steam/Air	914	20.4	70	0.005
Condensate	610	46.0	50	0.05

(a) From measured data presented and discussed in Appendix D.

(b) Estimated using the relationships shown in Figures E.2-2 through E.2-4.

The estimated radioactivity depositions in BWR piping are calculated using the deposition levels in Table E.2-4 and pipe information from Appendix C. The resulting values are shown in Table E.2-5. For this study, a total of 2,200 Ci of radionuclide inventory 4 is estimated to be present on internal pipe surfaces at shutdown.

E.2.1.4 Dose Rates from Internally Contaminated Equipment

Dose rates from radioactive contamination deposited on internal surfaces of BWR equipment in contact with the reactor water are determined by modeling a nonregenerative heat exchanger in the reactor water cleanup system. This equipment is selected since it is representative of BWR equipment containing reactor water. It is about 8.5 m long and contains 92 U tubes.

The surface dose rates are calculated for a unit deposition on a model of the nonregenerative heat exchanger (shown in Figure E.2-6) using the ISOSHL D computer code. The resulting surface dose rate/internal deposition relationship is illustrated in Figure E.2-7.

E.2.1.5 Estimated Total Radioactivity in Internally Contaminated Equipment

Internal surface contamination levels in BWR equipment are estimated using the following assumptions:

TABLE E.2-5. Estimated Contaminated Piping Internal Radioactivity Deposition^(a)

Material Pipe Category	60 mm Average O.D.			152 mm Average O.D.			356 mm Average O.D.		
	Length (m)	Surface Area (m ²)	Deposited Radio- activity (Ci)	Length (m)	Surface Area (m ²)	Deposited Radio- activity (Ci)	Length (m)	Surface Area (m ²)	Deposited Radio- activity (Ci)
Aluminum Steam/Air Condensate	4.3 x 10 ² ---(b)	8.1 x 10 ¹ ---	4.0 x 10 ⁻¹ ---	1.4 x 10 ³ 1.4 x 10 ¹	6.4 x 10 ² 6.7 x 10 ⁰	3.2 x 10 ⁰ 3.4 x 10 ⁻¹	1.3 x 10 ² ---	1.4 x 10 ² ---	7.0 x 10 ⁻¹ ---
Carbon Steel Reactor Water Steam/Air Condensate	3.8 x 10 ² 1.2 x 10 ³ 7.4 x 10 ³	7.1 x 10 ¹ 2.2 x 10 ² 1.4 x 10 ³	7.8 x 10 ¹ 1.1 x 10 ⁰ 7.0 x 10 ¹	1.5 x 10 ³ 1.8 x 10 ³ 8.3 x 10 ³	7.0 x 10 ² 8.8 x 10 ² 3.9 x 10 ³	7.7 x 10 ² 4.4 x 10 ⁰ 2.0 x 10 ²	6.1 x 10 ¹ 5.6 x 10 ³ 5.1 x 10 ³	6.8 x 10 ¹ 6.3 x 10 ³ 5.7 x 10 ³	7.5 x 10 ¹ 3.2 x 10 ¹ 2.8 x 10 ²
Stainless Steel Reactor Water Steam/Air Condensate	8.0 x 10 ⁰ 2.8 x 10 ² 7.0 x 10 ³	1.5 x 10 ⁰ 5.3 x 10 ¹ 1.3 x 10 ³	1.6 x 10 ⁰ 2.7 x 10 ⁻¹ 6.6 x 10 ¹	3.4 x 10 ¹ --- 1.6 x 10 ³	1.6 x 10 ¹ --- 7.8 x 10 ²	1.8 x 10 ¹ --- 3.9 x 10 ¹	6.1 x 10 ¹ --- 2.2 x 10 ²	6.8 x 10 ¹ --- 2.4 x 10 ²	7.5 x 10 ¹ --- 1.2 x 10 ¹
Totals			2.2 x 10 ²			1.0 x 10 ³			4.7 x 10 ²

Material Pipe Category	533 mm Average O.D.			660 mm Average O.D.			914 mm Average O.D.		
	Length (m)	Surface Area (m ²)	Deposited Radio- activity (Ci)	Length (m)	Surface Area (m ²)	Deposited Radio- activity (Ci)	Length (m)	Surface Area (m ²)	Deposited Radio- activity (Ci)
Aluminum Steam/Air	---	---	---	---	---	---	---	---	---
Carbon Steel Reactor Water Steam/Air Condensate	5.5 x 10 ¹ 1.2 x 10 ³ 2.8 x 10 ³	9.2 x 10 ¹ 2.0 x 10 ³ 4.6 x 10 ³	1.0 x 10 ² 1.0 x 10 ¹ 2.3 x 10 ²	9.5 x 10 ² 3.7 x 10 ²	2.0 x 10 ³ 7.7 x 10 ²	9.8 x 10 ⁰ 3.8 x 10 ¹	4.4 x 10 ² 2.1 x 10 ²	1.3 x 10 ³ 6.1 x 10 ²	6.3 x 10 ⁰ 3.1 x 10 ¹
Stainless Steel Reactor Water Steam/Air Condensate	5.5 x 10 ¹ --- ---	9.2 x 10 ¹ --- ---	1.0 x 10 ² --- ---	---	---	---	---	---	---
Totals			4.4 x 10 ²			4.8 x 10 ¹			3.7 x 10 ¹

Estimated total deposited radioactivity on all contaminated piping internal surfaces = 2.2×10^3 Ci.

(a)Based on piping data in Tables C.3-7 through C.3-10, and internal deposition levels given in Table E.2-4

(b)Indicates no piping in the particular category and size.

FIGURE E.2-6. Cross Section of the Cylindrical Model of a Nonregenerative Heat Exchanger

FIGURE E.2-7. Calculated Contact Dose Rate from a Nonregenerative Heat Exchanger as a Function of Deposited Internal Surface Radioactivity

1. The dose rate for the nonregenerative heat exchanger (reported in Appendix D) is 10 R/hr at contact. From Figure E.2-7, a deposition of 0.36 Ci/m^2 is estimated. This value is used for all equipment surfaces in contact with reactor water.
2. In general, equipment in contact with steam or condensate is assumed to reach the same deposition levels calculated previously for BWR piping, with some exceptions. Steam surfaces in the turbine and heat transfer surfaces in the main condenser and the feedwater heaters are assumed to be a factor of 10 cleaner than the corresponding piping to reflect the low dose rate levels reported in Appendix D.
3. Valves and pumps are assumed to be modeled by equivalent lengths of similar diameter pipe, and are included in the total for pipe.
4. Tanks in the Radwaste and Control Building that contain concentrated waste slurries are assumed to have a deposition level 100 times higher than condensate piping.

A summary of the deposition levels calculated for or assumed in this study is shown in Table E.2-6.

TABLE E.2-6. Summary of Equipment Internal Radioactivity Deposition Levels

<u>Equipment Category</u>	<u>Radioactivity Deposition Level (Ci/m²)</u>
Reactor Water Equipment	3.6×10^{-1}
Steam Equipment	5.0×10^{-3}
Turbine	5.0×10^{-4}
Condensate Equipment	5.0×10^{-2}
Main Condenser	5.0×10^{-3}
Feedwater Heaters	5.0×10^{-3}
Concentrated Waste Tanks/Equipment	5.0×10^0

The estimated radioactivity depositions in BWR equipment are calculated using the deposition levels in Table E.2-6 and the general assumptions previously listed. The results are shown in Table E.2-7. For this study, a total of 6,300 Ci of reference radionuclide inventory 4 is estimated to be present on internal surfaces of equipment at shutdown.

E.2.2 Radionuclide Inventories on Externally Contaminated Structural Surfaces

The radionuclide mixture found on externally contaminated structural surfaces in the reference BWR is assumed to reflect the mixture of radionuclides found in the reactor water. Leaks occurring in normally accessible areas are assumed to be repaired and cleaned up according to standard operating procedures. Leaks that occur in areas not normally accessible are assumed to accumulate and build up over the 40-year plant operating lifetime. No data relating to general surface contamination mixtures or levels are found in the literature. Therefore, for this study, the radionuclides contained in a unit leak of reactor water are assumed to accumulate on a structural surface to define the resultant radionuclide deposition. The dose rate/deposition relationship for the resulting radionuclide inventory is determined, and estimates of the total surface contamination are made.

TABLE E.2-7. Estimated Contaminated BWR Equipment Internal Radioactivity Deposition^(a)

Building Equipment Key Number ^(b)	Equipment Name	Total Estimated Internal Surface Area (m ²)	Radioactivity Deposition Level (Ci/m ²)	Deposited Radio- activity (Ci)
Reactor Building				
56	Fuel Pool Heat Exchangers	8.0×10^2	5.0×10^{-2}	4.0×10^1
64	Skimmer Surge Tanks	1.0×10^2	5.0×10^{-2}	5.0×10^1
90, 63, 112	Fuel Pool, Reactor Well, Dryer and Separator Pool	1.4×10^3	5.0×10^{-2}	7.0×10^1
51	RBCC Water Heat Exchangers	1.8×10^3	5.0×10^{-2}	9.0×10^1
61	RWCU Regenerative Heat Exchangers	2.5×10^2	3.6×10^{-1}	9.0×10^1
62	RWCU Nonregenerative Heat Exchangers	1.7×10^2	3.6×10^{-1}	6.0×10^1
54	RHR Heat Exchangers	1.5×10^3	3.6×10^{-1}	5.4×10^2
1	Reactor Vessel	2.6×10^3 (c)	3.6×10^{-1}	9.4×10^2
Total				1.9×10^3
Turbine Generator Building				
200	Main Condenser	7.9×10^4	5.0×10^{-3}	3.9×10^2
216	Steam Jet Air Ejector Condenser	1.6×10^3	5.0×10^{-2}	8.0×10^1
217	Gland Seal Steam Condenser	3.5×10^2	5.0×10^{-2}	1.7×10^1
228	Condensate Storage Tanks	1.6×10^3	5.0×10^{-2}	8.0×10^1
253	Low-Pressure Feedwater Heaters	7.5×10^4	5.0×10^{-3}	3.7×10^2
255	Evaporator Drain Tanks	1.0×10^1	5.0×10^{-2}	5.0×10^{-1}
256	Reheater Drain Tanks	8.4×10^2	5.0×10^{-2}	4.2×10^1
263	Moisture Separator Drain Tank	3.0×10^1	5.0×10^{-3}	1.5×10^{-1}
201	Main Turbine	2.6×10^3 (c)	5.0×10^{-4}	1.3×10^0
254	Steam Evaporator	2.0×10^3	5.0×10^{-3}	1.0×10^1
258	Turbine Bypass Valve Assembly	1.5×10^1	5.0×10^{-3}	7.5×10^{-1}
270	Moisture Separator Reheaters	1.8×10^4	5.0×10^{-3}	9.0×10^1
226	Seal Water Liquid Tank	1.2×10^1	5.0×10^{-2}	6.0×10^{-1}
227	Pumped Drain Tank	2.7×10^1	5.0×10^{-2}	1.4×10^0
252	High-Pressure Feedwater Heaters	1.7×10^4	5.0×10^{-3}	8.5×10^1
Total				1.2×10^3
Radwaste and Control Building				
302	Condensate Phase Separator Tanks	1.8×10^2	5.0×10^0	9.0×10^2
304	Condensate Backwash Receiver Tank	8.5×10^1	5.0×10^0	4.2×10^2
301	Waste Collector Tank	1.0×10^2	5.0×10^{-2}	5.0×10^0
312	Waste Surge Tank	1.9×10^2	5.0×10^0	9.5×10^2
313	Waste Sample Tanks	1.6×10^2	5.0×10^{-2}	8.0×10^0
300	Floor Drain Collector Tank	1.1×10^2	5.0×10^{-2}	5.5×10^0
306	Waste Sludge Phase Separator Tank	6.1×10^1	5.0×10^0	3.0×10^2
314	Floor Drain Sample Tank	7.8×10^1	5.0×10^{-2}	3.9×10^0
308	Chemical Waste Tanks	1.5×10^2	5.0×10^{-2}	7.5×10^0
309	Distillate Tanks	1.5×10^2	5.0×10^{-2}	7.5×10^0
310	Detergent Drain Tank	3.2×10^1	5.0×10^{-2}	1.6×10^1
311	Decontamination Solution Concentration Waste Tank	2.3×10^1	5.0×10^0	1.2×10^2
305	Spent Resin Tank	1.3×10^1	5.0×10^0	6.5×10^1
307	Cleanup Phase Separator Tanks	6.8×10^1	5.0×10^0	3.4×10^2
386	Decontamination Solution Concentrator	1.9×10^1	5.0×10^0	9.5×10^1
Total				3.2×10^3

Estimated total radioactivity deposited on all contaminated equipment internal surfaces = 6.3×10^3 Ci.

(a)The assumptions used to calculate the values in this table are listed in Section E.2.1.5.

(b)See Section C.2 in Appendix C.

(c)The internal surface area is estimated as being a factor of 5 times larger than the internal surface area of a right cylinder having dimensions similar to each particular piece of equipment.

E.2.2.1 Reference Radionuclide Inventory 5

The radionuclide mixture resulting from a small continuous leak is described by the differential equation:

$$\frac{dq}{dt} = P - \lambda q \quad (E.1)$$

where: q • the quantity of radioactivity present, Ci
t • the time unit of interest, day
P • the leak rate, Ci/day
 λ • the decay constant, days⁻¹, 0.693/half-life.

Integrating Equation E.1 and solving for q gives:

$$q = \frac{P(1 - e^{-\lambda t})}{\lambda} \quad (E.2)$$

The mixture of radionuclides found in the reactor water is shown in Table E.2-8.⁽²⁸⁾ Assuming a release of reactor water at the rate of 1 liter per day for 40 years, the resulting surface contamination inventory is found using Equation E.2. This mixture is shown in Table E.2-9 as reference radionuclide inventory 5. This inventory is used in Appendix F in determining the maximum annual dose to the maximum-exposed individual working in the decommissioned reference BWR facility. It is also used in Appendix N to assess public safety impacts of decommissioning.

E.2.2.2 Dose Rates from Externally Contaminated Structural Surfaces

Dose rates from structural surface contamination inside the reference BWR are calculated using the ISOSHL D computer code with a unit deposition level of radionuclide inventory 5. Floor surface areas are modeled by a disk source of 7-m radius (a floor area of 154 m²). The dose rate 1 m in air above this floor as a function of the surface deposition is shown in Figure E.2-8.

TABLE E.2-8. Reactor Water Radionuclide Concentrations in an Operating BWR

Radionuclide (a)	Half-Life (days)	Concentration ($\mu\text{Ci/g}$) (b)	Fractional Concentration
^3H	4.5×10^3	1×10^{-2} (c)	--- (c)
^{32}P	1.4×10^1	2×10^{-4}	1.5×10^{-2}
^{51}Cr	2.8×10^1	5×10^{-3}	3.8×10^{-1}
^{54}Mn	3.1×10^2	6×10^{-5}	4.5×10^{-3}
^{55}Fe	9.5×10^2	1×10^{-3}	7.5×10^{-2}
^{59}Fe	4.5×10^1	3×10^{-5}	2.2×10^{-3}
^{58}Co	7.2×10^1	2×10^{-4}	1.5×10^{-2}
^{60}Co	1.9×10^3	4×10^{-4}	3.0×10^{-2}
^{63}Ni	3.6×10^4	1×10^{-6}	7.5×10^{-5}
^{65}Zn	2.4×10^2	2×10^{-4}	1.5×10^{-2}
^{89}Sr	5.3×10^1	1×10^{-4}	7.5×10^{-3}
^{90}Sr	1.0×10^4	6×10^{-6}	4.5×10^{-4}
^{90}Y	2.7×10^0	6×10^{-6}	4.5×10^{-4}
^{91}Y	5.9×10^1	4×10^{-5}	3.0×10^{-3}
^{95}Zr	6.5×10^1	7×10^{-6}	5.2×10^{-4}
^{103}Ru	4.0×10^1	2×10^{-5}	1.5×10^{-3}
^{106}Ru	3.7×10^2	3×10^{-6}	2.2×10^{-4}
$^{110\text{m}}\text{Ag}$	2.5×10^2	1×10^{-6}	7.5×10^{-5}
$^{129\text{m}}\text{Te}$	3.4×10^1	4×10^{-5}	3.0×10^{-3}
^{131}I	8.0×10^0	5×10^{-3}	3.8×10^{-1}
^{134}Cs	7.5×10^2	3×10^{-5}	2.2×10^{-3}
^{136}Cs	1.4×10^1	2×10^{-5}	1.5×10^{-5}
^{137}Cs	1.1×10^4	7×10^{-5}	5.2×10^{-3}
^{140}Ba	1.3×10^1	4×10^{-4}	3.0×10^{-2}
^{140}Ca	1.7×10^0	4×10^{-4}	3.0×10^{-2}
^{141}Ce	3.2×10^1	3×10^{-5}	2.2×10^{-3}
^{144}Ce	2.8×10^2	3×10^{-6}	3.0×10^{-3}
^{143}Pr	1.4×10^1	4×10^{-5}	3.0×10^{-3}
^{147}Nd	1.1×10^1	3×10^{-6}	2.2×10^{-4}
Totals		1.3×10^{-2}	1.0

(a) Radionuclides with half-lives greater than 8 days and short-lived daughters of long-lived parents are included.

(b) Based on data from Reference 28.

(c) ^3H is not included in the total.

TABLE E.2-9. Reference Radionuclide Inventory 5,
BWR Structural Surface External Contamination

Radionuclide ^(a)	Fractional Radioactivity at Decay Times of:				
	Shutdown	10 Years	30 Years	50 Years	100 Years
³² P	1.1×10^{-3}	--- ^(b)	---	---	---
⁵¹ Cr	5.3×10^{-2}	---	---	---	---
⁵⁴ Mn	7.2×10^{-4}	2.3×10^{-7}	---	---	---
⁵⁵ Fe	3.7×10^{-1}	2.5×10^{-2}	1.2×10^{-4}	6.0×10^{-7}	---
⁵⁹ Fe	5.3×10^{-4}	---	---	---	---
⁵⁸ Co	5.6×10^{-3}	---	---	---	---
⁶⁰ Co	2.9×10^{-1}	7.8×10^{-2}	5.6×10^{-3}	4.0×10^{-4}	5.6×10^{-7}
⁶³ Ni	3.4×10^{-3}	3.2×10^{-3}	2.8×10^{-3}	2.4×10^{-3}	1.7×10^{-3}
⁶⁵ Zn	1.8×10^{-2}	4.5×10^{-7}	---	---	---
⁸⁹ Sr	2.0×10^{-3}	---	---	---	---
⁹⁰ Sr	1.5×10^{-2}	1.2×10^{-2}	7.0×10^{-3}	4.2×10^{-3}	1.2×10^{-3}
⁹⁰ Y	1.5×10^{-2}	1.2×10^{-2}	7.0×10^{-3}	4.2×10^{-3}	1.2×10^{-3}
⁹¹ Y	8.1×10^{-4}	---	---	---	---
⁹⁵ Zr	1.6×10^{-4}	---	---	---	---
⁹⁵ Nb	1.6×10^{-4}	---	---	---	---
¹⁰³ Ru	2.9×10^{-4}	---	---	---	---
¹⁰⁶ Ru	3.9×10^{-4}	---	---	---	---
^{110m} Ag	8.8×10^{-6}	3.2×10^{-10}	---	---	---
^{129m} Te	4.9×10^{-4}	---	---	---	---
¹³¹ I	1.5×10^{-2}	---	---	---	---
¹³⁴ Cs	8.8×10^{-3}	3.1×10^{-4}	3.7×10^{-7}	4.4×10^{-10}	---
¹³⁶ Cs	1.0×10^{-4}	---	---	---	---
¹³⁷ Cs	1.8×10^{-1}	1.4×10^{-1}	9.0×10^{-2}	5.7×10^{-2}	1.8×10^{-2}
¹⁴⁰ Ba	2.0×10^{-3}	---	---	---	---
¹⁴⁰ La	2.0×10^{-3}	---	---	---	---
¹⁴¹ Ce	3.4×10^{-4}	---	---	---	---
¹⁴⁴ Ce	2.9×10^{-4}	3.1×10^{-8}	---	---	---
¹⁴³ Pr	2.0×10^{-4}	---	---	---	---
¹⁴⁷ Nd	1.2×10^{-5}	---	---	---	---
Totals	1.0	2.7×10^{-1}	1.1×10^{-1}	6.8×10^{-2}	2.2×10^{-2}

(a) Radionuclides with half-lives greater than 8 days and short-lived daughters of long-lived parents are included.

(b) Indicates a value less than 1.0×10^{-10} .

FIGURE E.2-8. Calculated Dose Rates in Air 1 m Above Structural Surfaces as a Function of Deposited External Surface Radioactivity

E.2.2.3 Estimated Total Radioactivity on Externally Contaminated Structural Surfaces

The deposited amounts of reference radionuclide inventory 5 present on contaminated structural surfaces in the reference BWR are estimated using the following assumptions:

1. Contaminated concrete surfaces are based on the surfaces discussed in Appendix D.
2. Since it is difficult to characterize the quantity and location of structural surface contamination, surfaces are placed in three contamination categories. The first category is defined as nonradioactive areas with no reactor water, steam, or condensate process systems. The second category is defined as low-level contamination areas with dose rates of 10 mR/hr in air 1 m from the surface. The third category is defined as areas of higher contamination with dose rates of 100 mR/hr in air 1 m from the surface.

The structural surface contamination levels that correspond to dose rates of 10 and 100 mR/hr for radionuclide inventory 5 are estimated from Figure E.2-8

to be $2.5 \times 10^{-3} \text{ Ci/m}^2$ and $2.5 \times 10^{-2} \text{ Ci/m}^2$, respectively. The estimated structural surface external contamination in the reference BWR at shutdown is given in Table E.2-10. The total deposited radioactivity on structural surfaces is calculated to be about 110 Ci.

TABLE E.2-10. Estimated Structural Surface External Contamination in the Reference BWR

Building Associated Equipment System	Estimated Contaminated Surface Area (m^2) (a)	Radioactivity Deposition Level (b)	Deposited Radioactivity (Ci)
Reactor Building			
Containment Atmosphere Control	1.6×10^1	1	4.0×10^{-2}
Condensate (Nuclear Steam)	3.3×10^1	1	8.2×10^{-2}
Control Rod Drive	1.8×10^2	1	4.5×10^{-1}
Equipment Drain (Radioactive)	1.8×10^1	2	4.5×10^{-1}
Floor Drain (Radioactive)	7.4×10^1	2	1.8×10^0
Fuel Pool Cooling and Cleanup	1.2×10^3	1	3.0×10^0
Fuel Pool Cooling and Cleanup	2.8×10^2	2	7.0×10^0
High-Pressure Core Spray	1.1×10^2	1	2.7×10^{-1}
Low-Pressure Core Spray	1.4×10^1	1	3.5×10^{-2}
Main Steam	3.0×10^2	1	7.5×10^{-1}
Miscellaneous Wastes (Radioactive)	8.3×10^1	1	2.1×10^{-1}
Reactor Building Closed Cooling	1.2×10^1	1	3.0×10^{-2}
Reactor Core Isolation Cooling	1.5×10^1	1	3.8×10^{-2}
Reactor Water Cleanup	1.5×10^2	1	3.8×10^{-1}
Reactor Water Cleanup	1.7×10^2	2	4.2×10^0
Residual Heat Removal	1.7×10^2	1	4.2×10^{-1}
Standby Gas Treatment	4.0×10^1	1	1.0×10^{-1}
Traversing Incore Probe	8.0×10^1	1	2.0×10^{-1}
Primary Containment	2.2×10^3	2	5.5×10^1
Total			7.4×10^1
Turbine Generator Building			
Air Removal	3.9×10^1	1	9.7×10^{-2}
Condensate (Nuclear Steam)	6.6×10^2	1	1.6×10^{-1}
Condenser Off Gas Treatment	1.8×10^2	1	4.5×10^{-1}
Equipment Drain (Radioactive)	2.5×10^1	2	6.2×10^{-1}
Floor Drain (Radioactive)	2.5×10^1	2	6.2×10^{-1}
Heater Drain	9.1×10^1	1	2.3×10^{-1}
Main Steam	1.7×10^2	1	4.2×10^{-1}
Miscellaneous Drain and Vent	1.9×10^1	1	4.7×10^{-2}
Reactor Feedwater	6.9×10^2	1	1.7×10^0
Miscellaneous Wastes (Radioactive)	9.0×10^0	1	2.2×10^{-2}
Total			4.4×10^0
Radwaste and Control Building			
Condensate Filter Demineralizer	3.6×10^2	2	9.0×10^0
Condenser Off Gas Treatment	3.2×10^2	1	8.0×10^{-1}
Equipment Drain (Radioactive)	4.3×10^1	1	1.1×10^{-1}
Equipment Drain (Radioactive)	1.8×10^2	2	4.5×10^0
Floor Drain (Radioactive)	1.2×10^1	1	3.0×10^{-2}
Floor Drain (Radioactive)	1.9×10^2	2	4.8×10^0
Floor Pool Cooling and Cleanup	5.4×10^1	2	1.4×10^0
Miscellaneous Wastes (Radioactive)	2.4×10^1	1	6.0×10^{-2}
Miscellaneous Wastes (Radioactive)	1.9×10^2	2	4.8×10^0
Process Waste (Radioactive)	1.8×10^2	1	4.5×10^{-1}
Process Waste (Radioactive)	2.7×10^2	2	6.7×10^0
Reactor Water Cleanup	1.3×10^2	2	3.2×10^0
Total			3.6×10^1

Estimated total deposited radioactivity on contaminated external surfaces = $1.1 \times 10^2 \text{ Ci}$.

(a) The surface area is estimated using information in Appendices C and D

(b) 1 = $2.5 \times 10^{-3} \text{ Ci/m}^2$, and 2 = $2.5 \times 10^{-2} \text{ Ci/m}^2$

It should be noted that the dose rates used to estimate the structural surface external contamination are chosen to maximize the results and only partially reflect the expected levels at the time of final shutdown of the reference BWR. These dose rates are not used in calculating the occupational exposure to workers as they work around or remove the contaminated structural surfaces. The dose rates used for that purpose are presented in Appendix D.

E.2.3 Radionuclide Inventory on the Reference Site

This section contains a discussion of the postulated deposited radionuclide contamination level and mixture on the reference site resulting from normal BWR operation. Accidental releases of radionuclides are not expected to significantly increase the radioactivity on the reference site and, thus, are not considered in this analysis. Information about the level and nature of the radioactive contamination present at the time of decommissioning is needed to determine the alternative future uses of the site.

E.2.3.1 Radionuclide Release Data and Inventory Calculational Methods

Low levels of radioactive contamination are present on the reference site as a result of deposition of airborne radionuclide releases during 40 years of normal BWR operation. In addition, naturally occurring radionuclides and those resulting from nuclear weapons testing are present on the site, but deposition of these latter radionuclides is not quantified in this study.

Annual airborne radionuclide releases from operating BWRs vary widely and are dependent on such plant factors as size, operating conditions, and gaseous radwaste systems. For this study, the releases are based on gaseous and particulate airborne releases from 23 operating BWRs reported for 1975.⁽²⁹⁾ This was a time when fuel failures were higher than at present, and may produce an overestimate of normal releases expected over the operating life of a plant. The values for ^3H and ^{14}C are obtained from Reference 30. The reported airborne release values are divided by the total heat production of all 23 plants to obtain an average release rate for each radionuclide. These average release rates are multiplied by the reference plant thermal production capacity, the plant availability factor, and the number of hours in a year to obtain the annual release rates from the reference plant. These are reported in Table E.2-11.

TABLE E.2-11. Annual Airborne Radionuclide Release Rates from the Reference BWR

Radionuclide	Half-Life (days)	Average Airborne Release Rate (Ci/MWht)(a)	Calculated Reference BWR Annual Airborne Release Rate (Ci/yr)(b)
³ H	4.5×10^3	---(c)	4.0×10^0 (d)
¹⁴ C	2.1×10^6	---(c)	8.9×10^0 (d)
⁵¹ Cr	2.8×10^1	1.6×10^{-10}	3.5×10^{-3}
⁵⁴ Mn	3.0×10^2	1.9×10^{-9}	4.1×10^{-2}
⁵⁹ Fe	4.5×10^1	5.8×10^{-11}	1.3×10^{-3}
⁵⁸ Co	7.2×10^1	6.6×10^{-10}	1.4×10^{-2}
⁶⁰ Co	1.9×10^3	4.1×10^{-9}	8.9×10^{-2}
⁶⁵ Zn	2.4×10^2	1.8×10^{-10}	3.9×10^{-3}
⁸⁹ Sr	5.3×10^1	3.1×10^{-8}	6.8×10^{-1}
⁹⁰ Sr	1.0×10^4	6.3×10^{-9}	1.4×10^{-1}
⁹⁰ Y	2.7×10^0	6.3×10^{-9}	1.4×10^{-1}
⁹⁵ Zr	6.5×10^1	1.6×10^{-10}	3.5×10^{-3}
⁹⁵ Nb	3.5×10^1	1.6×10^{-10}	3.5×10^{-3}
¹⁰⁶ Ru	3.7×10^2	3.0×10^{-10}	6.5×10^{-3}
^{110m} Ag	2.6×10^2	2.3×10^{-10}	5.0×10^{-3}
¹²⁴ Sb	6.0×10^1	7.4×10^{-13}	1.6×10^{-5}
¹²⁵ Sb	9.9×10^2	4.0×10^{-13}	8.7×10^{-6}
¹³¹ I	8.0×10^0	9.0×10^{-9}	2.0×10^{-1}
¹³³ I	8.4×10^{-1}	8.0×10^{-9}	1.7×10^{-1}
¹³⁴ Cs	7.5×10^2	4.6×10^{-9}	1.0×10^{-1}
¹³⁶ Cs	1.4×10^1	2.0×10^{-11}	4.4×10^{-4}
¹³⁷ Cs	1.1×10^4	8.6×10^{-9}	1.9×10^{-1}
¹⁴⁰ Ba	1.3×10^1	3.7×10^{-8}	8.1×10^{-1}
¹⁴⁰ La	1.7×10^0	3.7×10^{-8}	8.1×10^{-1}
¹⁴¹ Ce	3.2×10^1	3.8×10^{-10}	8.3×10^{-3}
¹⁴⁴ Ce	2.8×10^2	1.3×10^{-10}	2.8×10^{-3}
Total			3.4×10^0

(a) Calculated from release data from 23 operating BWRs in 1975 (Reference 29).

(b) Calculated using the reference BWR size of 3320 MWt, plant availability factor of 0.75, and 8.8×10^3 hr/yr.

(c) No value calculated from Reference 29 data.

(d) Calculated from release value found in Reference 30. Not included in the total since it is assumed to be released as a gas and not deposited on the site.

Ground deposition on the site from the continuous, elevated (150-m stack height) atmospheric release is calculated. Meteorological data on joint frequency of wind speed and direction by atmospheric stability class are not available for the reference site. Therefore, these data are taken from Reference 31. With these data, standard atmospheric models are used to estimate ground deposition.⁽³²⁾ Radioactive decay of the released radionuclides is accounted for over the duration of the release and the following ground deposition period. Tritium (^3H) and ^{14}C are assumed to be released as gases and not deposited on the site. The concentrations of radionuclides deposited on the site over the 40-year plant life are calculated using the XOQDOQ computer program.⁽³³⁾ No credit is taken in the calculation for plume rise from either buoyancy or momentum.

The procedure used to estimate relative deposition rates is based on numerical solutions to the flux-gradient ("K-Theory") diffusion equation.⁽³⁴⁾ The effluent is not allowed to diffuse beyond a height of 200 m in a stable condition and 1000 m in neutral and unstable conditions. At the ground surface, a partial-sink boundary condition involving the deposition velocity is assumed. The wind and eddy diffusivity profiles required as input to the diffusion equation are those in Reference 34. Deposition velocity is allowed to vary with wind speed in accordance with an empirical equation.⁽³⁵⁾ An areal grass density of approximately 70 g/m^2 is assumed. No allowance is made for normal weathering conditions (runoff, soil characteristics, and vegetational coverage) or for resuspension of deposited material.

E.2.3.2 Reference Radionuclide Inventory 6

The resulting radionuclide depositions on the reference site from 40 years of normal reference BWR operation are listed in Table E.2-12 as reference radionuclide inventory 6. This inventory is used in Appendix F in determining the maximum annual dose to the maximum-exposed individual living on the decommissioned reference site.

TABLE E.2-12. Reference Radionuclide Inventory 6,
Reference BWR Site Surface Contamination^(a)

Radionuclide	Deposited Radioactivity ^(b) ($\mu\text{Ci}/\text{m}^2$) at Decay Times of:				
	Shutdown	10 Years	30 Years	50 Years	100 Years
⁵¹ Cr	1.1×10^{-6}	--- ^(c)	---	---	---
⁵⁴ Mn	1.4×10^{-4}	3.0×10^{-8}	---	---	---
⁵⁹ Fe	6.6×10^{-7}	---	---	---	---
⁵⁸ Co	1.2×10^{-5}	---	---	---	---
⁶⁰ Co	1.9×10^{-3}	5.0×10^{-4}	3.5×10^{-5}	2.4×10^{-6}	3.1×10^{-9}
⁶⁵ Zn	1.1×10^{-5}	2.8×10^{-10}	---	---	---
⁸⁹ Sr	4.1×10^{-4}	---	---	---	---
⁹⁰ Sr	1.0×10^{-2}	7.8×10^{-3}	4.7×10^{-3}	2.8×10^{-3}	8.0×10^{-4}
⁹⁰ Y	1.0×10^{-2}	7.8×10^{-3}	4.7×10^{-3}	2.8×10^{-3}	8.0×10^{-4}
⁹⁵ Zr	2.6×10^{-6}	---	---	---	---
⁹⁵ Nb	2.6×10^{-6}	---	---	---	---
¹⁰⁶ Ru	2.7×10^{-5}	2.9×10^{-8}	---	---	---
^{110m} Ag	1.5×10^{-6}	8.7×10^{-10}	---	---	---
¹²⁴ Sb	1.1×10^{-8}	---	---	---	---
¹²⁵ Sb	9.7×10^{-8}	7.5×10^{-9}	---	---	---
¹³¹ I	1.8×10^{-5}	---	---	---	---
¹³³ I	1.6×10^{-6}	---	---	---	---
¹³⁴ Cs	8.5×10^{-4}	2.9×10^{-5}	3.4×10^{-8}	---	---
¹³⁶ Cs	6.9×10^{-8}	---	---	---	---
¹³⁷ Cs	1.4×10^{-2}	1.1×10^{-2}	7.1×10^{-3}	4.5×10^{-3}	1.4×10^{-3}
¹⁴⁰ Ba	1.2×10^{-4}	---	---	---	---
¹⁴⁰ La	1.2×10^{-4}	---	---	---	---
¹⁴¹ Ce	3.0×10^{-6}	---	---	---	---
¹⁴⁴ Ce	8.8×10^{-6}	1.1×10^{-9}	---	---	---
Totals	3.8×10^{-2}	2.7×10^{-2}	1.6×10^{-2}	1.0×10^{-2}	3.0×10^{-3}

(a) Based on 1975 reported airborne radionuclide releases from 23 operating BWRs (Reference 29).

(b) Based on 30 EFY of operation.

(c) Indicates a value less than $1.0 \times 10^{-10} \mu\text{Ci}/\text{m}^2$.

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APPENDIX F

PUBLIC RADIATION DOSE MODELS AND CALCULATED MAXIMUM ANNUAL DOSES

This appendix contains support material required for Chapters 8 and 11. Terminology, models, equations and parameters used in radiation dose calculations, as well as tabulated maximum annual doses to the maximum-exposed individual, are presented. The tabulated maximum annual doses are based on the reference radionuclide inventories detailed in Appendix E. These doses are used in determining the acceptable contamination levels for public release of the decommissioned reference BWR power station. It should be noted that the occupational radiation doses during decommissioning operations are not calculated using the dose models presented in this appendix. Rather, they are estimated using measured dose rate data, as discussed in Appendix D.

F.1 RADIATION DOSE TERMINOLOGY AND DEFINITIONS

The following terminology and definitions apply for the radiation dose calculations made for this study:

Routine Release. An atmospheric release of radionuclides from routine decommissioning tasks that is assumed to occur at a constant rate for 1 year.

Acute Release. A short-duration (usually less than 8 hours) release of radionuclides. The accident analysis, discussed in Chapter 11 and Appendix N, are based on acute releases.

Organs of Reference. The organs of the human body for which radiation doses are calculated. For this study, the organs of reference are the total body, lungs, bone, and thyroid. The total body is the head and trunk of the human body and includes active blood-forming organs, eye lenses, and gonads.

Exposure Pathways. The potential routes by which people may be exposed to radionuclides or radiation. Radiation exposure pathways in the environment

that are considered in this study are: external exposure to contamination deposited on the ground, ingestion of food products containing radionuclides, and inhalation of airborne radionuclides. Radiation exposure pathways inside the BWR facility are: external exposure from contaminated room surfaces and inhalation of airborne radionuclides. External exposure from airborne radionuclides (air submersion) is not considered, since previous decommissioning studies have shown this exposure pathway to be insignificant compared to the others.⁽¹⁻³⁾

Maximum-Exposed Individual. The individual who receives the maximum radiation dose to an organ of reference. The maximum-exposed individual is assumed to reside at the location of the highest airborne radionuclide concentration. Maximized exposure pathway parameters are used.

Radiation Dose Equivalent. A quantity used to express the absorbed dose from all radiations on a common scale in units of rem. It is defined as the product of the absorbed dose in rads, a quality factor intended to allow for the effect of the microscopic distribution of absorbed energy, and other modifying factors as defined by the International Commission on Radiological Protection (ICRP).

Committed Radiation Dose Equivalent. The dose equivalent accumulated over a specific number of years from radionuclides deposited in the body from the various radiation pathways during a given year.

Collective Dose. The summation of radiation dose equivalents received by all individuals in the population of concern. It is calculated by multiplying the dose to the average individual by the population distribution for the site listed in Appendix B. Average parameter values are used for the exposure pathways.

First-Year Dose. The radiation dose equivalent accumulated during the year of the release period.

Fifty-Year Committed Dose Equivalent. The committed radiation dose equivalent for 50 years following a 1-year exposure to radionuclides.

Annual Dose. The radiation dose equivalent calculated during any year following the start of continuous exposure. It is the sum of the dose received by an organ of reference during the year of interest from all exposure pathways and the dose received that year from radionuclides deposited in the organ of reference during the previous years.

Maximum Annual Dose. The largest of the 50 annual doses calculated to occur during the 50 years following the start of continuous exposure.

Biological Half-life. The time required for the body to eliminate half of any substance by regular body elimination processes.

Solubility Classes D, W, and Y. Radionuclides are classified according to the rate at which they are dissolved into body fluids in the lung after inhalation. The three classifications used in this study are: Class D materials, with a maximum biological half-life of less than 1 day; Class W materials, with a maximum biological half-life ranging from a few days to a few months; and Class Y materials, with a maximum biological half-life of from 6 months to a few years.⁽⁴⁾ Table F.1-1 contains a list of the solubility classes for the elements considered in this study.

F.2 GENERAL MODELS FOR ESTIMATING RADIATION DOSE

The methodology and parameters necessary to calculate the first-year dose or the fifty-year committed dose equivalent are presented in this section. The fundamental relationship for calculating radiation doses to people from any radionuclide or exposure pathway is given in Equation F.1.⁽⁶⁾

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

where:

- R_{ipr} • first-year dose or fifty-year committed dose equivalent from radionuclide i via exposure pathway p to organ r

TABLE F.1-1. Solubility Classes for the Elements Present
in the Radionuclide Inventory at Final
Shutdown of the Reference BWR

<u>Element</u>	<u>Solubility Class^(a) (all other organs/lungs)</u>	<u>Reference^(b)</u>
Hydrogen (H)	D/D	4
Carbon (C)	D/D	4
Phosphorus (P)	D/W	4
Chromium (Cr)	W/Y	4
Manganese (Mn)	D/W	5
Iron (Fe)	W/Y	5
Cobalt (Co)	W/Y	5
Nickel (Ni)	W/W	4
Zinc (Zn)	W/Y	4
Strontium (Sr)	D/Y	5
Yttrium (Y)	D/Y	5
Zirconium (Zr)	W/Y	5
Niobium (Nb)	W/Y	5
Technetium (Tc)	D/W	5
Ruthenium (Ru)	Y/Y	5
Iodine (I)	D/D	5
Cesium (Cs)	D/D	5
Barium (Ba)	D/W	4
Lanthanum (La)	W/Y	4
Cerium (Ce)	W/Y	4

(a) Solubility classes are for use in the ICRP Task
Group Lung Model.

(b) Reference numbers shown at the end of this appendix.

- C_{ip} • concentration of radionuclide i in the media of exposure pathway p ; for calculations involving airborne radionuclides, C_{ip} is replaced with the term $\bar{X}_i$, which represents the average airborne concentration of radionuclide i
- U_p • usage parameter (exposure rate or intake rate) associated with exposure pathway p
- D_{ipr} • radiation dose equivalent factor or fifty-year committed radiation dose equivalent factor (for a given radionuclide i , exposure pathway p , and organ r) that converts the concentration and usage parameters to the first-year dose or fifty-year committed dose equivalent.

Each of the terms in the general dose equation (Equation F.1) are discussed in detail below:

C_{ip} . An analysis of the radiation doses from separate exposure pathways requires a calculation of the radionuclide concentrations associated with each pathway. An external exposure analysis requires a soil concentration or surface contamination level. The radionuclide concentration in food crops grown in contaminated soil is directly related to the radionuclide concentration in the soil plow layer (a depth of 0.15 m). Exposure pathways that require the concentration of radionuclides in air from airborne radionuclide releases use dilution factors discussed in the site description in Appendix B. They are: $7.2 \times 10^{-8} \text{ sec/m}^3$ for the maximum-exposed individual located 1 km from the site and $2.3 \times 10^{-11} \text{ sec/m}^3$ for the average member of the population residing within 80 km of the site.

U_p . Each radiation exposure pathway has an associated exposure rate or intake rate. Parameters used to calculate radiation doses from the consumption of food products grown in contaminated soil are listed in Table F.2-1. For the inhalation pathway, the intake rate (or ventilation rate) is assumed to be $2.7 \times 10^{-4} \text{ m}^3/\text{sec}$ for members of the public and $3.3 \times 10^{-4} \text{ m}^3/\text{sec}$ for workers with a 40-hour work week.

TABLE F.2-1. Parameters Used for Calculation of Radiation Doses from Consumption of Foods^(a)

Food	Growing Period (days)	Yield (kg/m ²)	Holdup (days) ^(b)	Up, Maximum Individual Consumption (kg/year) ^(c)	Up, Average Individual Consumption (kg/year)
Leafy Vegetables	90	1.5	1	30	15
Other Above-Ground Vegetables	60	0.7	1	30	15
Potatoes	90	4.0	10	110	7.1
Other Root Vegetables	90	5.0	1	72	47
Berries	60	2.7	1	30	6
Melons	90	0.8	1	40	8
Orchard Fruit	90	1.7	10	265	50
Wheat	90	0.72	10	80	72
Other Grain	90	1.4	1	8	80
Eggs	90	0.84 ^(d)	2	30	20
Milk	30	1.3 ^(d)	2	274 ^(e)	230 ^(e)
Beef	90	0.84 ^(d)	15	40	40
Pork	90	0.84 ^(d)	15	40	30
Poultry	90	0.84 ^(d)	2	18	8

(a)Based on information in Reference 7.

(b)Time between harvest and consumption.

(c)Only that fraction of the diet grown locally (and, therefore, potentially contaminated) is listed.

(d)Yield of animal feeds (i.e., grain or pasture grass).

(e)Units of g/year.

D_{ipr} . The radiation dose equivalent factors and the fifty-year committed radiation dose equivalent factors that are used in this study for internal exposure pathways are obtained from several sources. Those for ingestion are obtained primarily from References 6, 8, 9, 10, and 11, which are derived from information given by the ICRP for the body burden and the maximum permissible concentration of each radionuclide.⁽¹²⁾ Ingestion factors not obtainable from the references are calculated by the method given in Reference 11. In this method, effective decay energies for the radionuclides are calculated from the ICRP model, which assumes that all of a given radionuclide is in the center of a spherical organ with an appropriate effective radius. Dose factors for inhalation are calculated using the computer code DACRIN.⁽¹³⁾ This code

incorporates the ICRP Task Group Lung Model to calculate the dose to the lung and other organs of reference. A particle size of 1 micrometer in diameter is assumed. Radiation dose equivalent factors for internal pathways have units of mrem per year per pCi taken into the body during that same year. Committed radiation dose equivalent factors have units of mrem/50-yr per pCi taken into the body in the first year. External exposure can arise from either contaminated land or contaminated building surfaces. In this study, external dose equivalent factors are calculated for these sources using the ISOSHL computer code.^(14,15) Ground contamination is modeled by calculating the dose at a point 1 m in air above a semi-infinite slab of radionuclides evenly distributed in a 0.15-m layer of soil. Total body dose rates are obtained by using a 50-mm tissue equivalent thickness around the dose point.

Discussion of the various radiation exposure pathways considered in this study, together with appropriate modifications to the general dose equation and its parameters, are given in the following sections.

F.2.1 External Exposure

Calculation of the radiation dose resulting from external total body exposure to surface contamination is done with no modification to Equation F.1. Soil concentrations are calculated using a simple cloud-depletion model, as is described for the food ingestion pathway analysis in the next section. For routine releases from decommissioning operations, the maximum-exposed individual is assumed to be exposed to contaminated soil for 12 hours a day for a full year. The average individual is assumed to be exposed for 6 hours a day for a full year. The fifty-year committed dose equivalent for external exposure to surface contamination is equal to the first-year dose, since there is no internal radionuclide deposition.

F.2.2 Ingestion

The radiation dose equivalent or committed radiation dose equivalent to a person consuming vegetation grown on the contaminated site is given in Equation F.2.

$$R_{vr} = \sum_{i=1}^n C_{iv} U_v D_{ir} \quad (F.2)$$

Similarly, the radiation dose equivalent or committed radiation dose equivalent to a person consuming a contaminated animal product is given in Equation F.3.

$$R_{ar} = \sum_{i=1}^n C_{ia} U_a D_{ir} \quad (F.3)$$

where:

- R_{vr}, R_{ar} • radiation dose equivalent or committed radiation dose equivalent from vegetation v or animal a food product pathways, mrem or mrem/50 yr
- C_{iv}, C_{ia} • concentration of radionuclide i in vegetation v or animal product a, pCi/kg or pCi/l
- U_v, U_a • annual consumption of contaminated vegetable v or animal products a, kg or l
- D_{ir} • radiation dose equivalent factor or committed radiation dose equivalent factor, mrem/pCi or mrem/50 yr per pCi.

Specific values of U_v and U_a are shown in Table F.2-1.

F.2.2.1 Deposition on Food Products or Ground Surfaces

Equation F.4 is used to describe the deposition of airborne particulate radionuclides directly on food products or ground surfaces.

$$d_i = 86,400 \bar{X}_i V_{di} \quad (F.4)$$

where:

- d_i • deposition rate of flux of radionuclide 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 from airborne releases of radionuclide i , pCi/m³.

Specific values of the deposition velocity, V_{di} , are obtained from Reference 16.

F.2.2.2 Concentration in Vegetation

The concentration of radioactive material in vegetation resulting from direct deposition onto plant foliage and uptake of prior depositions from the soil is given in Equation F.5.

$$C_{iv} = d_i \left[\frac{r T_v (1 - e^{-\lambda_{Ei} t_e})}{Y_v \lambda_{Ei}} + \frac{B_{vi} (1 - e^{-\lambda_i t_b})}{P \lambda_i} \right] e^{-\lambda_i t_h} \quad (F.5)$$

where:

- C_{iv} • concentration of radionuclide i in the edible portion of the vegetation v , pCi/kg
- d_i • deposition rate or flux of radionuclide i , pCi/(m²-day)
- r • fraction of deposition retained on the vegetation (dimensionless), assumed to be 0.25 in this study
- T_v • factor for translocation of externally deposited radionuclides to the edible parts of the vegetation (dimensionless). For simplicity it is assumed in this study to be independent of the radionuclide and is 1.0 for leafy vegetables and fresh forage, and 0.1 for all other produce, including grain. (Reference 9 lists values of this parameter for various radionuclides.)
- λ_i • radiological decay constant for radionuclide i , day⁻¹
- λ_{Ei} • effective removal constant for radionuclide i , day⁻¹;
 $\lambda_{Ei} = \lambda_i + \lambda_w$, where λ_i is as defined above and λ_w is the weathering removal constant for vegetation, day⁻¹, assumed to be $0.693/14 = 5 \times 10^{-2}$ day⁻¹

- Y_v • yield of vegetation v, kg (wet weight)/m²
- B_{vi} • concentration factor for uptake of radionuclide i from the soil in vegetation v, pCi/kg (wet weight) per pCi/kg (dry soil)
- t_b • time for buildup of radionuclides in the soil, days; assumed in this study to be 40 years of routine BWR plant operation at 75% efficiency
- t_e • time of exposure of above-ground vegetation to contamination during growing season, days
- t_h • holdup time between harvest and food consumption, days
- P • soil "surface density," kg (dry soil)/m²; a value of 224 kg/m² is used assuming the contaminated ground is plowed to a depth of 0.15 m after decommissioning.

The first term inside the brackets of Equation F.5 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 BWR plant operating life. Specific values for the parameters in Equation F.5 are obtained from References 7, 8, and 16.

F.2.2.3 Concentration in Animal Products

The radionuclide concentration in animal products 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 F.6.

$$C_{ia} = S_{ia} C_{iF} Q_F \quad (F.6)$$

where:

- C_{ia} • concentration of radionuclide i in animal product a, pCi/kg or pCi/l
- S_{ia} • transfer coefficient of radionuclide i to the edible portion of animal product a from daily intake by the animal, pCi/l (milk) per pCi/day or pCi/kg (animal product) per pCi/day

C_{iF} • concentration of radionuclide i in feed or forage F , pCi/kg;
calculated from Equation F.5

Q_F • animal consumption rate of contaminated feed or forage F , kg/day.

Specific values for the parameters used in Equation F.6 are obtained from References 7, 8 and 16. Where data are lacking, comparisons are made with biological data from chemically similar elements.

F.2.3 Inhalation

The radiation dose equivalent or the committed radiation dose equivalent via inhalation is calculated using Equation F.7.

$$R_{ir} = \sum_{i=1}^n \bar{X}_i B T D_{ir} \quad (F.7)$$

where:

R_{ir} • radiation dose equivalent or committed radiation dose equivalent to organ r from radionuclide i , mrem or mrem/50 yr

$\bar{X}_i$ • annual average airborne concentration of radionuclide i , pCi/m³

B • ventilation rate of exposed individual:
for a chronic release, $B = 2.7 \times 10^{-4}$ m³/sec;
for an acute release, $B = 3.3 \times 10^{-4}$ m³/sec⁽¹²⁾

T • time of exposure to the airborne radionuclide concentration, sec

D_{ir} • radiation dose equivalent factor or committed radiation dose equivalent factor, mrem/pCi or mrem/50 yr per pCi.

The inhalation dose model presented in Equation F.7 is consistent with the ICRP Task Group Lung Model.

F.3 MODEL FOR ESTIMATING ANNUAL DOSE

The determination of acceptable residual contamination levels at the decommissioned reference BWR (discussed in Chapter 8) is made by direct comparisons of calculated annual doses with an assumed annual dose limit. The

annual dose to internal body organs from internally deposited radionuclides tends to increase for a time (after the start of continuous exposure to a radioactively decaying source) until a maximum is reached. The annual dose then tends to decrease with time because of radioactive decay, decrease in the radionuclide concentrations of specific exposure pathways, and biological elimination of radionuclides deposited in the organ. For continuous exposure to a radioactively decaying source, the year in which the annual dose is a maximum depends on the chemical and physical characteristics of the radionuclides in the source, the organ of reference, and the exposure pathway.

The calculated first-year dose from ingested or inhaled radionuclides will most likely underestimate the maximum annual dose during continuous exposure, and therefore does not provide the best comparison to an annual dose limit. Nor is it appropriate to compare a fifty-year dose (accounting for 50 years of committed radiation dose equivalent) to an annual dose limit. Therefore, an extension of the general dose relationship given in Equation F.1 is required to calculate annual doses for each year of a continuous exposure period. The maximum annual dose occurring in the continuous exposure period is then compared to the annual dose limit to determine acceptable residual contamination levels.

Calculation of the annual dose to an organ of reference requires the dose equivalent from exposure during the year of interest. It also requires a detailed accounting of the committed dose equivalents to that organ from radionuclides deposited during previous years of intake. The general expression for annual dose calculations is deduced by inspecting the annual dose equations for the first 3 years of continuous exposure. The annual dose for the first year to an organ of reference is simply the summation of the radiation dose equivalents from all internal and external exposure pathways. For the second year, the annual dose is calculated by the following expression:

$$A_2 = R_2^* + (R_{1,2} - R_{1,1}) \quad (\text{F.8a})$$

where:

- A_2 • the annual dose during the second year from all exposure pathways to the organ of reference, mrem
- R_2^* • the radiation dose equivalent in the second year to the organ of reference from all internal and external exposure pathways from intake and exposure in the second year of continuous exposure, mrem
- $R_{1,2}$ • the committed dose equivalent to the organ of reference for the first two years from radionuclides internally deposited during intake from exposure pathways in the first year, mrem
- $R_{1,1}$ • the radiation dose equivalent to the organ of reference for the first year from radionuclides internally deposited during intake from exposure pathways in the first year (no external component to the dose equivalent), mrem.

The second-year annual dose to an organ of reference (A_2 in Equation F.8a) is the summation of the radiation dose equivalents from all exposure pathways during the second year and the dose equivalent resulting from the radionuclides internally deposited in that organ during the first year. The term in parentheses in Equation F.8a is the expression for the dose equivalent to the organ of reference from radionuclides deposited in that organ in the first year. It is found by subtracting the first-year dose equivalent from internally deposited radionuclides from the two-year committed dose equivalent.

The mathematical expression for the annual dose to an organ of reference in the third year of continuous exposure is shown in Equation F.8b.

$$A_3 = R_3^* + (R_{1,3} - R_{1,2}) + (R_{2,2} - R_{2,1}) \quad (\text{F.8b})$$

where:

- A_3 • the annual dose during the third year from all exposure pathways to the organ of reference, mrem

- R_3^* • the radiation dose equivalent in the third year to the organ of reference from all internal and external exposure pathways from intake and exposure in the third year of continuous exposure, mrem.

The terms $R_{1,3}$, $R_{1,2}$, $R_{2,2}$, and $R_{2,1}$ are of similar form, each having two subscripts. The first subscript defines the year of intake or exposure after the start of continuous exposure, and the second defines the number of years used in calculating the committed dose equivalent.

The quantity in the first parentheses in Equation F.8b is the dose equivalent to the organ of reference in the third year from radionuclides deposited in the first year of continuous exposure (i.e., the difference between the 3-year committed dose equivalent and the 2-year committed dose equivalent). The quantity in the second parentheses is the dose equivalent in the third year to the organ of reference from radionuclides deposited in the second year of continuous exposure (i.e., the difference between the 2-year committed dose equivalent and the first-year committed dose equivalent).

The general expression for calculating the annual dose to an organ of reference during any year after the start of continuous exposure can be expressed as:

$$A_t = R_t^* + \sum_{i=1}^{t-1} [R_{i,(t-i+1)} - R_{i,(t-i)}] \quad (F.9)$$

where:

- A_t • the annual dose during the year t from all exposure pathways to the organ of reference, mrem
- R_t^* • the radiation dose equivalent in year t to the organ of reference from all internal and external exposure pathways from intake and exposure in the year t , mrem.

The summation term represents the dose equivalent delivered to the organ of reference in year t from radionuclides deposited in the organ from intake in all previous years since the start of continuous exposure.

The summation term in Equation F.9 is valid only for integer values of t greater than 1. For values of t equal to or less than 1, the subscripts define a non-real case, and the summation term is set equal to zero.

The annual dose, A_t , to the organ of reference is calculated for each value of t from 1 to 50, and the maximum annual dose is determined by inspection. The radionuclide inventories are adjusted for radionuclide decay and daughter-product buildup during the calculation.

To determine acceptable residual contamination levels for the BWR facility, a model of a reference room is required. Modifications to the models for the airborne exposure pathway are also required since no further airborne release occurs after decommissioning. The following sections contain discussions of these topics.

F.3.1 Model of a Reference BWR Room

To determine the acceptable contamination levels in the decommissioned BWR facility, the dimensions of a reference room must be defined. The results of a sensitivity study of the relationship between room volume, surface radionuclide levels, and calculated dose rate are shown in Figure F.3-1. Dose rates are calculated at a point in air (in a tissue equivalent 50 mm thick) 1 m above the floor of a room with 1 Ci/m² of ⁶⁰Co uniformly distributed on the room surfaces. The curves in Figure F.3-1 graphically illustrate the difference between an evenly contaminated room and one whose walls and ceiling contain 50% and 10%, respectively, of the floor contamination level. The straight line represents the dose rate from an infinite flat plane of evenly distributed ⁶⁰Co contamination with a concentration of 1 Ci/m². It can be seen that the dose rate for a large, non-uniformly contaminated room may exceed that of a uniformly contaminated infinite plane.

To be conservative, a room with uniformly contaminated walls, floor, and ceiling is used in this study. It has a floor area of 154 m² and 3-m-high

FIGURE F.3-1. Dose Rate as a Function of Room Volume for a ^{60}Co Deposition of 1 Ci/m^2

walls, for a total volume of about 460 m^3 . External radiation dose equivalent factors for this reference room are calculated using the ISOSHL D computer code.^(14,15) Annual doses are determined for a person working in this reference room at the decommissioned BWR for 40 hours a week for 50 years.

F.3.2 Airborne Radioactivity Concentrations for Annual Dose Calculations

Resuspended surface or ground contamination is assumed to be the only source of airborne radionuclides following decommissioning of the reference BWR. Airborne radionuclide concentrations for the decommissioned BWR are calculated using a resuspension analysis. The average airborne concentration is the product of a resuspension factor and the surface contamination level as shown in Equation F.10.

$$\bar{X} = S_f S_A \quad (\text{F.10})$$

where:

- $\bar{X}$ • average airborne concentration, Ci/m³
- S_f • resuspension factor, m⁻¹
- S_A • surface radioactivity deposition, Ci/m².

Inside the decommissioned BWR facility, airborne concentrations are calculated using a constant resuspension factor of $5 \times 10^{-6} \text{ m}^{-1}$. This value is one tenth of the value suggested by the IAEA for surfaces with removable contamination.⁽¹⁷⁾ This lower value is assumed since unrestricted release of nuclear facilities should not be contemplated with readily removable contamination remaining on accessible surfaces. Actual facility surface resuspension factors may be considerably lower than this value.

Airborne radionuclide concentrations on the reference site are calculated using a time dependent resuspension factor to account for the environmental "aging" of radionuclides. The relationship for this factor is given in Equation F.11.⁽¹⁸⁾

$$S_f = (10^{-4} e^{-\lambda\sqrt{t}}) + 10^{-9} \quad (\text{F.11})$$

where:

- S_f • resuspension factor, m⁻¹
- 10^{-4} • resuspension factor at time $t = 0$, m⁻¹
- λ • effective decay constant controlling the availability of material for resuspension, $0.15 \text{ day}^{-1/2}$
- t • time after deposition, days
- 10^{-9} • resuspension factor after 17 years, m⁻¹.

The second term in Equation F.11 (10^{-9}) is added based on the assumption that there is no further measurable decrease in the resuspension factor process after 17 years, which is the longest period for which data are available.

This is deemed appropriate since this model is derived empirically to simulate experimental measurements, and contains no fundamental understanding of the resuspension process.⁽¹⁸⁾ The time dependence of the site resuspension factor is illustrated in Figure F.3-2.

After decommissioning, unrestricted use of the site will allow activities such as unlimited farming. Therefore, 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 0.15 m, and the original surface contamination is assumed to be uniformly mixed into that volume of soil. This reduces the amount of material available for resuspension.

FIGURE F.3-2. Time-Dependent Surface Resuspension Factor for the Site

F.4 CALCULATED MAXIMUM ANNUAL DOSES AT THE DECOMMISSIONED REFERENCE BWR

This section contains the assumptions, specifies the radionuclide inventories, and presents listings of the resultant calculated maximum annual doses. These maximum annual doses are used in determining the acceptable residual contamination levels for the decommissioned reference BWR that are presented in Chapter 8. Doses are calculated and reported for the maximum-exposed individual working in the BWR facility or residing on its site after decommissioning.

F.4.1 Assumptions for Calculating Maximum Annual Dose

The following assumptions are used in calculating the maximum annual doses at the decommissioned BWR:

- The radionuclide inventories are undisturbed, except for radioactive decay after plant shutdown. Doses are calculated for the mixtures at shutdown, and for decay periods of 10, 30, 50, and 100 years.
- The maximum-exposed individual is considered to have realistically maximized exposure pathway parameters.
- The maximum-exposed individual working in the facility receives continuous exposure for 40 hours a week, 50 weeks a year.
- The maximum-exposed individual residing on the site is continuously exposed for 168 hours per week, 52 weeks per year.
- The airborne radionuclide concentrations are calculated using a resuspension factor of $5 \times 10^{-6} \text{ m}^{-1}$ inside the facility, and the time dependent factor from Equation F.11 on the site.

F.4.2 Radionuclide Inventories for Calculating Maximum Annual Dose

The radionuclide inventories used to calculate the maximum annual doses at the decommissioned BWR are reference radionuclide inventories 5 and 6, which are presented and discussed in Appendix E. Table E.2-7 contains a listing of reference radionuclide inventory 5, which characterizes surface contamination inside the reference BWR.⁽¹⁹⁾ Table E.2-10 contains a listing of reference radionuclide inventory 6, which characterizes the contamination on the ground surface of the site.⁽²⁰⁾

F.4.3 Maximum Annual Dose Listings

Calculated maximum annual doses to the maximum-exposed individual using the decommissioned BWR facility or its site are listed in Tables F.4-1 and F.4-2. These doses are calculated using the models, parameters, and assumptions presented earlier in this appendix. Dose pathways of concern in the reference BWR facility are inhalation of resuspended surface contamination and external exposure to surface contamination. The calculated maximum annual doses to an individual working in the decommissioned facility for 40 hours a week, 50 weeks a year, are listed in Table F.4-1. Dose pathways of concern on the decommissioned reference site are inhalation of resuspended ground contamination, external exposure to ground contamination, and ingestion of food products raised on the site. The calculated maximum annual doses to an individual residing on the reference site 24 hours a day, 52 weeks a year, are listed in Table F.4-2. These tables contain doses calculated using the radionuclide mixtures present at plant shutdown, and at decay periods of 10, 30, 50, and 100 years after plant shutdown. These doses are used to determine the acceptable residual radionuclide contamination levels for unrestricted release of the decommissioned BWR facility and site, which are presented in Chapter 8.

TABLE F.4-1. Estimated Maximum Annual Dose to the Maximum-Exposed Individual Working in the Decommissioned BWR Facility at Various Times After Shutdown(a)

Continuous Exposure in the Facility Starts at Plant Shutdown					Continuous Exposure in the Facility Starts 10 Years After Plant Shutdown					Continuous Exposure in the Facility Starts 30 Years After Plant Shutdown										
Organ of Reference/ Maximum Year(b)	Radionuclide(c)	Maximum Annual Dose (rem) via			Organ of Reference/ Maximum Year(b)	Radionuclide(c)	Maximum Annual Dose (rem) via			Organ of Reference/ Maximum Year(b)	Radionuclide(c)	Maximum Annual Dose (rem) via								
		Inhalation	External(d)	All Pathways			Inhalation	External(d)	All Pathways			Inhalation	External(d)	All Pathways						
Total Body/ Year 1	55Fe	4.6 x 10 ⁻⁷	---	4.6 x 10 ⁻⁷	Total Body/ Year 1	54Mn	9.8 x 10 ⁻¹²	2.0 x 10 ⁻⁸	2.0 x 10 ⁻⁸	Total Body/ Year 1	55Fe	2.1 x 10 ⁻¹⁰	---	2.1 x 10 ⁻¹⁰						
	59Fe	2.5 x 10 ⁻⁸	6.0 x 10 ⁻⁵	6.0 x 10 ⁻⁵		55Fe	3.6 x 10 ⁻⁸	---	3.6 x 10 ⁻⁸		60Co	2.3 x 10 ⁻⁷	1.3 x 10 ⁻³	1.3 x 10 ⁻³						
	58Co	7.8 x 10 ⁻⁸	4.2 x 10 ⁻³	6.4 x 10 ⁻⁴		60Co	3.2 x 10 ⁻⁶	1.8 x 10 ⁻²	1.8 x 10 ⁻²		90Sr + D	6.2 x 10 ⁻⁶	7.1 x 10 ⁻⁶	7.1 x 10 ⁻⁶						
	60Co	1.2 x 10 ⁻⁵	6.9 x 10 ⁻¹	6.9 x 10 ⁻²		65Zn	---	3.6 x 10 ⁻⁸	3.6 x 10 ⁻⁸		129I	1.4 x 10 ⁻¹⁶	---	1.4 x 10 ⁻¹⁶						
	65Zn	---	1.1 x 10 ⁻³	1.1 x 10 ⁻³		90Sr + D	1.0 x 10 ⁻⁵	1.2 x 10 ⁻⁵	2.2 x 10 ⁻⁵		134Cs	1.5 x 10 ⁻¹⁰	7.4 x 10 ⁻⁸	7.4 x 10 ⁻⁸						
	90Sr + D(f)	1.3 x 10 ⁻⁵	1.5 x 10 ⁻⁵	1.5 x 10 ⁻⁵		134Cs	1.2 x 10 ⁻⁷	6.2 x 10 ⁻⁵	6.2 x 10 ⁻²		137Cs + D	2.1 x 10 ⁻⁵	6.7 x 10 ⁻³	6.7 x 10 ⁻³						
	95Zr	8.7 x 10 ⁻⁹	1.4 x 10 ⁻⁵	1.4 x 10 ⁻⁵		137Cs + D	3.3 x 10 ⁻⁵	1.1 x 10 ⁻²	1.1 x 10 ⁻²		Totals	2.7 x 10 ⁻⁵	8.0 x 10 ⁻³	8.0 x 10 ⁻³						
	95Nb	2.5 x 10 ⁻⁹	1.5 x 10 ⁻⁵	1.5 x 10 ⁻⁵		Totals	4.6 x 10 ⁻⁵	2.9 x 10 ⁻²	2.9 x 10 ⁻²		Bone/ Year 1	55Fe	2.2 x 10 ⁻¹⁰	---	2.2 x 10 ⁻¹⁰					
	103Ru	1.8 x 10 ⁻¹⁰	2.1 x 10 ⁻⁵	2.1 x 10 ⁻⁵		Bone/ Year 1	54Mn	---	2.0 x 10 ⁻⁸			2.0 x 10 ⁻⁸	60Co	---	1.3 x 10 ⁻³	1.3 x 10 ⁻³				
	131I	3.6 x 10 ⁻⁷	8.0 x 10 ⁻⁴	8.0 x 10 ⁻⁴			55Fe	3.7 x 10 ⁻⁸	---			3.7 x 10 ⁻⁸	90Sr + D	2.7 x 10 ⁻⁵	7.1 x 10 ⁻⁶	7.1 x 10 ⁻⁶				
	134Cs	3.5 x 10 ⁻⁶	1.8 x 10 ⁻³	1.8 x 10 ⁻³			60Co	---	1.8 x 10 ⁻²			1.8 x 10 ⁻²	134Cs	7.7 x 10 ⁻¹¹	7.4 x 10 ⁻⁸	7.4 x 10 ⁻⁸				
	136Cs	5.1 x 10 ⁻⁹	2.2 x 10 ⁻⁵	2.2 x 10 ⁻⁵			65Zn	---	3.6 x 10 ⁻⁸			3.6 x 10 ⁻⁸	137Cs + D	2.9 x 10 ⁻⁵	6.7 x 10 ⁻³	6.7 x 10 ⁻³				
	137Cs + D	4.1 x 10 ⁻⁵	5.4 x 10 ⁻⁶	5.4 x 10 ⁻⁶			90Sr + D	4.3 x 10 ⁻⁵	1.2 x 10 ⁻⁵			5.5 x 10 ⁻⁵	Totals	5.5 x 10 ⁻⁵	8.0 x 10 ⁻³	8.0 x 10 ⁻³				
	Totals	7.2 x 10 ⁻⁵	8.8 x 10 ⁻²	8.8 x 10 ⁻²			134Cs	6.4 x 10 ⁻⁸	6.2 x 10 ⁻⁵			6.2 x 10 ⁻²	Lung/ Year 1	55Fe	1.4 x 10 ⁻⁸	---	1.4 x 10 ⁻⁸			
Bone/ Year 1	55Fe	4.9 x 10 ⁻⁷	---	4.7 x 10 ⁻⁷	137Cs + D		4.6 x 10 ⁻⁵	1.1 x 10 ⁻²	1.1 x 10 ⁻²	60Co		5.2 x 10 ⁻⁵		1.3 x 10 ⁻³	1.3 x 10 ⁻³					
	59Fe	---	6.0 x 10 ⁻⁵	6.0 x 10 ⁻⁵	Lung/ Year 1	Totals	8.9 x 10 ⁻⁵	2.9 x 10 ⁻²	2.9 x 10 ⁻²	90Sr + D		2.5 x 10 ⁻⁶		7.1 x 10 ⁻⁶	9.6 x 10 ⁻⁶					
	58Co	---	6.4 x 10 ⁻⁴	6.4 x 10 ⁻⁴		54Mn	2.2 x 10 ⁻¹⁰	2.0 x 10 ⁻⁸	2.0 x 10 ⁻⁸	134Cs		4.5 x 10 ⁻¹¹		7.4 x 10 ⁻⁸	7.4 x 10 ⁻⁸					
	60Co	---	6.9 x 10 ⁻²	6.9 x 10 ⁻²		55Fe	2.3 x 10 ⁻⁶	---	2.3 x 10 ⁻⁶	137Cs + D		8.2 x 10 ⁻⁶		6.7 x 10 ⁻³	6.7 x 10 ⁻³					
	65Zn	---	1.1 x 10 ⁻³	1.1 x 10 ⁻³		60Co	7.2 x 10 ⁻⁴	1.8 x 10 ⁻²	1.8 x 10 ⁻²	Totals		6.2 x 10 ⁻⁵		8.0 x 10 ⁻³	8.0 x 10 ⁻³					
	90Sr + D	5.5 x 10 ⁻⁵	1.5 x 10 ⁻⁵	1.5 x 10 ⁻⁵		65Zn	---	2.6 x 10 ⁻⁸	3.6 x 10 ⁻⁸	Thyroid/ Year 1		60Co	---	1.3 x 10 ⁻³	1.3 x 10 ⁻³					
	95Zr	3.3 x 10 ⁻⁸	1.4 x 10 ⁻⁵	1.4 x 10 ⁻⁴		90Sr + D	4.1 x 10 ⁻⁶	1.2 x 10 ⁻⁵	1.6 x 10 ⁻⁵			90Sr + D	---	7.1 x 10 ⁻⁶	7.1 x 10 ⁻⁶					
	95Nb	6.9 x 10 ⁻⁹	1.5 x 10 ⁻⁵	1.5 x 10 ⁻⁵		134Cs	3.7 x 10 ⁻⁸	6.2 x 10 ⁻⁵	6.2 x 10 ⁻⁵			129I	1.2 x 10 ⁻¹³	---	1.2 x 10 ⁻¹³					
	103Ru	3.6 x 10 ⁻¹⁰	2.1 x 10 ⁻⁵	2.1 x 10 ⁻⁵		137Cs + D	1.3 x 10 ⁻⁵	1.1 x 10 ⁻²	1.1 x 10 ⁻²			134Cs	---	7.4 x 10 ⁻⁷	7.4 x 10 ⁻⁷					
	131I	3.9 x 10 ⁻⁷	8.0 x 10 ⁻⁴	8.0 x 10 ⁻⁴	Thyroid/ Year 1	Totals	7.4 x 10 ⁻⁴	2.9 x 10 ⁻²	2.9 x 10 ⁻²			137Cs + D	---	6.7 x 10 ⁻³	6.7 x 10 ⁻³					
	134Cs	1.8 x 10 ⁻⁶	1.8 x 10 ⁻³	1.8 x 10 ⁻³		54Mn	---	2.0 x 10 ⁻⁸	2.0 x 10 ⁻⁸			Totals	1.2 x 10 ⁻¹³	8.0 x 10 ⁻³	8.0 x 10 ⁻³					
	136Cs	2.4 x 10 ⁻⁹	2.2 x 10 ⁻⁵	2.2 x 10 ⁻⁵		60Co	---	1.8 x 10 ⁻²	1.8 x 10 ⁻²			Total Body/ Year 1	60Co	2.3 x 10 ⁻¹¹	1.3 x 10 ⁻⁷	1.3 x 10 ⁻⁷				
	137Cs + D	5.7 x 10 ⁻⁵	1.4 x 10 ⁻²	1.4 x 10 ⁻²		65Zn	---	3.6 x 10 ⁻⁸	3.6 x 10 ⁻⁸				90Sr + D	1.2 x 10 ⁻⁶	1.3 x 10 ⁻⁶	2.4 x 10 ⁻⁶				
	Totals	1.3 x 10 ⁻⁴	8.8 x 10 ⁻²	8.8 x 10 ⁻²		90Sr + D	---	1.2 x 10 ⁻⁵	1.2 x 10 ⁻⁵				129I	1.4 x 10 ⁻¹⁶	---	1.4 x 10 ⁻¹⁶				
Lungs/ Year 1	55Fe	3.0 x 10 ⁻⁵	---	3.0 x 10 ⁻⁵	Continuous Exposure in the Facility Starts 50 Years After Plant Shutdown	Maximum Annual Dose (rem) via				Continuous Exposure in the Facility Starts 100 Years After Plant Shutdown				Maximum Annual Dose (rem) via						
	59Fe	9.8 x 10 ⁻⁷	6.0 x 10 ⁻⁵	6.0 x 10 ⁻⁵		Organ of Reference/ Maximum Year(b)	Radionuclide(c)	Inhalation	External(d)	All Pathways	Organ of Reference/ Maximum Year(b)	Radionuclide(c)	Inhalation	External(d)	All Pathways					
	58Co	9.7 x 10 ⁻⁶	6.4 x 10 ⁻⁴	6.4 x 10 ⁻⁴		Total Body/ Year 1	60Co	1.7 x 10 ⁻⁸	9.4 x 10 ⁻⁵	9.4 x 10 ⁻⁵	60Co	2.3 x 10 ⁻¹¹	1.3 x 10 ⁻⁷	1.3 x 10 ⁻⁷						
	60Co	2.7 x 10 ⁻³	6.9 x 10 ⁻²	7.2 x 10 ⁻²		Total Body/ Year 1	90Sr + D	3.8 x 10 ⁻⁶	4.4 x 10 ⁻⁶	8.2 x 10 ⁻⁶	90Sr + D	1.2 x 10 ⁻⁶	1.3 x 10 ⁻⁶	2.4 x 10 ⁻⁶						
	65Zn	---	1.1 x 10 ⁻³	1.1 x 10 ⁻³			129I	1.4 x 10 ⁻¹⁶	---	1.4 x 10 ⁻¹⁶	129I	1.4 x 10 ⁻¹⁶	---	1.4 x 10 ⁻¹⁶						
	90Sr + D	5.2 x 10 ⁻⁶	1.5 x 10 ⁻⁵	2.0 x 10 ⁻⁵			134Cs	1.8 x 10 ⁻¹³	8.9 x 10 ⁻¹¹	8.9 x 10 ⁻¹¹	137Cs + D	4.1 x 10 ⁻⁶	1.3 x 10 ⁻³	1.3 x 10 ⁻³						
	95Zr	5.0 x 10 ⁻⁷	1.4 x 10 ⁻⁵	1.4 x 10 ⁻⁵			137Cs + D	1.3 x 10 ⁻⁵	4.2 x 10 ⁻³	4.3 x 10 ⁻³	Totals	5.3 x 10 ⁻⁶	1.3 x 10 ⁻³	1.3 x 10 ⁻³						
	95Nb	1.5 x 10 ⁻⁷	1.5 x 10 ⁻⁵	1.5 x 10 ⁻⁵			Totals	1.7 x 10 ⁻⁵	4.3 x 10 ⁻³	4.3 x 10 ⁻³	Bone/ Year 1	60Co	---	1.3 x 10 ⁻⁷	1.3 x 10 ⁻⁷					
	103Ru	3.0 x 10 ⁻⁷	2.1 x 10 ⁻⁵	2.1 x 10 ⁻⁵			Bone/ Year 1	60Co	---	9.4 x 10 ⁻⁵		9.4 x 10 ⁻⁵	90Sr + D	4.3 x 10 ⁻⁶	1.3 x 10 ⁻⁶	6.0 x 10 ⁻⁶				
	131I	4.4 x 10 ⁻⁷	8.0 x 10 ⁻⁴	8.0 x 10 ⁻⁴				90Sr + D	1.6 x 10 ⁻⁵	4.4 x 10 ⁻⁸		2.0 x 10 ⁻⁵	137Cs + D	5.7 x 10 ⁻⁶	1.3 x 10 ⁻³	1.3 x 10 ⁻³				
	134Cs	1.1 x 10 ⁻⁶	1.8 x 10 ⁻³	1.8 x 10 ⁻³				134Cs	9.3 x 10 ⁻¹⁴	8.9 x 10 ⁻¹¹		8.9 x 10 ⁻¹¹	Totals	1.0 x 10 ⁻⁵	1.3 x 10 ⁻³	1.3 x 10 ⁻³				
	136Cs	4.1 x 10 ⁻⁹	2.2 x 10 ⁻⁵	2.2 x 10 ⁻⁵				137Cs + D	1.8 x 10 ⁻⁵	4.3 x 10 ⁻³		4.3 x 10 ⁻³	Lung/ Year 1	60Co	5.1 x 10 ⁻⁹	1.3 x 10 ⁻⁷	1.3 x 10 ⁻⁷			
	137Cs + D	1.6 x 10 ⁻⁵	1.4 x 10 ⁻²	1.4 x 10 ⁻²				Totals	3.4 x 10 ⁻⁵	4.3 x 10 ⁻³		4.3 x 10 ⁻³		90Sr + D	4.5 x 10 ⁻⁷	1.3 x 10 ⁻⁶	1.7 x 10 ⁻⁶			
	Totals	2.8 x 10 ⁻³	8.8 x 10 ⁻²	9.1 x 10 ⁻²				Lung/ Year 1	60Co	3.7 x 10 ⁻⁶	8.4 x 10 ⁻⁵	9.4 x 10 ⁻⁵		137Cs + D	1.6 x 10 ⁻⁶	1.3 x 10 ⁻³	1.3 x 10 ⁻³			
Thyroid/ Year 1	55Fe	---	---	---					90Sr + D	1.5 x 10 ⁻⁶	4.4 x 10 ⁻⁶	5.9 x 10 ⁻⁶		Thyroid/ Year 1	60Co	---	1.3 x 10 ⁻⁷	1.3 x 10 ⁻⁷		
	59Fe	---	6.0 x 10 ⁻⁵	6.0 x 10 ⁻⁵					134Cs	9.3 x 10 ⁻¹⁴	8.9 x 10 ⁻¹¹	8.9 x 10 ⁻¹¹			90Sr + D	---	1.3 x 10 ⁻⁶	1.3 x 10 ⁻⁶		
	58Co	---	6.4 x 10 ⁻⁴	6.4 x 10 ⁻⁴					137Cs + D	1.8 x 10 ⁻⁵	4.3 x 10 ⁻³	4.3 x 10 ⁻³			129I	1.2 x 10 ⁻¹³	---	1.2 x 10 ⁻¹³		
	60Co	---	6.8 x 10 ⁻²	6.9 x 10 ⁻²					Totals	3.4 x 10 ⁻⁵	4.3 x 10 ⁻³	4.3 x 10 ⁻³			137Cs + D	---	1.3 x 10 ⁻³	1.3 x 10 ⁻³		
	65Zn	---	1.1 x 10 ⁻³	1.1 x 10 ⁻³					Thyroid/ Year 1	60Co	---	9.4 x 10 ⁻⁵			9.4 x 10 ⁻⁵	Totals	1.2 x 10 ⁻¹³	1.3 x 10 ⁻³	1.3 x 10 ⁻³	
	90Sr + D	---	1.5 x 10 ⁻⁵	1.5 x 10 ⁻⁵						90Sr + D	---	4.4 x 10 ⁻⁶			4.4 x 10 ⁻⁶	Total Body/ Year 1	60Co	2.3 x 10 ⁻¹¹	1.3 x 10 ⁻⁷	1.3 x 10 ⁻⁷
	95Zr	---	1.4 x 10 ⁻⁵	1.4 x 10 ⁻⁵						129I	1.2 x 10 ⁻¹³	---			1.2 x 10 ⁻¹³		90Sr + D	---	1.3 x 10 ⁻⁶	1.3 x 10 ⁻⁶
	95Nb	---	1.5 x 10 ⁻⁵	1.5 x 10 ⁻⁵						134Cs	---	8.9 x 10 ⁻¹¹			8.9 x 10 ⁻¹¹		129I	1.4 x 10 ⁻¹⁶	---	1.4 x 10 ⁻¹⁶
	103Ru	---	2.1 x 10 ⁻⁵	2.1 x 10 ⁻⁵						137Cs + D	---	4.2 x 10 ⁻³			4.2 x 10 ⁻³		137Cs + D	---	1.3 x 10 ⁻³	1.3 x 10 ⁻³
	131I	2.0 x 10 ⁻⁴	8.0 x 10 ⁻⁴	1.0 x 10 ⁻³			Thyroid/ Year 1	60Co	---	9.4 x 10 ⁻⁵	9.4 x 10 ⁻⁵	Totals	1.2 x 10 ⁻¹³	1.3 x 10 ⁻³	1.3 x 10 ⁻³					
134Cs	---	1.8 x 10 ⁻³	1.8 x 10 ⁻³	Thyroid/ Year 1	90Sr + D	---	4.4 x 10 ⁻⁶	4.4 x 10 ⁻⁶	Total Body/ Year 1	60Co	2.3 x 10 ⁻¹¹	1.3 x 10 ⁻⁷	1.3 x 10 ⁻⁷							
136Cs	---	2.2 x 10 ⁻⁵	2.2 x 10 ⁻⁵	129I	1.2 x 10 ⁻¹³	---	1.2 x 10 ⁻¹³	90Sr + D		---	1.3 x 10 ⁻⁶	1.3 x 10 ⁻⁶								
137Cs + D	---	1.4 x 10 ⁻²	1.4 x 10 ⁻²	134Cs	---	8.9 x 10 ⁻¹¹	8.9 x 10 ⁻¹¹	129I		1.4 x 10 ⁻¹⁶	---	1.4 x 10 ⁻¹⁶								
Totals	2.0 x 10 ⁻⁴	8.8 x 10 ⁻²	8.8 x 10 ⁻²	Totals	1.2 x 10 ⁻¹³	4.3 x 10 ⁻³	4.3 x 10 ⁻³	Totals		1.2 x 10 ⁻¹³	1.3 x 10 ⁻³	1.3 x 10 ⁻³								

(a)Based on the reference radionuclide inventory 5, Table E 2-9 (19)

(b)The year in which the maximum annual dose occurs after the start of continuous exposure

(c)The dose for all radionuclides in Reference Inventory 5 is calculated, but only radionuclides that contribute more than 0.5% to any pathway total are reported in this table

(d)External doses are calculated using the reference room model discussed in Section F 3.1

TABLE F.4-1. Est Dos

TABLE F.4-2. Estimated Maximum Annual Dose to the Maximum-Exposed Individual Residing on the Decommissioned BWR Site at Various Times After Shutdown(a)

Continuous Exposure on the Site Starts at Plant Shutdown						Continuous Exposure on the Site Starts 10 Years After Plant Shutdown						Continuous Exposure on the Site Starts 30 Years After Plant Shutdown																																																																																																																																																																																																																																																																																																																														
Organ of Reference/ Maximum Year(b)	Radionuclide(c)	Maximum Annual Dose (rem) via:				Organ of Reference/ Maximum Year(b)	Radionuclide(c)	Maximum Annual Dose (rem) via:				Organ of Reference/ Maximum Year(b)	Radionuclide(c)	Maximum Annual Dose (rem) via:																																																																																																																																																																																																																																																																																																																												
		Ingestion	Inhalation	External	All Pathways			Ingestion	Inhalation	External	All Pathways			Ingestion	Inhalation	External	All Pathways																																																																																																																																																																																																																																																																																																																									
Total Body/ Year 20	60Co	2.1 x 10-8	---	7.4 x 10-6	7.4 x 10-6	Total Body/ Year 20	60Co	5.6 x 10-9	---	2.0 x 10-6	2.0 x 10-6	Total Body/ Year 20	60Co	4.0 x 10-10	---	1.4 x 10-7	1.4 x 10-7																																																																																																																																																																																																																																																																																																																									
	90Sr + D(e)	4.0 x 10-3	2.1 x 10-8	6.7 x 10-7	4.0 x 10-3		134Cs	1.6 x 10-9	---	4.7 x 10-8	4.9 x 10-8		137Cs + D	5.0 x 10-6	1.4 x 10-10	1.1 x 10-4	1.1 x 10-4	Totals	4.0 x 10-3	2.1 x 10-8	1.2 x 10-4	4.1 x 10-3	Bone/ Year 21	60Co	---	---	6.4 x 10-6	6.4 x 10-6	Bone/ Year 21	60Co	---	---	1.7 x 10-6	1.7 x 10-6	Bone/ Year 21	60Co	---	---	1.2 x 10-7	1.2 x 10-7	90Sr + D	1.5 x 10-2	8.5 x 10-8	6.5 x 10-7	1.5 x 10-2	134Cs	---	---	3.4 x 10-8	3.4 x 10-8	137Cs + D	5.6 x 10-6	2.5 x 10-10	1.1 x 10-4	1.1 x 10-4	Totals	1.5 x 10-2	8.5 x 10-8	1.2 x 10-4	1.5 x 10-2	Lung/ Year 1	51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Lung/ Year 1	54Mn	---	---	6.6 x 10-10	6.6 x 10-10	Lung/ Year 1	60Co	---	1.6 x 10-11	1.7 x 10-6	1.7 x 10-6	54Mn	---	---	2.2 x 10-6	2.2 x 10-6	59Fe	---	---	1.4 x 10-8	1.4 x 10-8	58Co	---	---	2.1 x 10-7	2.1 x 10-7	60Co	---	1.0 x 10-8	9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-8	131I	---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Thyroid/ Year 1	54Mn	---	---	6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1	60Co	---	---	1.7 x 10-6	1.7 x 10-6	54Mn	---	---	2.2 x 10-6	2.2 x 10-6	59Fe	---	---	1.4 x 10-8	1.4 x 10-8	58Co	---	---	2.1 x 10-7	2.1 x 10-7	60Co	---	---	9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.														
	134Cs	1.6 x 10-9	---	4.7 x 10-8	4.9 x 10-8		137Cs + D	5.0 x 10-6	1.4 x 10-10	1.1 x 10-4	1.1 x 10-4		Totals	4.0 x 10-3	2.1 x 10-8	1.2 x 10-4	4.1 x 10-3	Bone/ Year 21	60Co	---	---	6.4 x 10-6		6.4 x 10-6	Bone/ Year 21	60Co	---	---		1.7 x 10-6	1.7 x 10-6	Bone/ Year 21	60Co	---		---	1.2 x 10-7	1.2 x 10-7	90Sr + D	1.5 x 10-2	8.5 x 10-8	6.5 x 10-7	1.5 x 10-2	134Cs	---	---	3.4 x 10-8	3.4 x 10-8	137Cs + D	5.6 x 10-6	2.5 x 10-10	1.1 x 10-4	1.1 x 10-4	Totals	1.5 x 10-2	8.5 x 10-8	1.2 x 10-4	1.5 x 10-2	Lung/ Year 1	51Cr		---	---	6.4 x 10-10	6.4 x 10-10	Lung/ Year 1		54Mn	---	---	6.6 x 10-10	6.6 x 10-10		Lung/ Year 1	60Co	---	1.6 x 10-11	1.7 x 10-6	1.7 x 10-6	54Mn	---	---	2.2 x 10-6	2.2 x 10-6	59Fe	---	---	1.4 x 10-8	1.4 x 10-8	58Co	---	---	2.1 x 10-7	2.1 x 10-7	60Co	---	1.0 x 10-8	9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-8	131I	---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4		3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---		6.4 x 10-10	6.4 x 10-10	Thyroid/ Year 1	54Mn	---		---	6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1	60Co	---	---	1.7 x 10-6	1.7 x 10-6	54Mn	---	---	2.2 x 10-6	2.2 x 10-6	59Fe	---	---	1.4 x 10-8	1.4 x 10-8	58Co	---	---	2.1 x 10-7	2.1 x 10-7	60Co	---	---	9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.										
	137Cs + D	5.0 x 10-6	1.4 x 10-10	1.1 x 10-4	1.1 x 10-4		Totals	4.0 x 10-3	2.1 x 10-8	1.2 x 10-4	4.1 x 10-3		Bone/ Year 21	60Co	---	---	6.4 x 10-6		6.4 x 10-6	Bone/ Year 21	60Co	---		---		1.7 x 10-6	1.7 x 10-6	Bone/ Year 21		60Co	---		---	1.2 x 10-7		1.2 x 10-7	90Sr + D	1.5 x 10-2	8.5 x 10-8	6.5 x 10-7	1.5 x 10-2	134Cs	---	---	3.4 x 10-8	3.4 x 10-8	137Cs + D	5.6 x 10-6	2.5 x 10-10	1.1 x 10-4	1.1 x 10-4	Totals	1.5 x 10-2	8.5 x 10-8	1.2 x 10-4	1.5 x 10-2	Lung/ Year 1	51Cr		---		---	6.4 x 10-10	6.4 x 10-10	Lung/ Year 1			54Mn	---	---	6.6 x 10-10	6.6 x 10-10			Lung/ Year 1	60Co	---	1.6 x 10-11	1.7 x 10-6	1.7 x 10-6	54Mn	---	---	2.2 x 10-6	2.2 x 10-6	59Fe	---	---	1.4 x 10-8	1.4 x 10-8	58Co	---	---	2.1 x 10-7	2.1 x 10-7	60Co	---	1.0 x 10-8	9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-8	131I	---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8		3.0 x 10-4		3.0 x 10-4	Thyroid/ Year 1	51Cr		---	---		6.4 x 10-10	6.4 x 10-10		Thyroid/ Year 1	54Mn	---		---	6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1	60Co	---	---	1.7 x 10-6	1.7 x 10-6	54Mn	---	---	2.2 x 10-6	2.2 x 10-6	59Fe	---	---	1.4 x 10-8	1.4 x 10-8	58Co	---	---	2.1 x 10-7	2.1 x 10-7	60Co	---	---	9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						
	Totals	4.0 x 10-3	2.1 x 10-8	1.2 x 10-4	4.1 x 10-3		Bone/ Year 21	60Co	---	---	6.4 x 10-6			6.4 x 10-6	Bone/ Year 21	60Co	---		---		1.7 x 10-6	1.7 x 10-6		Bone/ Year 21		60Co	---			---	1.2 x 10-7		1.2 x 10-7	90Sr + D		1.5 x 10-2	8.5 x 10-8	6.5 x 10-7	1.5 x 10-2	134Cs	---	---	3.4 x 10-8	3.4 x 10-8	137Cs + D	5.6 x 10-6	2.5 x 10-10	1.1 x 10-4	1.1 x 10-4	Totals	1.5 x 10-2	8.5 x 10-8	1.2 x 10-4	1.5 x 10-2	Lung/ Year 1	51Cr		---		---		6.4 x 10-10	6.4 x 10-10	Lung/ Year 1				54Mn	---	---	6.6 x 10-10	6.6 x 10-10				Lung/ Year 1	60Co	---	1.6 x 10-11	1.7 x 10-6	1.7 x 10-6	54Mn	---	---	2.2 x 10-6	2.2 x 10-6	59Fe	---	---	1.4 x 10-8	1.4 x 10-8	58Co	---	---	2.1 x 10-7	2.1 x 10-7	60Co	---	1.0 x 10-8	9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-8	131I	---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7		1.2 x 10-8		3.0 x 10-4		3.0 x 10-4		Thyroid/ Year 1	51Cr		---	---			6.4 x 10-10	6.4 x 10-10		Thyroid/ Year 1	54Mn	---		---	6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1	60Co	---	---	1.7 x 10-6	1.7 x 10-6	54Mn	---	---	2.2 x 10-6	2.2 x 10-6	59Fe	---	---	1.4 x 10-8	1.4 x 10-8	58Co	---	---	2.1 x 10-7	2.1 x 10-7	60Co	---	---	9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.		
Bone/ Year 21	60Co	---	---	6.4 x 10-6	6.4 x 10-6	Bone/ Year 21		60Co	---	---	1.7 x 10-6	1.7 x 10-6		Bone/ Year 21		60Co	---		---		1.2 x 10-7	1.2 x 10-7																																																																																																																																																																																																																																																																																																																				
	90Sr + D	1.5 x 10-2	8.5 x 10-8	6.5 x 10-7	1.5 x 10-2			134Cs	---	---	3.4 x 10-8	3.4 x 10-8				137Cs + D	5.6 x 10-6		2.5 x 10-10		1.1 x 10-4	1.1 x 10-4	Totals			1.5 x 10-2	8.5 x 10-8		1.2 x 10-4	1.5 x 10-2	Lung/ Year 1		51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Lung/ Year 1	54Mn	---	---	6.6 x 10-10	6.6 x 10-10	Lung/ Year 1	60Co	---	1.6 x 10-11	1.7 x 10-6	1.7 x 10-6	54Mn	---	---	2.2 x 10-6	2.2 x 10-6		59Fe		---		---		1.4 x 10-8	1.4 x 10-8					58Co	---	---	2.1 x 10-7	2.1 x 10-7					60Co	---	1.0 x 10-8	9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-8	131I	---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Thyroid/ Year 1	54Mn	---	---	6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1	60Co	---	---	1.7 x 10-6		1.7 x 10-6		54Mn		---			---		2.2 x 10-6	2.2 x 10-6			59Fe	---			---	1.4 x 10-8		1.4 x 10-8	58Co	---		---	2.1 x 10-7	2.1 x 10-7	60Co	---	---	9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																			
	134Cs	---	---	3.4 x 10-8	3.4 x 10-8			137Cs + D	5.6 x 10-6	2.5 x 10-10	1.1 x 10-4	1.1 x 10-4				Totals	1.5 x 10-2	8.5 x 10-8	1.2 x 10-4		1.5 x 10-2	Lung/ Year 1	51Cr		---	---	6.4 x 10-10		6.4 x 10-10	Lung/ Year 1		54Mn	---	---	6.6 x 10-10	6.6 x 10-10	Lung/ Year 1		60Co	---	1.6 x 10-11	1.7 x 10-6	1.7 x 10-6		54Mn	---	---	2.2 x 10-6	2.2 x 10-6	59Fe	---	---	1.4 x 10-8	1.4 x 10-8		58Co		---		---		2.1 x 10-7	2.1 x 10-7					60Co	---	1.0 x 10-8	9.0 x 10-5	9.0 x 10-5					65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-8	131I	---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10		6.4 x 10-10	Thyroid/ Year 1	54Mn	---	---		6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1	60Co	---		---	1.7 x 10-6	1.7 x 10-6	54Mn		---		---		2.2 x 10-6			2.2 x 10-6		59Fe	---			---	1.4 x 10-8			1.4 x 10-8	58Co		---	---	2.1 x 10-7		2.1 x 10-7	60Co	---	---	9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																					
	137Cs + D	5.6 x 10-6	2.5 x 10-10	1.1 x 10-4	1.1 x 10-4			Totals	1.5 x 10-2	8.5 x 10-8	1.2 x 10-4	1.5 x 10-2	Lung/ Year 1			51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Lung/ Year 1		54Mn		---	---	6.6 x 10-10	6.6 x 10-10	Lung/ Year 1			60Co	---	1.6 x 10-11	1.7 x 10-6	1.7 x 10-6			54Mn	---	---	2.2 x 10-6	2.2 x 10-6		59Fe	---	---	1.4 x 10-8	1.4 x 10-8	58Co	---	---	2.1 x 10-7	2.1 x 10-7		60Co		---		1.0 x 10-8		9.0 x 10-5	9.0 x 10-5					65Zn	---	---	1.4 x 10-7	1.4 x 10-7					90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-8	131I	---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10		6.4 x 10-10	Thyroid/ Year 1	54Mn	---		---		6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1		60Co	---		---	1.7 x 10-6		1.7 x 10-6	54Mn	---	---		2.2 x 10-6		2.2 x 10-6		59Fe			---		---	1.4 x 10-8			1.4 x 10-8	58Co			---	---		2.1 x 10-7	2.1 x 10-7	60Co		---	---	9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																							
	Totals	1.5 x 10-2	8.5 x 10-8	1.2 x 10-4	1.5 x 10-2		Lung/ Year 1	51Cr	---	---	6.4 x 10-10	6.4 x 10-10			Lung/ Year 1	54Mn	---	---	6.6 x 10-10	6.6 x 10-10			Lung/ Year 1	60Co	---	1.6 x 10-11	1.7 x 10-6	1.7 x 10-6				54Mn	---	---	2.2 x 10-6	2.2 x 10-6			59Fe	---	---	1.4 x 10-8	1.4 x 10-8		58Co	---	---	2.1 x 10-7	2.1 x 10-7	60Co	---	1.0 x 10-8	9.0 x 10-5	9.0 x 10-5		65Zn		---		---		1.4 x 10-7	1.4 x 10-7					90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8					95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-8	131I	---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10		6.4 x 10-10	Thyroid/ Year 1	54Mn	---		---		6.6 x 10-10	6.6 x 10-10		Thyroid/ Year 1		60Co	---			---	1.7 x 10-6		1.7 x 10-6	54Mn		---	---	2.2 x 10-6	2.2 x 10-6		59Fe		---		---			1.4 x 10-8		1.4 x 10-8	58Co			---	---			2.1 x 10-7	2.1 x 10-7		60Co	---	---		9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																									
Lung/ Year 1	51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Lung/ Year 1		54Mn	---	---	6.6 x 10-10	6.6 x 10-10		Lung/ Year 1		60Co	---	1.6 x 10-11	1.7 x 10-6	1.7 x 10-6																																																																																																																																																																																																																																																																																																																						
	54Mn	---	---	2.2 x 10-6	2.2 x 10-6			59Fe	---	---	1.4 x 10-8	1.4 x 10-8				58Co	---	---	2.1 x 10-7	2.1 x 10-7				60Co	---	1.0 x 10-8	9.0 x 10-5	9.0 x 10-5				65Zn	---	---	1.4 x 10-7	1.4 x 10-7			90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8		95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8		110mAg		---		---	8.5 x 10-7	8.5 x 10-8	131I				---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8	---				2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Thyroid/ Year 1	54Mn	---	---	6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1	60Co	---	---	1.7 x 10-6	1.7 x 10-6	54Mn	---	---	2.2 x 10-6	2.2 x 10-6		59Fe	---	---	1.4 x 10-8		1.4 x 10-8		58Co	---		---		2.1 x 10-7	2.1 x 10-7				60Co	---			---	9.0 x 10-5		9.0 x 10-5	65Zn		---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---		---		1.6 x 10-8	1.6 x 10-8		95Zr		---	---	3.6 x 10-8		3.6 x 10-8	95Nb			---	---		2.2 x 10-8	2.2 x 10-8	110mAg		---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																
	59Fe	---	---	1.4 x 10-8	1.4 x 10-8			58Co	---	---	2.1 x 10-7	2.1 x 10-7				60Co	---	1.0 x 10-8	9.0 x 10-5	9.0 x 10-5				65Zn	---	---	1.4 x 10-7	1.4 x 10-7				90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8			95Zr	---	---	3.6 x 10-8	3.6 x 10-8		95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-8		131I		---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8			---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals			8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Thyroid/ Year 1	54Mn		---	---	6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1		60Co	---	---	1.7 x 10-6	1.7 x 10-6		54Mn	---	---	2.2 x 10-6	2.2 x 10-6	59Fe	---	---	1.4 x 10-8	1.4 x 10-8		58Co	---	---	2.1 x 10-7		2.1 x 10-7		60Co	---		---		9.0 x 10-5	9.0 x 10-5				65Zn	---			---	1.4 x 10-7		1.4 x 10-7	90Sr + D		---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8		3.6 x 10-8	95Nb		---	---	2.2 x 10-8	2.2 x 10-8	110mAg		---	---	8.5 x 10-7		8.5 x 10-7	131I		1.7 x 10-6	1.4 x 10-10	1.3 x 10-7		1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																								
	58Co	---	---	2.1 x 10-7	2.1 x 10-7			60Co	---	1.0 x 10-8	9.0 x 10-5	9.0 x 10-5				65Zn	---	---	1.4 x 10-7	1.4 x 10-7				90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8				95Zr	---	---	3.6 x 10-8	3.6 x 10-8			95Nb	---	---	2.2 x 10-8	2.2 x 10-8		110mAg	---	---	8.5 x 10-7	8.5 x 10-8	131I	---	---	1.3 x 10-7	1.3 x 10-7		134Cs	5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10		1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---		6.4 x 10-10	6.4 x 10-10	Thyroid/ Year 1	54Mn		---	---	6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1		60Co		---	---	1.7 x 10-6	1.7 x 10-6			54Mn	---	---	2.2 x 10-6	2.2 x 10-6		59Fe	---	---	1.4 x 10-8	1.4 x 10-8	58Co	---	---	2.1 x 10-7	2.1 x 10-7		60Co	---	---	9.0 x 10-5		9.0 x 10-5		65Zn	---		---		1.4 x 10-7	1.4 x 10-7				90Sr + D	---			---	1.6 x 10-8		1.6 x 10-8	95Zr		---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---		---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I		8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---		---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																
	60Co	---	1.0 x 10-8	9.0 x 10-5	9.0 x 10-5			65Zn	---	---	1.4 x 10-7	1.4 x 10-7				90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8				95Zr	---	---	3.6 x 10-8	3.6 x 10-8				95Nb	---	---	2.2 x 10-8	2.2 x 10-8			110mAg	---	---	8.5 x 10-7	8.5 x 10-8		131I	---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Thyroid/ Year 1		54Mn	---	---	6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1		60Co		---	---	1.7 x 10-6	1.7 x 10-6			54Mn		---	---	2.2 x 10-6	2.2 x 10-6			59Fe	---	---	1.4 x 10-8	1.4 x 10-8		58Co	---	---	2.1 x 10-7	2.1 x 10-7	60Co	---	---	9.0 x 10-5	9.0 x 10-5		65Zn	---	---	1.4 x 10-7		1.4 x 10-7		90Sr + D	---		---		1.6 x 10-8	1.6 x 10-8				95Zr	---			---	3.6 x 10-8		3.6 x 10-8	95Nb		---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																								
	65Zn	---	---	1.4 x 10-7	1.4 x 10-7			90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8				95Zr	---	---	3.6 x 10-8	3.6 x 10-8				95Nb	---	---	2.2 x 10-8	2.2 x 10-8				110mAg	---	---	8.5 x 10-7	8.5 x 10-8			131I	---	---	1.3 x 10-7	1.3 x 10-7		134Cs	5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10		6.4 x 10-10	Thyroid/ Year 1	54Mn	---	---			6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1	60Co	---			---		1.7 x 10-6	1.7 x 10-6	54Mn	---			---		2.2 x 10-6	2.2 x 10-6	59Fe	---			---	1.4 x 10-8	1.4 x 10-8	58Co	---		---	2.1 x 10-7	2.1 x 10-7	60Co	---	---	9.0 x 10-5	9.0 x 10-5	65Zn	---		---	1.4 x 10-7	1.4 x 10-7	90Sr + D		---		---	1.6 x 10-8		1.6 x 10-8		95Zr	---				---	3.6 x 10-8			3.6 x 10-8	95Nb		---	---		2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																										
	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8			95Zr	---	---	3.6 x 10-8	3.6 x 10-8				95Nb	---	---	2.2 x 10-8	2.2 x 10-8				110mAg	---	---	8.5 x 10-7	8.5 x 10-8			131I	---	---	1.3 x 10-7	1.3 x 10-7	134Cs		5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Thyroid/ Year 1	54Mn		---	---	6.6 x 10-10	6.6 x 10-10		Thyroid/ Year 1		60Co	---	---			1.7 x 10-6	1.7 x 10-6		54Mn	---			---		2.2 x 10-6	2.2 x 10-6	59Fe	---			---		1.4 x 10-8	1.4 x 10-8	58Co	---			---	2.1 x 10-7	2.1 x 10-7	60Co	---		---	9.0 x 10-5	9.0 x 10-5	65Zn	---	---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---		---	1.6 x 10-8	1.6 x 10-8	95Zr		---		---	3.6 x 10-8		3.6 x 10-8		95Nb	---	---			2.2 x 10-8	2.2 x 10-8		110mAg	---	---		8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																		
	95Zr	---	---	3.6 x 10-8	3.6 x 10-8			95Nb	---	---	2.2 x 10-8	2.2 x 10-8				110mAg	---	---	8.5 x 10-7	8.5 x 10-8		131I		---	---	1.3 x 10-7	1.3 x 10-7	134Cs		5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Thyroid/ Year 1	54Mn		---	---	6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1		60Co		---	---	1.7 x 10-6	1.7 x 10-6				54Mn	---	---			2.2 x 10-6	2.2 x 10-6		59Fe	---			---		1.4 x 10-8	1.4 x 10-8	58Co	---			---		2.1 x 10-7	2.1 x 10-7	60Co	---			---	9.0 x 10-5	9.0 x 10-5	65Zn	---		---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---		---	3.6 x 10-8	3.6 x 10-8	95Nb		---		---	2.2 x 10-8	2.2 x 10-8	110mAg		---	---	8.5 x 10-7		8.5 x 10-7	131I	1.7 x 10-6		1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																										
	95Nb	---	---	2.2 x 10-8	2.2 x 10-8			110mAg	---	---	8.5 x 10-7	8.5 x 10-8	131I			---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8	---		2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Thyroid/ Year 1	54Mn		---	---	6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1		60Co		---	---	1.7 x 10-6	1.7 x 10-6			54Mn		---	---	2.2 x 10-6	2.2 x 10-6				59Fe	---	---			1.4 x 10-8	1.4 x 10-8		58Co	---			---		2.1 x 10-7	2.1 x 10-7	60Co	---			---		9.0 x 10-5	9.0 x 10-5	65Zn	---			---	1.4 x 10-7	1.4 x 10-7	90Sr + D	---		---	1.6 x 10-8	1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---		---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---		8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I		8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																																		
	110mAg	---	---	8.5 x 10-7	8.5 x 10-8		131I	---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8		---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Thyroid/ Year 1	54Mn		---	---	6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1		60Co		---	---	1.7 x 10-6	1.7 x 10-6			54Mn		---	---	2.2 x 10-6	2.2 x 10-6			59Fe		---	---	1.4 x 10-8	1.4 x 10-8				58Co	---	---			2.1 x 10-7	2.1 x 10-7		60Co	---			---		9.0 x 10-5	9.0 x 10-5	65Zn	---			---		1.4 x 10-7	1.4 x 10-7	90Sr + D	---			---	1.6 x 10-8	1.6 x 10-8	95Zr	---		---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																																										
131I	---	---	1.3 x 10-7	1.3 x 10-7	134Cs	5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10	6.4 x 10-10	Thyroid/ Year 1	54Mn		---	---	6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1		60Co		---	---	1.7 x 10-6	1.7 x 10-6			54Mn		---	---	2.2 x 10-6	2.2 x 10-6			59Fe		---	---	1.4 x 10-8	1.4 x 10-8			58Co		---	---	2.1 x 10-7	2.1 x 10-7				60Co	---	---			9.0 x 10-5	9.0 x 10-5		65Zn	---			---		1.4 x 10-7	1.4 x 10-7	90Sr + D	---			---		1.6 x 10-8	1.6 x 10-8	95Zr	---			---	3.6 x 10-8	3.6 x 10-8	95Nb	---		---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																																															
134Cs	5.8 x 10-8	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10		6.4 x 10-10	Thyroid/ Year 1	54Mn	---	---		6.6 x 10-10		6.6 x 10-10	Thyroid/ Year 1	60Co	---			---		1.7 x 10-6	1.7 x 10-6	54Mn	---			---		2.2 x 10-6	2.2 x 10-6	59Fe	---			---		1.4 x 10-8	1.4 x 10-8	58Co	---			---		2.1 x 10-7	2.1 x 10-7	60Co	---				---	9.0 x 10-5	9.0 x 10-5			65Zn	---		---	1.4 x 10-7			1.4 x 10-7		90Sr + D	---	---	1.6 x 10-8			1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8		95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																																																				
137Cs + D	7.5 x 10-7	7.5 x 10-10	1.8 x 10-4	7.5 x 10-7	Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10		6.4 x 10-10	Thyroid/ Year 1	54Mn	---		---		6.6 x 10-10	6.6 x 10-10	Thyroid/ Year 1		60Co		---		---	1.7 x 10-6			1.7 x 10-6		54Mn	---	---	2.2 x 10-6			2.2 x 10-6		59Fe	---	---	1.4 x 10-8			1.4 x 10-8		58Co	---	---	2.1 x 10-7			2.1 x 10-7		60Co	---	---	9.0 x 10-5				9.0 x 10-5	65Zn	---			---	1.4 x 10-7		1.4 x 10-7	90Sr + D			---	---	1.6 x 10-8	1.6 x 10-8	95Zr	---		---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																																																									
Totals	8.1 x 10-7	1.2 x 10-8	3.0 x 10-4	3.0 x 10-4	Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10		6.4 x 10-10	Thyroid/ Year 1	54Mn	---		---		6.6 x 10-10	6.6 x 10-10		Thyroid/ Year 1		60Co	---			---		1.7 x 10-6		1.7 x 10-6	54Mn			---		---	2.2 x 10-6	2.2 x 10-6	59Fe			---		---	1.4 x 10-8	1.4 x 10-8	58Co			---		---	2.1 x 10-7	2.1 x 10-7	60Co			---		---	9.0 x 10-5	9.0 x 10-5	65Zn				---	---	1.4 x 10-7		1.4 x 10-7	90Sr + D	---		---	1.6 x 10-8		1.6 x 10-8	95Zr	---	---	3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																																																														
Thyroid/ Year 1	51Cr	---	---	6.4 x 10-10		6.4 x 10-10	Thyroid/ Year 1	54Mn	---		---		6.6 x 10-10	6.6 x 10-10		Thyroid/ Year 1		60Co	---				---	1.7 x 10-6			1.7 x 10-6																																																																																																																																																																																																																																																																																																															
	54Mn	---	---	2.2 x 10-6		2.2 x 10-6		59Fe	---		---		1.4 x 10-8	1.4 x 10-8				58Co	---				---	2.1 x 10-7			2.1 x 10-7		60Co		---	---			9.0 x 10-5		9.0 x 10-5	65Zn	---	---			1.4 x 10-7		1.4 x 10-7	90Sr + D	---	---			1.6 x 10-8		1.6 x 10-8	95Zr	---	---			3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8	2.2 x 10-8		110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																																																																																					
	59Fe	---	---	1.4 x 10-8		1.4 x 10-8		58Co	---		---		2.1 x 10-7	2.1 x 10-7				60Co	---				---	9.0 x 10-5			9.0 x 10-5		65Zn		---	---			1.4 x 10-7		1.4 x 10-7	90Sr + D	---	---			1.6 x 10-8		1.6 x 10-8	95Zr	---	---			3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8		2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																																																																																													
	58Co	---	---	2.1 x 10-7		2.1 x 10-7		60Co	---		---		9.0 x 10-5	9.0 x 10-5				65Zn	---				---	1.4 x 10-7			1.4 x 10-7		90Sr + D		---	---			1.6 x 10-8		1.6 x 10-8	95Zr	---	---			3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8		2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																																																																																																					
	60Co	---	---	9.0 x 10-5		9.0 x 10-5		65Zn	---		---		1.4 x 10-7	1.4 x 10-7				90Sr + D	---				---	1.6 x 10-8			1.6 x 10-8		95Zr		---	---			3.6 x 10-8	3.6 x 10-8	95Nb	---	---	2.2 x 10-8		2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																																																																																																													
	65Zn	---	---	1.4 x 10-7		1.4 x 10-7		90Sr + D	---		---		1.6 x 10-8	1.6 x 10-8				95Zr	---				---	3.6 x 10-8			3.6 x 10-8	95Nb	---		---	2.2 x 10-8		2.2 x 10-8	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I	1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																																																																																																																					
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	95Nb	---	---	2.2 x 10-8		2.2 x 10-8		110mAg	---	---	8.5 x 10-7		8.5 x 10-7	131I	1.7 x 10-6		1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I		8.4 x 10-9	---	1.5 x 10-8	2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																																																																																																																																													
	110mAg	---	---	8.5 x 10-7	8.5 x 10-7	131I		1.7 x 10-6	1.4 x 10-10	1.3 x 10-7	1.8 x 10-6	133I	8.4 x 10-9	---	1.5 x 10-8		2.3 x 10-8	134Cs	---	---	2.8 x 10-5	2.8 x 10-5	137Cs + D	---	---	1.8 x 10-4	3.2 x 10-8	Totals	1.7 x 10-6	1.4 x 10-10	3.0 x 10-4	3.0 x 10-4	(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(a)Based on reference radionuclide inventory 6, Table E.2-10, from 1975 reported release information.(20)						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(b)The year in which the maximum annual dose occurs after the start of continuous exposure.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(c)The dose from all radionuclides in Reference Radionuclide Inventory 6 is calculated, but only select radionuclide contributors to any pathway total dose are reported in this table.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(d)Indicates a contribution to the total pathway dose of less than 0.5%.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.						(e)+ D means plus daughter product radionuclides.																																																																																																																																																																																																																					
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APPENDIX G

DECOMMISSIONING METHODS

Methods and equipment that can be used to accomplish various decommissioning activities at nuclear facilities such as the reference BWR are discussed in this appendix. These activities, in order of their presentation here, are:

- decontamination
- equipment disassembly
- radioactive waste packaging and shipping
- contamination control
- quality assurance
- environmental surveillance.

G.1 DECONTAMINATION

Three basic methods can be used to remove radioactive materials from contaminated surfaces. These are: 1) dissolution of the surface film containing the radionuclides, 2) physical cleaning of the surface, and 3) physical removal of the contaminated structural material. The various techniques used for each of these methods are discussed below.

G.1.1 Surface Film Dissolution

Two methods can be used to dissolve surface films containing radionuclide contamination: 1) chemical and 2) electrochemical. The chemical method is generally used to decontaminate the internal and (in some cases) external surfaces of in-place piping and equipment. The electrochemical method (electropolishing) can sometimes be used in situ for internal decontamination, but it is more commonly applied to disassembled or segmented piping and equipment.

G.1.1.1 Chemical Decontamination

Several chemical decontamination methods are available for radioactive contamination removal from surfaces. Some are designed for piping loops where the chemicals can be recirculated until the desired degree of decontamination is obtained. Others are designed to complete the decontamination in one pass,

so that they can be used where recirculation is impractical. Selected chemical decontamination methods for BWR systems, with associated advantages and disadvantages, are presented in Table G.1-1. More detailed information on chemical decontamination is available in References 1 through 4. System decontamination and draining and radioactive liquid disposal during decommissioning of the reference BWR are discussed in detail in Section H.5 of Appendix H.

TABLE G.1-1. Selected Chemical Decontamination Methods for BWR Systems

Methods	Advantages	Disadvantages
<u>Recirculatory (Loop-Type)</u>		
Alkaline permanganate, citric/oxalic acid (citrox)	Demonstrated effective solutions, <i>nondamaging to components</i> , decontamination factor (DF) ~10 to 20	Generally requires repeated cycles to obtain good decontamination. Generates large volumes of waste solutions requiring processing. Large tankage volume required for chemical preparation and handling
Ethylenediaminetetraacetic acid (EDTA) and citrox, 5% solution	Only one cycle necessary. Contaminants can be removed by a side-stream through an online cation resin bed, if desired. Chemicals can be added directly to system water. DF ~10 to 20 at 80 to 100°C	Recirculation time ~2 days, minimum. Potentially damaging to system components. Produces large volumes of waste solutions requiring processing and/or radioactive ion-exchange resins requiring disposal. Additional R&D needed to determine optimum conditions
EDTA, nitrilotriacetic acid (NTA), and reducing agent(s)	Produces relatively small volumes of waste solutions requiring processing. DF ~2 to 10	Recirculation time from 1 to 7 days. Requires elevated temperature (~150 to 180°C) to be effective.
<u>Single-pass</u>		
Ammonium citrate and ammonium oxalate, 4 to 10% solution	No preconditioning required, DF ~5 to 15 at 80°C	Attacks carbon steel, requiring waste solutions to be neutralized prior to long-term storage. Requires up to 20 hours for maximum effectiveness. Produces large volumes of waste solutions
Inhibited phosphoric acid (H ₃ PO ₄), ≤10% solution	Demonstrated fast (10 to 20 min) and effective method, DF ~5 to 10 at 80°C	Produces large volumes of waste solutions. Possible redeposition of contaminants. Most effective on carbon steel.
Hydrochloric acid (HCl), ~10% solution	Fast, effective method for decontamination of carbon steel (No reliable data on DF)	Incompatible with stainless steel. Possible redeposition of contaminants. Potentially more hazardous than other decontamination chemicals

Isolatable equipment and piping sections in the reference BWR that are highly contaminated but not amenable to system-wide internal decontamination are postulated to be chemically decontaminated using mobile, shielded, decontamination units. Five such units, at an estimated cost of \$20,000 each, are assumed to be used.

Each unit is used for both chemical recirculation and rinse-water flushing and, after use, is itself decontaminated by backflushing. Each unit consists of two parts: 1) a remotely controlled operating unit containing a recirculating pump, an expansion tank, a thermostatically controlled electric heater, and a valve manifold; and 2) a control unit, connected by an electrical cable to the operating unit, located an appropriate distance away from the operating unit for ALARA considerations.

After isolation, the item being decontaminated is connected (with short-run flexible hoses) to the mobile decontamination unit to form a recirculation loop. Chemicals are injected into the loop through the valve manifold at the unit, valving is aligned, and recirculation is begun. The required solution temperature is automatically maintained by the unit's electric heater during the entire recirculation period. A minimum of operator attention is required during recirculation.

G.1.1.2 Electropolishing

Electropolishing is an electrochemical process using electrolytic dissolution to produce a smooth, polished surface. It has been demonstrated effective in decontaminating a wide variety of equipment items with varying alloy compositions.⁽⁵⁾

The electropolishing process, shown schematically in Figure G.1-1, produces a two-fold result: 1) surface material is dissolved and removed (together with any contaminants) and 2) the new surface of the material is smoothed, minimizing recontamination from the electrolytic solution and allowing a simple water rinse (the final process step) to produce a contamination-free surface. Electropolishing offers a number of potential advantages over other decontamination processes:

- It can be used on a wide variety of metals and their alloys, equipment geometries, and types of contamination.
- It reduces personnel radiation exposure and decontamination costs.
- It is rapid and effective.
- It can significantly reduce the amount of solid waste requiring expensive controlled storage, transportation, and disposal.
- It permits reclamation, reuse, and/or recycling of valuable materials.

FIGURE G.1-1. Schematic Diagram of an Electropolishing Cell

For decontamination of disassembled (or segmented) piping and equipment, an electropolishing station is required. It consists of several open-top tanks: one contains the electrolyte (typically phosphoric and/or sulfuric acid) and appropriate fixtures, and the others are used for rinsing. A large constant-current, constant-voltage, DC power supply, capable of producing up to 10,000 A at about 10 V, is required to provide a current density of 500 to

1500 A/m² at the anode surface. A recirculation system constantly mixes the electrolyte and also provides on-line filtration and ion exchange to remove suspended solids and dissolved metal ions from the electrolyte. When the capacity of the ion-exchange column is exhausted, it is backflushed and regenerated, with the resulting waste solutions going to the in-plant liquid radwaste system for processing. An air sweep must be maintained over the electrolyte tank to sufficiently remove and dilute the hydrogen gas evolved by electrolysis during the cleaning process. The electropolisher is designed to automatically shut down if the ventilation system is inoperative, in order to prevent the accumulation of explosive concentrations of hydrogen gas.

Electropolishing can also be used for in-situ decontamination of the internal surfaces of cylindrical tanks. The electrolyte is sprayed onto the tank wall from nozzles mounted on rotating arms. A power supply provides current to these arms and, via the electrolyte stream, to the tank wall. Contaminated electrolyte is collected at the tank bottom and returned to an electrolyte handling system for cooling, filtration, and radiation monitoring. When the electrolyte becomes too contaminated, it is cleaned by processing it through the ion-exchange/filtration system of the electropolishing station and is then reused.

G.1.2 Physical Cleaning

Removal of smearable radioactive contamination from surfaces such as walls, floors, and tank exteriors can be accomplished using a variety of techniques. For small quantities of loose contamination on floors, vacuuming or simple sweeping is often effective. For more tenacious contaminants, various cleansing compounds are utilized in combination with handwiping and scrubbing techniques. Several proprietary decontamination solutions are available. Ordinary household detergents are quite effective but produce sizable quantities of waste water that may require special processing. Aerosol-type foaming cleansers are effective and eliminate the wastewater problem, but their use produces sizable quantities of contaminated wiping material.

Trichloroethylene, Freon-113, and other solvents are effective degreasing agents that can be used to decontaminate equipment surfaces. Organic solvents

have the advantage of not being corrosive to equipment and electrical connections. However, their use generates contaminated organic solutions that must be processed. Use of organic solutions can cause degradation of plastic materials and can produce sizable quantities of contaminated wiping materials.

Variable-pressure, high- or low-velocity liquid jets are quite effective for some types of surface decontamination work.⁽⁶⁾ These devices are hand-held lances that can be operated by one man and can produce pressures up to about 30 MPa. Where effective, the liquid jet is a very rapid decontamination method. However, it produces significant volumes of contaminated liquid requiring processing for offsite disposal.

G.1.3 Removal of Structural Material

During facility decontamination, removal of both metal and concrete surfaces may be required. However, the techniques for metal-surface removal are the same as those for equipment disassembly, discussed in Section G.2. The present discussion is limited to concrete removal.

Some concrete in nuclear facilities is contaminated below the surface and cannot be decontaminated to release levels by physical surface cleaning alone. In addition, some of the concrete and structural steel in the sacrificial shield surrounding the reactor vessel is activated as a result of neutron bombardment. In both instances, the structural materials must be physically removed and disposed of offsite to sufficiently decontaminate the facility.

Several criteria must be considered when selecting a material-removal method for a particular location. The method chosen should minimize personnel radiation exposure and airborne contamination dispersion. In addition, the size and weight of removed materials must facilitate packaging and shipping for offsite disposal.

The various methods available for concrete removal, with associated advantages and disadvantages, are listed in Table G.1-2. Of particular interest are the blasting techniques for use in bulk removal of concrete and the rock-splitting techniques for localized removal of contaminated concrete surfaces.

TABLE G.1-2. Concrete Removal Methods

Removal Method	Advantages	Disadvantages
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 control of size and weight of material removed • Selective removal of radioactive contamination possible 	<ul style="list-style-type: none"> • May cause vibrations and shocks (vibrational control possible by limiting maximum charge per delay) • Requires oxyacetylene torch for cutting exposed reinforcing steel • High noise level • Generates moderate quantities of dust that must be 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 • Cooling water spray minimizes airborne particulate material • Moderate-to-slow cutting speeds • Controlled rate of material removal 	<ul style="list-style-type: none"> • Reinforcing steel slows cutting speed and damages core drill • Requires oxyacetylene torch for cutting exposed reinforcing steel • Cooling water required for drilling may have to be filtered and recycled • Not ideally suited for operation on irregular surfaces or in cramped quarters
Flame Cutting	<ul style="list-style-type: none"> • Concrete and reinforcing steel 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 quantities of bottled oxygen • Smoke can 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
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 rock jack to break away cut concrete sections • Consumes large quantities of bottled oxygen • Smoke can potentially spread gross contamination throughout contamination-control envelope unless adequately controlled

Two techniques of lesser interest, primarily because of the copious quantities of toxic gases and/or smoke produced, are flame cutting and thermic lance cutting.

Flame cutting utilizes commercially available, 3-m-long by 19-mm-dia., thin-wall-conduit burning bars. Each bar contains up to seven dissimilar metals in the form of strips or strands, 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, with about 0.5 m of the bar left, the oxygen supply is shut off and the exothermic reaction stops. The remaining portion of the spent bar can be attached to a second bar and the burning process repeated, thus conserving bar material.

The thermic lance is also hand-held and uses powders of materials similar to those used in the aforementioned burning bar. The powders, together with oxygen, are forced through a tube or nozzle and ignited to form a high-temperature cutting medium.

G.1.3.1 Bulk Concrete Removal

A very effective way to remove and segment the activated concrete in the sacrificial shield is the use of explosives.⁽⁷⁾ Because the shield's interior can easily be enclosed within ventilation-confinement envelopes, dust and airborne contamination can be effectively controlled. Placement of blasting mats (~3 m by 4 m, weighing ~1600 kg) over the affected region prevents flying debris from penetrating the confinement envelopes. Fog sprays of water, typically used from 1 minute before to about 15 minutes after blasting, help to settle the dust created by the blasts. Although blasting sequences are designed to minimize air-pressure surges, the ventilation enclosures must be designed to withstand those pressure surges that do occur. Similarly, attention must be given to the building ventilation system to prevent surge damage to filters, with monitoring of the system to verify 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 in from the surface of the concrete to be removed, outside of

any reinforcing steel. The holes extend beneath the entire length of the surface to be removed, and explosives are inserted in the holes. The holes are tamped with sand, if necessary, to produce the desired blast depth. In other areas, 127-mm-deep holes on 0.3-m centers may be drilled perpendicular to the surface to be removed. The holes are filled with mortar after the explosives are inserted. The explosives used may be of several varieties, as shown in Table G.1-3.

Four to five strands of 150-grain blasting cord twisted together and placed in a tube inserted in the hole are effective, as are partial sticks of dynamite. Liquid explosives can be used where it is difficult to insert solids. The size of the area to be blasted at one time is controlled, so that an accident involving the explosives used presents an acceptably low potential for damage. Explosives in the building are limited to those in actual use; those awaiting use are stored outside the building. Each blast is conducted with the individual charges sequenced to minimize the resulting shock wave. Blasting mats minimize the scattering of debris, and water sprays reduce airborne dust loading. After blasting, the area is inspected and surveyed to verify that all contamination is removed and that all explosives have 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 can be accomplished by closing existing dampers during the blast; normal ventilation may be used in these areas except during the actual blast. In other areas, it may be possible to install covers over ventilation openings and remove them after a blast. Some areas 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.

G.1.3.2 Concrete Surface Removal

A number of techniques can be used to remove contaminated concrete surfaces in nuclear facilities such as a BWR. A comparison of the various techniques is presented in Table G.1-4.⁽⁸⁾

TABLE G.1-3. Explosives for Bulk 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 deep. In addition, it is effective for surface spalling, which requires the removal of very small burdens about 200 mm deep from exposed surfaces.

2. 85% High-Velocity Gelatin Dynamite

This explosive is primarily used in holes ranging in depth from 0.5 m to 1.5 m and as a partial loading in some holes when a concentration of energy is needed in a particular area. Breakage is excellent; however, it produces a crushed zone around the borehole, thus generating a larger amount of dust than the PETN.

3. Cast TNT

This explosive produces a high degree of fragmentation with less heaving effect than with conventional dynamite.

4. Binary Energy System (Liquid Explosives)

This explosive is mixed onsite and, when not mixed, is not legally classed as an explosive by most states. It is used in place of Cast TNT during the spalling of surfaces and to make shaped charges that 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 exhibits good shattering characteristics and produces larger rubble than any of the other explosives. Its primary use is in areas with little or no reinforcing where larger size rubble is desired.

6. Others

High-strength ammonia dynamite and plastics have been investigated with small-scale blasts. The effects indicated that they did not produce the desired results.

TABLE G.1-4. Comparison of Concrete Surface Removal Techniques

Technique	Limitation	Type of Rubble Produced	Size of Air Filtration System Required	Relative Removal Speed
Sand Blasting	Contamination embedded in pores not effectively removed	Small particles	Large	Slow
Flame Spalling	Heat may cause undesirable chemical reactions	Small particles	Large	Slow
Jack Hammer	Awkward to use on walls	Medium-sized pieces and small particles	Medium	Medium Fast
Impactor Powered Pneumatically or Hydraulically	Limited to large accessible facilities	Medium-sized pieces and small particles	Medium	Fast
Scrubber	Awkward to use on walls	Small pieces and particles	Medium	Fast
Water Cannon				
Handheld Modified 458 Magnum Rifle	Gun powder combustion products are produced	Small pieces coated with glycerine and gun powder combustion products	Small	Slow
Rapid-Fire Model	Limited to large accessible facilities	Small pieces coated with water	Small	Fast
Concrete Spaller with Air Drill to make Holes	Awkward to use on irregular surfaces or in cramped quarters	Medium-sized pieces and small particles	Small	Medium Fast

Sand blasting, where the surface is mechanically eroded away, and flame spalling, where intense heat is applied to concrete surfaces, remove only a minimal surface thickness and produce large quantities of small, contaminated particles. Sand blasting primarily removes paint and a little of the concrete surface but does not effectively remove contamination in the pores of the concrete or at expansion joints.⁽⁹⁾ A large exhaust and air filtration system is needed with both of these methods. These two techniques are relatively slow if the contamination penetrates beyond a thin surface layer.

Two surface removal methods used more extensively than the rest are jack hammers and impactors. Jack hammers, powered by compressed air, are readily available and are easily operated by one man. They are used to chip off the surface material deep enough to remove the contamination.⁽⁹⁾ Because they are difficult to position on walls and ceilings, jack hammers are used primarily on floors. Impactors (or hoe rams), similar in operation to jack hammers but much larger, have been used successfully in several

decontamination projects.^(8,9) An impactor uses a pick chisel point that is driven into the concrete surface with high-energy impacts several times per second. Impactors are powered either pneumatically or hydraulically, and are held and positioned with linkages typical of those on tractor-mounted backhoes. A medium-size air filtration system is necessary to control the dust produced by both of these surface removal methods.

Another tool used to chip away concrete surfaces is a scrubber (or "scab-bler"), a handheld tool with a gang of carbide-tipped bits. The bits rapidly impact the surface, causing small concrete pieces to be spalled. A dust shield with a vacuum attachment is placed around the bits to remove the dust generated. Dust removal increases this method's effectiveness by preventing the tool from pounding contaminated particles back into the concrete surface.⁽⁹⁾ The scrubber is relatively slow and is difficult to use on walls and ceilings.

The water cannon removes concrete surfaces by shooting very-high-pressure liquid jets at the surface, causing it to spall. This method has three advantages: 1) no initial surface preparation is needed, 2) the equipment does not contact the surface, and 3) the resulting rubble is composed of small pieces coated with liquid, thus minimizing dust generation. Two versions of the water cannon can be used. The first is a modified 458 magnum rifle that shoots solidified glycerine through a nozzle; the device is handheld and typically removes less than 20 mm of concrete from the surface. The second version uses stored, compressed gas to drive a piston that forces water through a small-diameter nozzle; this large unit is mounted on a backhoe or excavator, and is therefore limited to use in large, readily accessible areas.

The last technique, use of a concrete spaller, is selected in this study as the principal method for removal of contaminated concrete surfaces. This device permits localized concrete removal to depths of 50 to 75 mm with no explosions and very little dust. (The principal source of dust is the drilling of the hole into which the splitting tool is inserted.) Surface-removal rates in the range of 5 to 7 m²/hr have been reported.⁽⁸⁾ A dust shield with a vacuum attachment minimizes the spread of contaminated dust and can be used to collect all but the largest pieces of rubble.

The concrete spaller consists of three basic parts: a hydraulic cylinder, a push rod, and a bit with expanding wedges. The hydraulic cylinder is attached at one end and activates the push rod, which is installed inside the bit. A schematic of the device is shown in Figure G.1-2.

FIGURE G.1-2. Schematic of Concrete Spaller

The spaller is operated by inserting the bit into a predrilled hole about 55 mm deep and 25 mm in diameter. The hydraulic cylinder is then activated, forcing the push rod toward the end of the bit. The wedges in the bit are forced radially outward into the walls of the hole. Then, when the tip of the push rod reaches the bottom of the hole, it forces the wedges away from the bottom and causes the concrete to be spalled off.

The spaller is small, lightweight, and fully portable, and can be readily adapted to remote operation. It is connected to the hydraulic power source with hydraulic hoses. The spaller is about 0.4 m long, with a mass of about 10 kg.

For rapid removal of large surfaces areas, a number of the devices can be ganged together with a corresponding set of concrete drills and operated as a unit. The spacing between holes (0.2 m optimum) and the pattern of the holes

(triangular array) are important parameters in the effectiveness of this technique. Arrangement of the concrete drills and the splitting tools into a fixed-geometry array would ensure a relatively uniform removal pattern. Combining these ganged units with a vacuum transfer system for rubble removal would result in a fast and dust-free concrete removal method, one ideally suited to nuclear facility decommissioning applications.

G.2 EQUIPMENT DISASSEMBLY

Decommissioning of a nuclear facility requires the disassembly and removal of various contaminated equipment systems. The equipment must be segmented into pieces small enough to facilitate either onsite entombment or packaging for offsite shipment and disposal, depending on the decommissioning mode selected.

Any of a number of methods can be used to disassemble and remove a particular piece of equipment. However, equipment-related parameters (e.g., size, location, design, and radioactive contamination and/or activation) and manpower/cost efficiencies of the various methods dictate the appropriate method for any given situation. In some cases, the required tools are available as part of the facility's normal operating-equipment complement; in others, the tools are readily available commercially (e.g., boltcutters, pipe saws, tubing cutters, and impact wrenches). However, some methods require the use of unique, specialized tools and equipment. These specialized items are discussed in the following subsections and include an underwater manipulator for handling cutting equipment, cutting equipment for sectioning large items, and linear-shaped explosives for certain cutting applications.

Meeting the need for special tools and equipment requires: development of functional and operational specifications for the devices required, preparation of detailed designs for noncommercial devices and for modifications to commercially available equipment, procurement or fabrication of the identified equipment items, establishment of a mockup facility for any necessary development testing, and personnel training in the operation of the equipment.

G.2.1 Underwater Manipulator

The underwater sectioning of the neutron-activated reactor vessel and vessel internals requires a manipulator to handle the cutting equipment. The equipment for cutting these structures (e.g., the core shroud) in large BWRs must be operated and maneuvered under 7 to 25 m of water and in intense radiation fields. In the reference BWR, the existing refueling platform cannot be modified for this purpose, since it is required for handling and packaging the spent fuel and the neutron-activated material segments for shipment. Therefore, specially designed manipulators, such as those presented in References 10 and 11, are required.

The particular underwater manipulator needed depends on the details of the dismantling plan and on the state-of-the-art in cutting equipment at the time. To ensure proper performance, the manipulator must be specifically designed and engineered for the planned application and must undergo extensive predismantlement testing. For dismantling the reference BWR, the total development program for this device is anticipated to cost as much as \$1 million.

The underwater manipulator for the reference BWR should be base-mounted, remotely controlled via bottom-entry services, and no taller (when mounted) than the reactor vessel flange. This would allow underwater transfer of the vessel internals sections to the dryer and separator storage pool without removing the manipulator. It should be supported radially, both at the base and at the height of the cutting equipment, for proper cutting alignment.

G.2.2 Cutting Equipment

The principal equipment anticipated to be used for cutting activated and/or contaminated items are the oxyacetylene torch, the plasma-arc torch, and the arc saw. This equipment can be used either under water or in air. The oxyacetylene torch is a relatively common device and, therefore, it is not discussed here; cutting rates and other information are available in Reference 12. Descriptions, cutting characteristics, and requirements of the plasma-arc torch and the arc saw follow.

G.2.2.1 Plasma-Arc Torch

Plasma-arc cutting employs an extremely high-temperature, high-velocity, ionized-gas arc between the electrode in the torch and the piece to be cut. The arc is constricted by passing it through a nozzle with a small-diameter orifice. This localizes the arc so that its energy is concentrated on a small area on the work piece, where its intense heat melts the metal. The melted metal is continuously removed by the jet-like action of the gas stream to form a kerf. The heat and force of the arc stream produce a high-quality, saw-like cut. If inert gases are used, the cutting process is dependent on thermal action alone. When cutting materials such as mild steel or cast iron, increased cutting speeds can be achieved using oxygen-bearing 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 G.2-1.

FIGURE G.2-1. Plasma-Arc Torch Schematic Diagram

The electrical circuit is basically the same as that used for gas tungsten-arc welding, with the exception that an electrical lead is taken from the grounded work piece and passed through a resistor to the nozzle of the torch. A high-frequency spark is used to complete the circuit between the electrode and the nozzle, thereby producing a pilot arc that initiates the main arc from the electrode to the work piece.⁽¹³⁾

There are two types of electrode holders and cutting controls, one designed for manual cutting and the other for mechanized cutting with the torch mounted on an automatic travel mechanism. Water cooling is used to prolong the operating life of the torch. Torch surfaces are insulated to protect the operator from electrical shock. Gas pressure regulators and flowmeters control the cutting gas. A contactor, usually built into the power supply, is required for turning the power on and off.

Arc-constricting nozzles are available in a variety of diameters and shapes. The required shape and diameter of the nozzle depends on the application and the current to be used; high currents require larger-diameter orifices. Both single-port and multiport nozzles are available. Multiport nozzles generally have auxiliary ports arranged in a circle around a central orifice.

The power supplies used have open-circuit voltages in the range of 120 to 400 VDC. The high open-circuit voltage is required for heavy-duty cutting, such as severing thick plates and piercing metal as thick as 50 mm.

Mixtures of argon or nitrogen with hydrogen are generally used for cutting stainless steel, as well as aluminum and other nonferrous metals. Nitrogen and oxygen, supplied separately to the torch and mixed in the nozzle, are used for cutting carbon steel, cast iron, and alloy steels.

For mechanized operations in air, the torch standoff distance is set between 6 and 16 mm. The cutting current, gas type, and gas flow rate are set at values recommended by the equipment manufacturer. These values are primarily determined by the type and thickness of the material being cut. After the operator initiates the cutting arc, the sequence of operations is regulated by an automatic cutting control.

For manual operation, the operator selects the gas flow rate and cutting current from the table of conditions supplied by the manufacturer or from data obtained from prior testing. The torch is held over the work piece, the cutting arc is struck, and the torch is manually guided along the cut. At the end of the cut, the arc automatically extinguishes, and the control automatically opens the contactor and closes the gas valves. The operator can extinguish the arc at any time simply by moving the torch away from the work piece.

The plasma-arc cutting process can be used in air or under water. It is especially adaptable to automation and is thus useful when highly radioactive material is to be cut (e.g., the pressure vessels of the Elk River Reactor in Minnesota and the Sodium Reactor Experiment in California).^(7,9) As it is not necessary to start the cut at the edge of the plate, the plasma-arc torch is particularly adaptable to cutting holes in large plates and vessels. It is also well adapted to gouging applications, including pad washing and scarfing. However, because of the short torch standoff distance, plasma-arc cutting is not suitable for some applications, particularly in tight spaces. Air carbon-arc cutting can be used for such work. Plasma-arc cutting is preferred where it is possible.

Some typical conditions for mechanized plasma-arc cutting of stainless steel in air are given in Table G.2-1. Recent tests with the plasma-arc torch have demonstrated its ability to cut 150-mm-thick stainless steel and 180-mm-thick carbon steel.⁽¹⁴⁾

TABLE G.2-1. Typical Conditions for Plasma-Arc Cutting of Stainless Steel in Air

Thickness (mm)	Speed (m/min)	Orifice Diameter (mm)	Current (amps)	Gas Flow (m ³ /hr)
13	2.5	3.2	300	4.2 N ₂
25	1.2	4.0	400	5.0 N ₂
50	0.5	4.0	500	3.7 A & 2.0 H ₂
75	0.4	4.8	500	3.7 A & 2.0 H ₂
100	0.2	4.8	500	3.7 A & 2.0 H ₂

G.2.2.2 Arc Saw

The arc saw, a state-of-the-art metal-cutting device, is currently being developed for contaminated-equipment segmentation, with initial development and demonstration work already completed.⁽¹⁵⁾ Results to date indicate that the device holds great promise, and additional development work will undoubtedly provide a more sophisticated, commercially available unit in the near future. The cutting process is both economical and expedient relative to other methods. Cutting speeds of around 10 m/min are anticipated for 10-mm-thick steel plate, with the speed being inversely proportional to the plate thickness.⁽¹⁶⁾

The prototype arc saw used a 11.2-kW motor to rotate a 0.91-m-diameter, 6.35-mm-thick blade made of copper or mild steel. Blade speed was approximately 880 rpm, and blade cooling was accomplished by a water-spray system installed in the blade guard. A material-cut/blade-wear ratio of 20 to 1 was reported for the 0.91-m-diameter blade. It is expected that increased blade diameters may reduce associated costs.⁽¹⁵⁾

Cutting can be accomplished remotely, either in air or under water, with automatic positioning and tracking of the saw blade during cutting operations. All equipment, with the exception of the blade, is commercially available, but modifications are necessary. Blades can be made in any well-equipped machine shop. The arc saw is shown schematically in Figure G.2-2.

FIGURE G.2-2. Arc Saw Schematic Diagram

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 servo controller for automatic positioning and tracking of the saw blade during cutting operations and a recorder to monitor voltage, amperage, blade travel, and servo current.

Low-voltage, high-amperage current is applied to the rotating blade, creating an electric arc discharge between it and the grounded work piece. The discharge spark-erodes both the blade and the work piece. The blade is advanced through the work piece to produce a kerf 1 to 4 mm wider than the blade thickness.

If, because of position change or vibration, a portion of the work piece falls against or pinches the blade, the point of contact spark-erodes away because the "electrical leading edge" of the blade is transferred to the point of contact. This reduces the potential for binding of the blade. Consequently, the arc saw can cut through a variety of materials, shapes, and loose components that would be difficult to cut with conventional saws.

The arc saw blade can be scaled to match the cutting requirements of the job. Blades up to 3 to 4 m in diameter (up to 25 mm thick) for cutting large-diameter vessels appear to be feasible. Heavier (thicker) blades are likely to tear most non-conducting materials (non-metals) they encounter, thus eliminating the need to remove them from the work piece before cutting. It also appears that several saws could be ganged together for simultaneous parallel cutting.

G.2.3 Linear-Shaped Explosive Charges

An economical and expedient method of reducing the physical size of equipment and piping to allow further processing or packaging for disposal is the use of self-contained, linear-shaped explosive charges. This method minimizes personnel radiation exposure and is particularly advantageous in areas with high radiation levels. Linear-shaped charges have been used extensively in the last 15 to 25 years.^(17,18) Recently, such methods (i.e., the perforation and cutting of solids by ultra-high-velocity particles under

explosive attack) have been used to segment and remove activated components and contaminated systems (e.g., fuel-pool liners and piping) that are not amenable to conventional removal techniques.^(9,19) Figure G.2-3 shows a cross section of a typically positioned linear-shaped charge.

A linear-shaped charge consists of a V-shaped tubular casing filled with an explosive (cyclotrimethylenetrinitramine) commonly called RDX. The casing is generally copper, but other soft materials such as aluminum and lead can be used. The charges are equipped with detonators that are activated using a primacord lead having a high-explosive core of pentaerythritol tetra-nitrate (commonly known as PETN) that, in turn, is initiated with an electric blasting cap. Clamp-on charges, available commercially, eliminate many problems in placement, handling, and detonating. The number of charges that can be detonated at one time is limited only by the blast effect on nearby equipment. Shock-wave and fragment damage can be reduced appreciably by placing blast curtains or other barriers in the vicinity of the detonation to disrupt the shock wave and intercept the fragments.⁽²⁰⁾

FIGURE G.2-3. Cross Section of Typically Positioned Linear-Shaped Charge

The principle of the linear-shaped charge is that, as the detonation wave collapses the inverted V, the casing material becomes a jet of extremely hot metal particles traveling at very high velocity. These particles then tear through the material to be cut. The melting and subsequent fusing of the casing material with the base material being cut, together with the ragged edges of the finished cut, can make electropolishing of those edges very difficult.⁽⁵⁾ Therefore, in-situ decontamination (either chemical or mechanical) prior to explosive cutting is recommended to minimize unnecessary waste of strategic materials.

The use of linear-shaped charges has been documented for cutting such diversely shaped objects as:^(20,21)

- 0.61-m-O.D. by 16-mm-wall stainless steel pipe
- 100-mm by 100-mm by 10-mm-thick angle iron
- 25-mm-diameter solid stainless steel bar
- 152-mm by 10-mm-thick web "H" beam.

G.3 RADIOACTIVE WASTE PACKAGING AND SHIPPING

Radioactive wastes are generated during the decommissioning of a BWR facility. This section presents the methods and requirements for packaging these wastes and shipping them to an authorized disposal site.

G.3.1 Packaging of Radioactive Wastes

The radioactive wastes that result from decommissioning can be classified as follows:

- combustible or noncombustible
- activated or contaminated
- wet or dry.

The bulk of the decommissioning wastes from the reference BWR are dry, noncombustible, and either activated or contaminated. They include the activated reactor vessel and internals, the activated and contaminated concrete from the sacrificial shield, contaminated concrete from walls and floors, and contaminated piping and equipment. The contact radiation dose rates from these

materials vary from a few mrem/hr to thousands of rem/hr. Different types of packaging and shielding are required, depending on the radiation levels involved.

Disposable, 0.28-m^3 (0.33-m^3 burial volume) steel cask liners are used for packaging, shipping, and burying the activated portions of the reactor vessel and of the vessel internals. These containers are estimated to cost \$450 each. In some cases, lead shielding must be added at an estimated average cost of \$1.23/kg, including labor and energy.

The bulk of the contaminated material is packaged in a standard container, illustrated in Figure G.3-1. The container is approximately 1.2 m by 1.2 m by 2.4 m, made from plywood, and internally reinforced and sealed with fiberglass to ensure confinement of loose contamination. Containers of this type are estimated to cost \$400 each. When necessary, containers of special sizes and shapes are fabricated. Some items do not require shielding but are larger and/or heavier than can be accommodated with plywood containers. In these cases, steel containers are specially fabricated at an average cost of about \$6,000 each.

FIGURE G.3-1. Typical Plywood Shipping Container

Certain contaminated equipment items (e.g., small heat exchangers and small tanks) are packaged by capping the piping connections with welded metal covers and using the items' outer shells as the containers. The moisture separator reheaters and the feedwater heaters are cut into sections, after which each section is capped and handled as its own container.

Dry, combustible radioactive wastes include filter cartridges from liquid streams, HEPA ventilation filters, and miscellaneous cleaning materials (e.g., rags, mops, anti-contamination clothing, and plastic sheeting and bags). These materials are compacted as necessary and packaged in 0.21-m^3 steel drums costing \$20 each.

Standard methods are in place for handling and disposing of the radioactive liquid wastes that are generated during decommissioning. The assumed practice in the reference BWR (and numerous operating power reactors) is to concentrate the liquids by evaporation to a product that can be packaged (together with dewatered, spent ion-exchange resins) and solidified in containers prior to disposal. For this study, the containers are assumed to be 2.3-m^3 (2.84-m^3 burial volume) steel cask liners costing \$550 each.

G.3.2 Shipment of Radioactive Wastes

The spent fuel and irradiated fuel channels from the reference BWR are assumed to be shipped offsite by rail; all other radioactive wastes are assumed to be shipped by truck. Rail shipment of spent fuel is standard operating practice at the reference plant and, therefore, is not discussed here. (References 22 and 23 discuss rail shipment of spent fuel.) Truck transport of radioactive wastes from decommissioning the reference BWR is discussed below.

All radioactive waste materials removed from the reference BWR during decommissioning are assumed to be shipped in exclusive-use trucks to offsite disposal, via the interstate highway system. The hauler is assumed to have the appropriate Nuclear Regulatory Commission (NRC) license and permits from the Department of Transportation (DOT) before he handles the radioactive waste material. Several commercial transportation companies have the special capabilities and trained drivers for hauling radioactive wastes.

All shipments of radioactive material must be made in compliance with federal, state, and local regulations. Federal (DOT and NRC) transportation regulations establish container specifications, dose rate limits, and handling procedures to ensure the safety of the public and the transportation workers during shipment of radioactive materials.⁽²⁴⁾ In addition, for highway transport, 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 (49 CFR 173.393):⁽²⁵⁾

- 1000 mrem/hr at ~1 m (i.e., 3 ft) from the external surface of a package shipped inside a closed vehicle
- 200 mrem/hr at any point on the external surface of a closed vehicle or an exposed shipping container (e.g., a shielding cask)
- 10 mrem/hr at 2 m from the external surface of the vehicle
- 2 mrem/hr at any normally occupied position in the vehicle.

These dose rate limits are illustrated in Figure G.3-2 for closed truck transport.⁽²⁶⁾ All of these criteria must be met for each shipment.

FIGURE G.3-2. Radiation Dose Limits for Closed Exclusive-Use Vehicles⁽²⁶⁾

Most of the activated waste packages will exceed the package surface dose-rate limit, as will some of the solidified wet waste packages and dry combustible waste packages. These must be shipped in nondisposable shielded shipping containers, some typical examples of which are shown in Figure G.3-3.

All wastes are assumed to be transported to a licensed commercial low-level waste burial ground. Such facilities are currently available in Washington, Nevada, and South Carolina. For this study, the average one-way shipping distance is assumed to be 800 km for all shipments. Disposal site restrictions are assumed to limit the maximum quantity of radioactivity per shipment to 50,000 Ci. There are currently no limits on the dose rate at the surface of a disposable container, but burial costs increase with increasing surface dose rate.

FIGURE G.3-3. Typical Shielded Shipping Containers

A formal accident control and recovery plan is assumed to be developed before the first radioactive shipment of decommissioning waste is made. The plan is to provide for rapid and orderly utilization of utility, carrier, state, and municipal emergency personnel, as well as NRC radiological assistance teams, as required in the event that any transportation accident occurs. Procedures for control of contamination, radiation exposure, bodily injury, and property damage are included in the recovery plan. Also included are procedures for salvage and recovery of the radioactive shipment.

G.4 CONTAMINATION CONTROL

Many decommissioning activities, particularly the cutting operations required for equipment disassembly, have the potential for generating significant amounts of airborne radioactive contamination. In addition, decommissioning can involve operations in areas with smearable radioactive contamination; the movement of personnel, equipment, and materials in these areas can result in the further spread of radioactive contamination. To minimize the personnel hazard and the potential for widespread contamination of work areas, contamination control is required.

Radioactive contamination control can be divided into three basic approaches:

- local mitigation of contamination sources
- collection of contamination
- isolation of contaminated areas.

These approaches are discussed in the following subsections.

G.4.1 Local Mitigation of Contamination Sources

Mechanical or physical measures can be used to limit the spread of radioactive contamination. Two methods that have been successfully used are 1) water sprays to reduce airborne dust dispersion and 2) painting of contaminated surfaces to prevent smearing.

The wetting of dust with water or other liquids is one of the oldest methods of contamination control and can be very effective if properly used.⁽²⁷⁾ Water sprays are widely used to control fugitive dust emissions from construc-

tion sites. The spraying of water containing detergent (as a wetting agent) has been used in the nuclear industry to reduce dust concentrations in air.⁽²⁸⁾ To be effective, the liquid application must be designed to blanket the dust source completely and to wet the dust particles thoroughly. Various types, sizes, and patterns of spray nozzles are used, depending on the physical properties of the dust, the type and size of the dust source, and the degree of control desired. Water sprays can be used in combination with other contamination control techniques, and are commonly used for dusty operations such as concrete removal.⁽⁹⁾

Nonflammable, strippable paint can be used to seal porous surfaces (e.g., concrete) to prevent penetration of contamination into the surfaces. Paint can also be used to seal smearable contamination already present on surfaces to prevent subsequent contamination spread.⁽⁹⁾ Spraying is generally the easiest and quickest method of application. Painting is especially useful in high-traffic areas, where smearable contamination is likely to be picked up and spread around on shoe covers and equipment wheels.

G.4.2 Collection of Contamination

Collection of radioactive contamination before it can be dispersed (preferably as it is generated) reduces the need for cleanup subsequent to some decommissioning activities. Various collection methods can be used. Vacuum collection and portable ventilation systems are discussed here.

G.4.2.1 Vacuum Collection

Contaminated materials can be collected as they are generated using vacuum systems. A dust shield with a vacuum attachment can be installed on the tool (e.g., concrete spaller or scrubber) being used.^(8,9) As the contaminated dust is generated, it is drawn into the vacuum system and deposited in a collection drum. The outlet air is filtered (with roughing and HEPA filters) to prevent the collected contamination from being expelled.

Various designs for vacuum collection systems are possible, depending on the required operating characteristics. One such system, shown schematically in Figure G.4-1, is described in Reference 29. This system, originally designed for collection of contaminated soil, uses a standard 0.21-m³ waste drum to

FIGURE G.4-1. Vacuum Collection System Schematic

collect the contaminated material. When the drum is filled, it is capped and sealed for disposal. A special, commercially available vacuum lid, employing a cyclone baffle arrangement to enhance dust settling, is modified to accept an inexpensive, disposable roughing filter. A HEPA filter and power/vacuum-blower unit, mounted on a steel pallet, complete the system. The system is reported to be capable of pulling up to $\sim 28 \text{ m}^3/\text{min}$ of air at 110-mm-Hg vacuum, and is estimated to cost less than \$5000.

G.4.2.2 Portable Ventilation Systems

Portable ventilation systems can be used to confine and collect airborne particulates generated during decommissioning operations.⁽⁹⁾ General design information concerning such systems is discussed at length in Reference 27. Two portable ventilation systems, a work enclosure and a fume exhauster, are discussed here.

Portable Filtered Ventilation Enclosure. A typical portable filtered ventilation enclosure unit is illustrated in Figure G.4-2. A large squirrel-

FIGURE G.4-2. Portable Filtered Ventilation Enclosure

cage blower is coupled with a high-efficiency particulate air (HEPA) filter preceded by a glass-fiber roughing filter, all mounted on a wheeled cart. A flexible duct couples the cart unit to the enclosure unit that surrounds the work area and collects the materials being emitted. Roughing filters are installed at both the inlet and the outlet of the enclosure unit. The enclosure unit may have whatever shape best performs the required function at a particular location. A simple, rectangular open-faced box will suffice for many applications.

Radiation detection devices are used to monitor the buildup of radioactive material on the filters. A differential pressure gauge is installed across the HEPA filter to monitor the increasing pressure drop as particulates build up on the filter. Filters are changed when either the dose rate from the collected radioactive particles or the differential pressure across the HEPA filter reaches a predetermined level.

Portable Filtered Fume Exhauster. Another type of portable filtered ventilation system, a fume exhauster, is illustrated in Figure G.4-3. This system has an electrostatic precipitator coupled with a roughing filter, HEPA filter, air-handling motor, squirrel-cage blower, and one or two free-standing intake ducts. The fume exhauster is used to collect radioactive and nonradioactive particulates at the point of generation. This high-volume ventilation system captures all types of particulate matter with efficiencies of greater than 97% for the electrostatic unit and at least 99.95% for the HEPA filter. The advantages of this unit are its portability, its ability to handle large volumes of particulate-laden air, and its generation of relatively small amounts of solid wastes (HEPA filters).

FIGURE G.4-3. Portable Filtered Fume Exhauster

Buildup of radioactive materials on the precipitator and filters is monitored by radiation detection devices. To monitor the buildup of particulates on the filter, a differential pressure gauge is installed across the HEPA filter. The electrostatic precipitator is flushed and the filters are changed when either the dose rate from collected radioactive particles or the differential pressure across the HEPA filter reaches a predetermined level.

G.4.3 Isolation of Contaminated Areas

One method of controlling contamination is the use of barriers to isolate contaminated areas from those with lesser or no contamination. Isolation is an important tool during continuing care as well as during active decommissioning.

G.4.3.1 Isolation During Active Decommissioning

One type of barrier commonly used in the nuclear industry to isolate contaminated areas is a "greenhouse." A greenhouse is constructed by covering a framework, usually steel scaffolding or wood frame, with plastic sheeting and sealing all joints. Overlapping flaps of plastic are generally used for the door. The greenhouse is connected either to the plant ventilation system or to a portable system (see Section G.4.2.2), which prevents outward leakage of contamination by drawing a slight vacuum on the greenhouse.⁽⁹⁾ Greenhouses can be semi-permanent, portable structures that can be moved from one location to another as needed, but are more often temporary confinement structures that are dismantled and discarded after each job.

In many cases, construction of a complete greenhouse is unnecessary. A simple plastic curtain partitioning off one section of a room may be all that is required to isolate a contaminated area. The type and degree of isolation required depends on the equipment or structures involved, the associated level and mixture of radioactive contamination, and the decommissioning operation being performed.

G.4.3.2 Isolation During Continuing Care

Prior to the continuing care period, portions of the facility containing significant amounts of radioactive contamination that are not removed during active decommissioning must be isolated from the remainder of the facility.

Potential pathways for the migration of contamination from these areas are blocked by the installation of physical barriers. Besides acting as contamination control barriers, the barriers are designed to discourage unauthorized personnel entry into contaminated areas. Structurally substantial barriers are used, with extensive use of steel.

The preferred method of barrier installation is by welding. This method is used on piping, ventilation ductwork, and equipment penetrations. Piping is cut and plates are welded over the open ends. Rectangular plates are welded in inlet ventilation ductwork. All welds are inspected by appropriate techniques. A polysulfide adhesive is used to seal the perimeter of pipes and other penetrations through the concrete walls into contaminated areas. Care must be taken during the sealing procedures not to interfere with any fire alarm sensors that may be present in the ducts.

The polysulfide adhesive is also used whenever seals must be formed on concrete surfaces or on metal surfaces where welding is not possible. This adhesive forms a strong, durable bond with these materials. After it has cured, the adhesive remains flexible, permitting differential thermal expansion of the two bonded surfaces without breaking the seal. This adhesive can be applied with a spatula or extrusion gun and will not run after application.

Metal doors are secured by welding them to the metal door frame. Spot welds are used where the intent is simply to make the door inoperative. Continuous welds are made or sealant is used where an airtight seal is required. Where the door frame meets the wall, it is sealed with the polysulfide adhesive.

Pressure equalization lines are installed between the outside environment and the interior of the Reactor Building, the Turbine Generator Building, and the Radwaste and Control Building. These lines prevent pressure differentials (due to temperature or atmospheric pressure changes) from developing between the insides of the buildings and the outside atmosphere. The lines are equipped with replaceable absolute filters to prevent contamination from being entrained in the air flow out of the buildings.

G.5 QUALITY ASSURANCE

To meet the QA regulations imposed upon it, a BWR decommissioning project requires QA planning from the earliest stages. As each detailed procedure is developed, the QA portions are included. Current regulations and guides that could apply to decommissioning are discussed in Chapter 5 of Volume 1.

In this study, the QA Supervisor is assumed to report directly to the Decommissioning Superintendent. Thus, the QA staff is organizationally independent of the project staff, up to the highest possible level of the decommissioning organization.

G.5.1 Method Review

When a new procedure is written, a documented peer review is held. The review ensures that adequate consideration is given to physics, radiation, safety, accident, security, thermal, stress, QA/QC, and hydraulic analyses. Other checks, tests, and formal design reviews may be required. Other aspects that are reviewed include reliability criteria, the choice of test methods, and the choice of processes or materials. The conservatism in margins of safety established for procedures, equipment, or structures also receives documented review.

G.5.2 Procurement Document Control

The project's suppliers, including fixed-price contractors and shipping cask vendors, are selected from a QA list of qualified suppliers. QA inspections of a vendor's plant are performed as required to place new firms on the qualified list.

Procurement Document Control verifies that any QA requirements for specified hardware, materials, or services are clearly and accurately identified in procurement documents or purchase orders. This includes references to applicable drawings, specifications, standards, codes, and regulations; required records of test data, certifications, and qualifications of personnel and procedures; and special handling, shipping, or packaging requirements. This control is 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.

G.5.3 Work Instructions, Procedures, and Drawings

Formal documentation of work instructions and procedures, drawings, and information management procedures is required, with responsible personnel clearly identified. These procedures verify the proper completion of activities or steps in the decommissioning process (e.g., the disassembly sequences that ensure that vital services remain intact). The latest revisions of all such documentation are required to be at the work location. Changes to such documents receive the same levels of review and approval as original planning.

G.5.4 Document Control

Document Control verifies that all essential documentation receives the proper review, approval, release, change, and distribution control. An established procedure is used to eliminate obsolete or erroneous information.

G.5.5 Identification and Control of Procured Items

Inspection of purchased materials or items is performed at the time of receipt. Items must be sufficiently identified to allow proper selection for use, completion of processing, or segregation of nonconforming items. The shelf lives and the required storage and use conditions of chemicals must be clearly stated.

G.5.6 Special Processes

Decommissioning may involve several tasks requiring new or unusual work procedures (e.g., large-scale chemical decontamination work and the use of explosive charges). Special provisions are made to identify and control any unusual processes or procedures, using reviews, written procedures, and audits.

G.5.7 Inspection and Test

All items produced for the project must be properly inspected or verified. Inspections in the vendor's plant may be necessary for some items. Inspection instructions are required, and records of inspections are prepared and retained. Inspection of each essential operation is required, as well.

G.5.8 Calibration

A system of calibration control is used to ensure the validity of instrument readings. Accuracy requirements and tolerances for project measurements are identified. Equipment with calibration requirements includes radiation detection devices and environmental monitoring systems. Calibration procedures for instruments that monitor and control vital services are reviewed and revised as necessary.

G.5.9 Shipping and Receiving

Special instructions are prepared for the packaging, shipping, and receiving of components and materials for the project. Included are radioactive shipments requiring QA procedures for each package type specifying all steps in the package inspection, preparation, loading, and closing. Signed checklists are used. Seals must be carefully verified on internally contaminated units having no additional outside packaging.

G.5.10 Operating Status

Identification procedures are used to prevent inadvertent operation of systems temporarily or permanently removed from service. Restricted use may be necessary for equipment (such as pumps) that has not been fully removed from service.

G.5.11 Nonconformance and Corrective Action

All nonconformances and resulting corrective actions are documented and the reports retained as part of the project records.

G.5.12 Special Quality Records

An index of records and their required retention periods is maintained. These records include operating logs; results of reviews, inspections, tests, audits, material analyses, and work-performance monitoring; orientation, training, and qualifications of personnel; procedures; drawings; specifications; criteria documents; procurement documents; shipping records; and nonconformance and corrective action reports. Extensive documentation of dose rate surveys is also included.

G.5.13 Quality Auditing

Regular audits verify that each QA procedure is being followed. Examples include witnessing radioactive shipments, auditing calibrations, verifying that up-to-date procedures are at all work locations, and reviewing inspection records.

G.6 ENVIRONMENTAL SURVEILLANCE

The required levels of environmental surveillance for active decommissioning differ from those for continuing care. The programs for each of these cases follow.

G.6.1 Environmental Surveillance During Active Decommissioning

The environmental surveillance program for the operating BWR can be applied to active decommissioning as well. The following objectives are relevant:⁽³⁰⁾

- detection of sudden changes and evaluation of long-term trends of (radio-nuclide) concentration in the environment, with the intent to detect failure to adequately control releases and then to initiate appropriate actions
- assessment of the actual or potential exposure of man to radioactive materials or radiation present in his environment, or 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.

Methods, procedures, and performance criteria for environmental surveillance at nuclear facilities are discussed in detail in Reference 31.

Basic radiation-exposure and radioactive effluent-release criteria applicable to population exposure are given in Title 10 of the Code of Federal Regulations.⁽²⁵⁾ For nonradioactive contaminants, consideration must be given

to applicable standards such as water quality criteria and the Environmental Protection Agency (EPA) ambient air quality standards.^(32,33) In any case, local or state air quality criteria presumably apply on a site-specific basis.

In addition, the interfaces (of the environmental monitoring program) between the plant owner, the appropriate state agencies, and the EPA should be stipulated in the application for the possession-only license.

A suggested minimum program of environmental radiological surveillance conducted outside the plant for the purpose of establishing population dose is shown in Table G.6-1. This minimum program continues until all fuel and source material is shipped from the site. At that time, approximately 1.5 years after reactor shutdown, the program is reduced in scope.

The analytical detection limits given in the table are based on practicable routine radioactivity-measurement techniques, and in all cases should be sufficient to quantify radionuclide concentrations resulting in conservatively estimated whole-body and individual-organ doses of <50 mrem/year, depending upon the values derived for the disposition criteria for specific nuclides. These nuclides are identified on completion of the operational monitoring program. They are then defined in terms of concentrations in environmental media at levels that, if sustained, result in doses exceeding acceptable levels for unrestricted public use. The derivation of these levels corresponds with the site-specific method of assessing offsite doses from radioactive materials in estimated gaseous and liquid effluents (if any), and is assumed to be consistent with the recommendations of the Federal Radiation Council and the International Commission on Radiological Protection.^(34,35) The monitoring program is designed to integrate fully with any ongoing programs of the state where the BWR is located. Quality assurance is 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 G.6-1 are assumed to be conducted according to specifications established during the operation of the reference BWR. The program, and any changes thereto, are documented with the NRC (and to other appropriate regulatory agencies) as environmental

**TABLE G.6-1. Recommended Basic Environmental Monitoring Program
for the Active Decommissioning Period**

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 ^(c)	¹³¹ I	0.1 pCi/m ³	2	4
Direct radiation	Quarterly	TLD ^(d)	1.25 mrem/quarter increase	8	10
Rainfall	Monthly	Gross Beta	0.5 pCi/l	1	2
		Tritium	1000 pCi/l	1	2
		Gamma Scan ^(e)	25 pCi/l/isotope	1	2
Soil	Semiannually	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	-(f)	5(g)
		¹³¹ I ^(c)	0.5 pCi/l	-	5(g)
		Gamma Scan	50 pCi/l/isotope	-	5(g)
Aquatic Samples					
Surface water	Monthly	Gross Beta	0.5 pCi/l	3	3
		Tritium	1000 pCi/l	3	3
		Gamma Scan ^(e)	25 pCi/l/isotope	3	3
Well water	Quarterly	Gross Beta	0.5 pCi/l	-	3
		Tritium	1000 pCi/l	-	3
		Gamma Scan ^(e)	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)	-	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 that concentration that is three standard deviations above the average concentration in a blank sample, ensures accuracy limits of ±25%.

(b) Performed if gross beta exceeds 0.1 pCi/m³.

(c) ¹³¹I analyses needed only for first month following shutdown.

(d) Thermoluminescent dosimeter.

(e) Performed if gross beta exceeds 10 pCi/l.

(f) Indicates no sample taken.

(g) Includes one sample from a local milk processor.

technical specifications, in accordance with the terms of the possession-only license. Quality control mechanisms are exercised over all procedures involved.

Environmental monitoring during the active decommissioning period is carried out by a specialty contractor. (See Section H.4 of Appendix H.)

G.6.2 Environmental Surveillance During Continuing Care

An abbreviated version of the environmental monitoring program conducted during plant operation is carried out during continuing care. The purpose of the program is to identify and quantify releases of radioactivity to the environment. There is no intent to provide a surveillance program adequate for all potential nonroutine or accidental releases, although the proposed program is useful in evaluating most lapses of control. For emergency situations involving radionuclide releases (from events such as fire or malicious acts) requiring prompt emergency actions to minimize public risk, special surveillance requirements would apply.

Quality control mechanisms are exercised over all procedures involved. The surveillance system operation is checked at least quarterly. Instruments are calibrated at least quarterly, and always following maintenance. Remedial measures are 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 analyzing audit samples of known concentrations, are likely to be instituted.

Changes in background radiation levels, in environmental radiation 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, trends in environmental results that would reasonably permit reduction of the program.

The entire program is anticipated to be reviewed at the following post-decommissioning milestones:

- after all fuel and source material are shipped from the site (approximately 19 months after plant shutdown)

- 10 years after plant shutdown
- after 10 half-lives of ^{60}Co decay (approximately 53 years), economic advantages of further decommissioning effort are ascertained by the owner, and environmental monitoring could conceivably be eliminated.

The suggested minimum program of environmental surveillance, shown in Table G.6-1, continues 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 are likely.

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APPENDIX H

GENERIC DECOMMISSIONING INFORMATION

Although all the alternative decommissioning modes are separate and distinct, they have some common requirements. This appendix provides information concerning these common requirements, in the following order:

- decommissioning staff
- planning and preparation
- specialty contractors
- essential systems and services
- system decontamination and draining and radioactive liquid handling
- equipment deactivation and contaminated area isolation.

H.1 DECOMMISSIONING STAFF

For the reference BWR, the decommissioning staff is organized as shown in Figure H.1-1. Two main and three auxiliary parallel branches report to a decommissioning superintendent. The operational branch, under a decommissioning engineer, plans and performs the actual decommissioning tasks. The safety branch, under a health and safety supervisor, plans and conducts both radiological and industrial safety programs. The auxiliary branches handle security, financial, and quality assurance matters.

The primary decommissioning activities are performed on a two-shift, 5-day-week basis. However, support activities (i.e., system decontamination and draining, spent-fuel/activated material shipping, and radwaste system operation) and security functions are carried out on three shifts, around-the-clock, 7 days per week. In addition, the main control room is manned full time for operation of essential systems and services.

The basic working unit is the shift, which is supervised by a shift engineer. The crew on each shift includes: a crew leader (typically a reactor operator), utility operators, and laborers, plus craftsmen (e.g., welders, pipefitters, electricians, and air-balance technicians) and health physics technicians assigned as needed. Craftsmen and health physics technicians

FIGURE H.1-1. Decommissioning Staff Organization

on the support crews report directly to the crew leaders because, on the third shift and on weekends, these crew leaders are the only supervisory personnel on plant. Craftsmen and health physics technicians assigned to the regular decommissioning crews report to the crafts supervisor and the senior health physics technician on the shift, respectively.

The number and specific makeup of crews required for a given decommissioning mode are discussed in the appendix detailing that mode (Appendix I, J, or K).

Detailed knowledge of and familiarity with the reference BWR increases the effectiveness of the decommissioning staff. Consequently, staff positions are filled with facility operations and maintenance personnel to the maximum extent possible. Specialty contractors and consultants are hired as needed to assist in areas outside the staff's expertise or capability.

This discussion is limited to the decommissioning staff, and does not include utility staff members involved in predecommissioning activities prior to final plant shutdown.

Key decommissioning staff members perform the functions described in the following subsections.

Decommissioning Superintendent

Directly responsible to corporate management, the superintendent coordinates and oversees all decommissioning activities. He directs the decommissioning engineer and the health and safety supervisor, as well as security, quality assurance, and contracts and accounting, to ensure the safety and cost-effectiveness of the decommissioning project. He provides necessary liaison with regulatory agencies and utility management.

Decommissioning Engineer

This individual plans, coordinates, and supervises the actual decommissioning tasks. He provides the engineering services and detailed procedures required to carry out the decommissioning plan safely and cost-effectively. He prepares all routine and special reports, as well as a chronological history of the project.

Assistant Decommissioning Engineer

This person supervises the decommissioning support personnel and assists the decommissioning engineer in developing detailed work procedures. He writes specifications for special equipment and tools that must be procured or fabricated. He also prepares reports requested by the decommissioning engineer.

Shift Engineer

Responsible for carrying out the actual decommissioning work during a shift, this person supervises the crew leader and craft supervisor. He reports to the decommissioning engineer. As he supervises the day-to-day performance of the shift, he recommends changes in procedures and schedules to improve the safety and/or cost-effectiveness of the project.

Crew Leader

Reporting to the shift engineer, this individual directs the work crews in the performance of the actual decommissioning tasks.

Craft Supervisor

This person is responsible for maintenance of essential plant equipment and services as well as for assigning craft labor to particular decommissioning tasks. He instructs craftsmen in their assigned tasks and ensures the availability of required tools and supplies.

Security Supervisor

This person is responsible for site security during decommissioning. He supervises the security personnel and, if necessary, provides liaison with offsite civil authorities. The security shift supervisor directs shift activities.

Contracts and Accounting Specialist

An experienced accountant, this individual is responsible for the financial aspects of the project. He prepares procurement documents and contracts and, with approval from the decommissioning superintendent and the decommissioning engineer, disburses funds. He maintains up-to-date financial accounts and provides the decommissioning superintendent with regular summary reports.

Health and Safety Supervisor

This person (typically a senior health physicist) recommends and enforces safety policy, both radiological and industrial. He advises the decommissioning superintendent on all safety matters. He maintains the occupational radiation exposure records, and also develops and implements the environmental survey (via a specialty contractor) and the emergency preparedness programs. He supervises and is assisted by the industrial safety specialist and the health physicist.

Health Physicist

Responsible for ensuring compliance with radiation work procedures, this individual directs the activities of the health physics technicians who monitor

all decommissioning activities, measure and record on-the-job radiation dose information, and operate the plant laboratory facilities, including sampling and analysis. The senior health physics technician assigns and trains the others on the shift.

Quality Assurance Supervisor

This person is responsible for preparing the quality assurance plan for decommissioning and works with the decommissioning engineer to implement it. To ensure the independence of the quality assurance program, he reports directly to corporate headquarters. He supervises a quality assurance unit, which maintains audit and job performance records and verifies that established safety review procedures are followed. (See Appendix G.5 for further discussion of quality assurance functions.)

Safety Review Committee

This committee advises corporate management (the utility's nuclear activities director) and the decommissioning superintendent on safety-related matters. It has six voting members: two from corporate headquarters and four independent consultants. The decommissioning superintendent, the quality assurance supervisor, the decommissioning engineer, and the health and safety supervisor are nonvoting members. Resolution of all issues requires agreement by a majority of the consultants. The committee meets about once a month during active decommissioning. The decommissioning superintendent implements the committee's decisions.

H.2 PLANNING AND PREPARATION

The planning and preparation phase of decommissioning is an important factor in the success of the project because it lays the groundwork for all of the subsequent activities. If planning and preparation is inadequate, the safety and/or cost-effectiveness of the project may be compromised.

Planning and preparation is carried out simultaneously with the last months of reactor operation. Without detailed study of the time required, it is postulated that the planning and preparation phase spans 2 years for either immediate dismantlement or entombment and 18 months for safe storage.

Planning and preparation includes the following concerns:

- regulatory requirements
- data gathering and analyses
- development of detailed work plans and procedures
- design, procurement, and testing of special equipment
- staff selection and training
- selection of specialty contractors
- removal of accumulated spent fuel and unneeded spent fuel storage racks
- install HEPA filters.

These are discussed in the following subsections.

H.2.1 Regulatory Requirements

Prior to the start of actual decommissioning tasks, the licensee must comply with applicable regulatory requirements. The current status of such requirements is detailed in Chapter 5 of Volume 1. This subsection summarizes the activities undertaken during planning and preparation to meet these requirements.

The major requirement is to provide the necessary documentation for amending the facility operating license to "possession-only" status and, if required, for obtaining an NRC dismantling order. In requesting an amended license, the licensee must provide:

- a description of the current facility status
- a inventory of the onsite radioactive materials
- a description of the proposed decommissioning activities
- a description of the proposed measures to prevent criticality and to minimize radioactive releases
- any proposed changes to the technical specifications (e.g., deletion of specifications relating solely to plant operation)
- safety analyses of both the proposed activities and the specification changes.

This information becomes the decommissioning plan for passive safe storage.

In addition to the amended license, an NRC dismantling order is required for immediate dismantlement or entombment. A request for such an order must include a dismantlement plan providing:

- a description of the ultimate facility status
- a description of the dismantling activities (including radioactive material disposal and site decontamination) and the associated environmental and safety precautions
- a safety analysis of the dismantlement and any resultant releases
- a safety analysis of the plant in its ultimate status.

In addition to the aforementioned documentation, the licensee must submit a radioactive waste handling plan, a quality assurance plan, an environmental report, and security and safeguards plans. Updated information concerning the financial qualification of the licensee may also be required. (See Section 5.1 of Volume 1 for further details.)

H.2.2 Data Gathering and Analyses

A large body of data is gathered and analyzed during the planning and preparation phase of decommissioning. These data support the regulatory requirements discussed above, particularly the inventory of radioactive materials and the various safety analyses. In addition, they provide the bases for planning decommissioning tasks and selecting appropriate methods and equipment.

Included in this activity is a comprehensive survey of radiation dose rates and contamination levels within the facility. This survey, taken after final reactor shutdown, provides information for determining decontamination and temporary shielding requirements. It also provides initial data on radiation dose rates likely to be encountered during the various decommissioning tasks.

H.2.3 Development of Detailed Work Plans and Procedures

Detailed work plans and procedures are developed based on the information gathered during data gathering and resultant analyses and provided to the NRC

with the license amendment/dismantling order request. These detailed plans and procedures contain all the information required to actually carry out the decommissioning tasks. They address the following items:

- decommissioning methods
- schedules and sequences of events
- radioactive waste management
- contamination control
- radiological and industrial safety
- equipment requirements.

Quality assurance, security, and environmental constraints are also considered. The plans and procedures cover all aspects of the decommissioning project.

H.2.4 Design, Procurement, and Testing of Special Equipment

Any special equipment required to complete the decommissioning project is identified during planning and preparation. Designs and specifications are prepared for each item required. When the item is procured, it is inspected to verify that it meets specifications and complies with applicable QA and safety requirements. It is then tested to ensure that it performs as required. The testing also serves to train personnel in the use of the equipment and to provide pertinent data on its operation.

H.2.5 Staff Selection and Training

At the start of planning and preparation, a decommissioning organization is created within the utility. Staffing requirements are identified, and critical positions are filled with key engineering and operating personnel. The personnel are trained as required to fulfill their roles in the organization; special emphasis is given to the use of new and unique equipment and procedures. The decommissioning staff is discussed in detail in Section H.1.

H.2.6 Selection of Specialty Contractors

During planning and preparation, the decommissioning staff identifies the needs for any specialty contractors required to decommission the facility. These contractors perform unique services outside of the expertise or capability

of the staff. After the needs are identified, contractors are invited to bid on the required work packages. Contractual agreements are concluded prior to the start of the actual decommissioning, if possible, to ensure the uninterrupted completion of the project. Specialty contractor requirements are discussed in Section H.3.

H.2.7 Removal of Accumulated Spent Fuel And Unneeded Spent Fuel Storage Racks

Any spent fuel stored from previous refueling activities is removed from the spent fuel storage pool and shipped to a repository. In addition, for immediate dismantlement and entombment, the spent fuel storage racks in excess of those required for final reactor defueling are removed and shipped offsite. By removing these excess items prior to the start of the actual decommissioning, extra space is made available in the spent fuel storage pool for interim storage and packaging of activated materials removed from the facility.

H.2.8 Install HEPA Filters

Prior to the start of the actual decommissioning tasks (for all decommissioning modes, for the sake of comparison in the safety assessment), HEPA filters are installed outboard of the blowers in the HVAC exhaust systems of the Reactor Building and the Turbine Generator Building. (The Radwaste and Control Building HVAC system is already equipped with HEPA filters; see Section C.4.3.) These filters are installed to prevent the atmospheric release of airborne radioactivity generated during the decommissioning tasks, because many tasks are expected to generate airborne contamination inside the facility exceeding that produced during normal plant operation.

H.3 SPECIALTY CONTRACTORS

During decommissioning, specialty contractors are employed to provide services beyond the capability of the utility's decommissioning staff. Use of these contractors increases the overall cost-effectiveness of the project by improving the efficiency of specialty operations and reducing the need for specialized staff training. In addition, specialized experience gained from similar projects is directly applied to the decommissioning by these contractors, eliminating the mistakes and wasted effort inherent in learn-as-you-go situations.

The specialty contractors used during immediate dismantlement of the reference BWR are:

- environmental monitoring specialists, for implementing the environmental surveillance program outlined in Section G.6
- explosive specialists, for breaking up the sacrificial shield
- hauling contractors, for transport of packaged radioactive materials to a disposal site
- temporary radwaste handling and solidification support, for radwaste handling and final cleanup after the installed radwaste handling systems are decontaminated.

If, following dismantlement, the facility is demolished and the site restored, demolition and landscaping contractors are also required.

For safe storage and entombment, explosive specialists are not required, but the rest of the specialty contractors are the same as for dismantlement.

H.4 ESSENTIAL SYSTEMS AND SERVICES

All or parts of certain facility systems and services must remain in place and in service until all radioactive material is either removed from the facility or secured on the site, to prevent the release of significant quantities of radionuclides (or other hazardous materials) to the environment. Some systems and services are required for cleanup and disassembly activities. Other systems provide personnel health and safety protection. The required systems and services are listed in Table H.4-1, together with the justification for retaining each.

As areas within the facility are readied for demolition or secured for storage, the essential systems and services in these areas are deactivated and, if contaminated, removed as required. Continuous service to the remaining work areas is maintained as long as necessary.

TABLE H.4-1. Systems and Services Required During Decommissioning

<u>System or Service</u>	<u>Justification</u>
Electrical Power	Operation of electrical equipment including HVAC, lighting, and radiation monitoring
HVAC Systems	Ventilation and radioactive contamination confinement
Condensate Supply System	Water supply to spent fuel pool and radwaste systems
Plant Makeup Water Treatment System	Decontamination, cleanup, fire protection, and potable water
Fire Protection System	Health and safety
Compressed Air Systems (control and service)	Operation of pneumatic controls and tools; personnel fresh air supply
Communications Systems	Facilitate and coordinate decommissioning activities
Radiation Monitoring Systems	Personnel safety considerations
Radwaste Systems	Treatment of radioactive liquids and solids
Fuel Pool Cooling and Cleanup System	Cleanup and cooling of water in spent fuel storage pool while spent fuel is there; in reactor well and dryer and separator pool during defueling and reactor vessel/internals removal
Reactor Building Closed Cooling Water System	Secondary cooling of other systems
Plant Service Water System	Air compressor cooling
Auxiliary Boiler	HVAC heating and radwaste concentration
Chemical Feed System	Radwaste handling and water demineralization
Fuel Oil System	Auxiliary boiler operation
Security Systems	Public safety and plant protection considerations.

For decommissioning by passive safe storage or entombment, certain systems and services are required during the continuing care period. These are:

- electrical power
- radiation monitoring systems
- security systems.

Requirements for the continuing care period of passive safe storage are discussed further in Appendix J, and those for entombment in Appendix K.

H.5 SYSTEM DECONTAMINATION AND DRAINING AND RADIOACTIVE LIQUID HANDLING

This section presents information concerning decontamination of selected systems and components, draining of contaminated water systems, and handling of the resultant radioactive liquids for the reference BWR. The details, time and manpower requirements, costs, and occupational radiation doses for these decommissioning activities are presented.

Since these activities are assumed to be undertaken regardless of the chosen decommissioning mode, the information in this section is presented in a manner in which it can be readily applied to each decommissioning mode.

H.5.1 Details of System Decontamination and Draining and Radioactive Liquid Handling

Upon final shutdown of the reference BWR, significant quantities of radioactive contamination are expected in certain piping and equipment systems in the Reactor Building, the Turbine Generator Building, and the Radwaste and Control Building (see Appendices C, D, and E). This contamination (both deposited on the internal surfaces and included in the contents of these systems), if allowed to remain, represents a hazard to the decommissioning personnel and the environment and a deterrent to effective completion of the decommissioning activities. The purpose of the decontamination, draining, and subsequent radioactive liquid handling activities is to minimize the hazard potential and to allow decommissioning to proceed on a predetermined schedule and in the most cost-effective manner. Each of these activities is discussed more fully in subsequent subsections.

H.5.1.1 Key Bases and Assumptions

For the chemical decontamination information in this section, a number of assumptions are made to establish a viable decontamination scenario for the reference BWR.

These are:

- System and component selection is based primarily on the equipment and piping radiation dose rate data represented in Appendix D.
- The methods used to accomplish the decontamination utilize presently available technology.

- The decontamination achieves a minimum decontamination factor (DF) of 10.
- The decontamination effort is carried out within the framework of existing federal regulations.
- ALARA principles of minimizing exposure to radiation are used in developing the work schedules and methods.
- The planning and preparations for decontamination of the selected systems and components takes place before the final plant shutdown, making it possible for the necessary approvals to be in place by the time the reactor is defueled.
- No water rinse is required following chemical decontamination.

A detailed, plant-specific, cost-benefit analysis is necessary to determine if a water flush following chemical-decontamination is cost-effective. The analysis should include the following considerations:

- history and results of past plant chemical decontaminations
- required amount of water, and associated costs incurred for the additional liquid radwaste management
- additional time, labor, and occupational radiation dose required
- industrial safety impacts on subsequent decommissioning activities (e.g., toxicity of residual chemicals)
- effect on overall DF.

Normally, a rinse and drain procedure is repeated as indicated by chemical analysis to assure removal of chemical solvent from a system that is to be returned to operational status. For this study, a decontamination factor of 10 is assumed to be achieved without a water flush. In addition, since removal of the anticipated minimal amounts of residual chemical cleaning solvent is not expected to present an industrial safety hazard during subsequent decommissioning activities, a water flush is assumed not a cost-effective decommissioning activity for the reference BWR.

The results obtained are specific to these bases and assumptions, and to other assumptions specifically stated throughout this section. Application of

these results to specific situations where conditions are different from those assumed in this study could produce different conclusions.

H.5.1.2 System Decontamination

The basic methods that can be used to decontaminate contaminated system surfaces are described in Section G.1 of Appendix G. The two methods that are selected for system decontamination activities at the reference BWR are described in detail in this subsection. They are:

- chemical decontamination (recirculatory and single-pass methods)
- water-jet cleaning.

Chemical Decontamination. Chemical decontamination is considered for selected systems or components in the reference BWR that contain deposited contamination representing a radiation dose rate hazard once the system or component is drained and dried for further decommissioning effort. The systems or components to be chemically decontaminated are selected on the following bases in descending order of importance:

- expected contact radiation dose rate after draining (see Section D.1 of Appendix D and Section E.2 of Appendix E); systems or components with expected readings <15 mR/hr not considered
- flow capability
- operational heating capability
- size.

The three categories of systems or components within the reference BWR that are selected for either recirculatory or single-pass chemical decontamination, together with their decontamination flow capabilities, are: 1) six reactor piping systems located primarily in the Reactor Building--recirculatory (some with piping jumpers); 2) the three contaminated drain systems located in the Reactor Building, the Turbine Generator Building, and the Radwaste and Control Building--single-pass (with mobile chemical mixing and heating units); and 3) thirty-two pieces of equipment in the liquid and solid radwaste systems located in the Radwaste and Control Building--recirculatory (with mobile chemical decontamination units described in Section G.1 of Appendix G).

The six reactor systems are depicted in Figure H.5-1 and include the reactor water recirculation (RRC) system, the reactor water cleanup (RWCU) system, the residual heat removal (RHR) system, the low-pressure core spray (LPCS) system, the high-pressure core spray (HPCS) system, and the fuel pool cooling and cleanup (FPC) system. The latter four systems require special piping jumpers to complete recirculation loops. The recirculation loop for each system may or may not include all the system piping. The reactor core isolation cooling (RCIC) system is not selected for chemical decontamination because it requires nuclear steam for operation of the steam-driven pump and none is available following final reactor shutdown.

While too extensive to judicially identify in a single figure, the three contaminated drain piping systems--equipment drain (radioactive) [EDR], floor drain (radioactive) [FDR], and miscellaneous waste (radioactive) [MWR] --are depicted in Figure C.5-1 of Appendix C from the collection pumps or tanks onward.

The equipment pieces in the liquid and solid radwaste systems that are selected for chemical decontamination are listed in Table H.5-1 and also are depicted in Figure C.5-1. Because of the assumed lack of operational heating capability to achieve and maintain the proper solution temperature and because of the large volumes involved, these systems are assumed to be drained to a minimum working volume of water prior to decontamination. This volume is assumed to be too small to prime the system pumps for recirculation through the system piping. Therefore, each individual equipment piece is assumed to be isolated from the piping and chemically decontaminated in a recirculatory loop formed with a mobile chemical decontamination unit. A special spray nozzle connection is required for a piece of equipment that exceeds the volumetric capacity of the mobile chemical decontamination unit.

Some contaminated systems in the reference BWR, for example, the main steam system, the condensate (nuclear steam) system, and the reactor feedwater system, are extremely difficult to isolate for recirculatory chemical decontamination. In addition, some equipment is not amenable to individual decontamination because of complex design or other detrimental reasons. In such cases, ALARA principles must be judiciously applied and all alternative dismantlement methods should be considered to ensure optimization of costs versus anticipated occupational exposure during decommissioning activities.

FIGURE H.5-1. Reference BWR Reactor Piping Systems Selected for Chemical Decontamination

TABLE H.5-1. Reference BWR Liquid and Solid Radwaste Equipment
Selected for Chemical Decontamination^(a)

SYSTEM	
EKN ^(b)	Equipment Piece (Quantity)
<hr/> Condensate Filter Demineralizer (CPR) <hr/>	
302	Condensate Phase Separator (2)
304	Condensate Backwash Receiving Tank (1)
<hr/> Equipment Drain (Radioactive) (EDR) <hr/>	
301	Waste Collector Tank (1)
305	Spent Resin Tank (1)
312	Waste Surge Tank (1)
313	Waste Sample Tank (2)
<hr/> Floor Drain (Radioactive) (FDR) <hr/>	
300	Floor Drain Collector Tank (1)
306	Waste Sludge Phase Separator (1)
314	Floor Drain Sample Tank (1)
<hr/> Miscellaneous Waste (Radioactive) (MWR) <hr/>	
308	Chemical Waste Tank (2)
309	Distillate Tank (2)
310	Detergent Drain Tank (2)
311	Decontamination Solution Concentrator Waste Tank (2)
386	Decontamination Solution Concentrator (2)
387	Decontamination Solution Concentrator Heating Element (2)
408	Concentrator Waste Measuring Tank (1)
457	Decontamination Solution Concentrator Condenser (2)
<hr/> Process Waste (Radioactive) (PWR) <hr/>	
379	Hopper Mixer (2)
409	Centrifuge (2)
<hr/> Reactor Water Cleanup (RWCU) <hr/>	
307	Cleanup Phase Separator (2)

(a) Recirculatory chemical decontamination is achieved with the mobile chemical decontamination units described in Section G.1 of Appendix G.

(b) EKN = Equipment Key Number (see Section C.2 of Appendix C).

The chemical decontamination methods applicable to the system flow capabilities are shown in Table G.1-1 in Appendix G. The two selected for the reference BWR are:

- recirculatory decontamination using 5-wt% EDTA/citrox solution
- single-pass decontamination using 10-wt% phosphoric acid solution, which is subsequently neutralized with sodium hydroxide.

The recirculatory method is used where the chemical solution can be recirculated until the desired degree of decontamination is obtained. The single-pass method completes the decontamination in one pass and is used where recirculation is impractical.

The estimated required volumes of chemical decontamination solutions for the reference BWR are listed in Table H.5-2. The estimated total volume of the recirculatory chemical decontamination solution is 978 m³, and the neutralized single-pass chemical decontamination solutions total 79 m³. The latter volume includes 66 m³ of decontamination solution at 10-wt% concentration of phosphoric acid and 13 m³ of neutralizing solution at 50-wt% concentration of sodium hydroxide. Final disposition of these chemical decontamination solutions is described in Section H.5.1.4, Radioactive Liquid Handling.

Water-Jet Cleaning. A description of water-jet cleaning is given in Section G.1.2 of Appendix G and is not repeated here. High-pressure water jets are assumed used to clean the surfaces of the following:

- suppression chamber (EKN 2 and EKN 3, reference)
- reactor well pool cavity (EKN 63)
- dryer separator pool cavity (EKN 112)
- spent fuel pool cavity (EKN 90)
- the internal surfaces of the condensate storage tanks (EKN 228) and the main condenser (EKN 200)
- the external surfaces of the liquid and solid radwaste systems equipment listed in Table H.5-1.

TABLE H.5-2. Estimated Volumes of Chemical
Decontamination Solutions

<u>Solution Category</u>	<u>Estimated Volumes (m³)</u>
<u>Source</u>	
<u>5-wt% EDTA/Citrox (Recirculatory)</u>	
<u>Reactor Piping Systems^(a)</u>	
RRC System	614 ^(b)
RWCU System	11
RHR System	157
LPCS System	13
HPCS System	16
FPC System	12
<u>Liquid and Solid Radwaste Systems Equipment^(c)</u>	
CPR System	24
EDR System	54
FDR System	22
MWR System	36
PWR System	15
RWCU System	4
Total, 5-wt% EDTA/Citrox	978
<u>10-wt% Phosphoric Acid (Single-Pass)</u>	
<u>Contaminated Drain System^(a)</u>	
Reactor Building (EDR and FDR systems)	22 ^(d)
Turbine Generator Building (EDR and FDR Systems)	6 ^(d)
Radwaste and Control Building (EDR, FDR, and MWR Systems)	51 ^(d)
Total, 10-wt% Phosphoric Acid	79 ^(d)

(a)Volumes based on data contained in Tables C.3-7 through C.3-10 in Appendix C.

(b)Includes the reactor vessel volume.

(c)Volumes based on either the minimum volume to maintain pump prime in the case of small-size components or on approximately 1/10 the volume of larger components in that case.

(d)Includes 84% decontamination solution and 16% neutralizing solution (sodium hydroxide at 50-wt% concentration).

In general, the water-jet decontamination activity proceeds concurrently with draining the particular volume of contaminated water. However, for effective decontamination, the internal surfaces of the main condenser are water-jet cleaned as they become exposed during disassembly.

The estimated volumes of contaminated water from water-jet cleaning activities are listed by building and by source in Table H.5-3. The total volume of contaminated water generated by these activities is 1,503 m³. Final disposition of this contaminated water is described in Section H.5.1.4, Radioactive Liquid Handling.

TABLE H.5-3. Estimated Volumes of Contaminated Water from Water-Jet Cleaning

Building EKN(a)	Source	Estimated Volumes (m ³)
<u>Reactor Building</u>		
2,3(Reference)	Suppression Chamber	49
63	Reactor Well Pool Cavity	5
112	Dryer and Separator Pool Cavity	16
90	Spent Fuel Pool Cavity	18
<u>Turbine Generator Building</u>		
228	Condensate Storage Tanks	52
200	Main Condensor	1 285
<u>Radwaste and Control Building</u>		
--(b)	Liquid and solid radwaste systems equipment	68
Total		1 503

(a)EKN = Equipment Key Number (see Section C.2 of Appendix C).

(b)See Table H.5-1 for equipment key numbers.

H.5.1.3 System Draining

Contaminated water remains in those reference BWR water systems discussed in Appendix C and not selected for chemical decontamination. This water must be drained before further decommissioning of these systems can proceed. In addition, contaminated water from miscellaneous sources is assumed to be generated at the estimated rate of $4.5 \text{ m}^3/\text{day}$ over the entire active decommissioning period.

Table H.5-4 lists the contaminated water sources and the estimated volumes. Submersible pumps, with cartridge filters in their effluent lines, are required to drain the suppression pool and the spent fuel pool. The estimated total volume of contaminated water to be drained is $15,370 \text{ m}^3$, including $2,970 \text{ m}^3$ from the miscellaneous sources. Final disposition of the contaminated water from system draining and miscellaneous sources is described in the following subsection.

H.5.1.4 Radioactive Liquid Handling

The chemical decontamination solutions and the contaminated water are drained to the liquid radwaste system (see Section C.5.1 of Appendix C) or to a temporary portable radwaste system. No attempt is made to identify the solution volumes that go to the temporary radwaste system, as this depends on the particular decommissioning mode. The data presented herein and in later sections assume the use of the system provided in the reference BWR.

To minimize potential plateout and redeposition of radioactive contaminants, the hotter-than-ambient (353 K) recirculatory chemical decontamination solutions are assumed to be drained and processed as rapidly as practicable. Prompt radwaste processing, starting simultaneously with system or component draining operations, optimize these decommissioning tasks, and is assumed throughout the chemical decontamination operations. Expedient radwaste processing avoids the need for additional liquid waste storage tanks. All systems designed to control the release of hazardous material to the environment or to the noncontaminated portions of the facility are in operation during the decontamination and draining tasks and the subsequent liquid radwaste processing.

TABLE H.5-4. Estimated Volumes of Contaminated Water from System Draining Activities

Building EKN(a)	Source	Estimated Volumes (m ³)
<u>Reactor Building</u>		
2,3(Reference)	Suppression Pool	3 890
63	Reactor Well Pool	796
112	Dryer and Separator Pool	1 110
90	Spent Fuel Pool	1 326
--(b)	Contaminated Systems:(c)	
	Control Rod Drive	23
	Condensate (Nuclear Steam)	24
	Reactor Building Closed Cooling Water	26
	Reactor Core Isolation Cooling	7
	Reactor Feedwater	15
Subtotal, Reactor Building		7 217
<u>Turbine Generator Building</u>		
228	Condensate Storage Tanks	3 030
200	Main Condenser	617
--	Contaminated Systems:(c)	
	Condensate (Nuclear Steam)	173
	Condensate Filter Demineralizer	1
	Reactor Feedwater	60
Subtotal, Turbine Generator Building		3 881
<u>Radwaste and Control Building</u>		
--	Liquid Radwaste Systems Equipment ^(d)	1 197
--	Contaminated Systems:(c)	
	Condensate (Nuclear Steam)	99
	Condensate Filter Demineralizer	6
Subtotal, Radwaste and Control Building		1 302
<u>Reactor Building, Turbine Generator Building, and Radwaste and Control Building</u>		
--	Drains ^(e)	2 970
Total		15 370

(a)EKN = Equipment Key Number (see Section C.2 of Appendix C).

(b)Implies "too numerous to list."

(c)Volumes calculated based on data contained in Tables C.3-7 through C.3-10 in Appendix C.

(d)Volume based on approximately 90% of original volumes drained before chemical decontamination of the equipment listed in Table H.5-1.

(e)Volume from miscellaneous sources based on a generation rate of 4.5 m³ per day over the active decommissioning period.

The chemical decontamination solutions that contain EDTA/citrox are processed by the miscellaneous waste (radioactive) subsystem at the rate of $2.3 \text{ m}^3/\text{hr}$. The concentrator distillate is sent to the river; the concentrator bottoms go to the process waste (radioactive) system for solidification, where they are mixed with cement and encapsulated in steel cask liners. The processing of these chemical decontamination solutions produces an estimated 438 m^3 of concentrator bottoms, which result in 517 m^3 of concreted waste and occupy 225 steel cask liners. Of these containers, 52 are assumed to have a maximum surface radiation dose rate of 50 R/hr, 108 are assumed to have a maximum surface dose rate of 1 R/hr, and 65 are assumed to have a maximum surface dose rate of 0.2 R/hr. The average radioactivity content for each container is 600 Ci in the first group, 12 Ci in the second group, and 2.4 Ci in the last group.

The chemical decontamination solutions containing phosphoric acid are assumed to be neutralized with sodium hydroxide. The neutralizing process results in a solution with an estimated 12-wt% solids concentration, which is the assumed concentrator effluent limit. Therefore, the neutralized solution is assumed to go directly to the process waste (radioactive) system for solidification. The solidification process results in approximately 95 m^3 of concreted waste, which occupies 42 steel cask liners. Each of these containers is assumed to have a maximum surface radiation dose rate of 0.2 R/hr and an average radioactivity content of 2.4 Ci.

The contaminated water is processed either by the equipment drain (radioactive) subsystem or the floor drain (radioactive) subsystems, at a total combined rate of $43.56 \text{ m}^3/\text{day}$. The effluent water is sent to the river; the resultant radioactive filter sludges and spent demineralizer resins go to the process waste (radioactive) system, where they are mixed with cement and encapsulated in steel cask liners. The radioactive filter sludges and demineralizer resins are produced at an estimated rate of $0.25 \text{ m}^3/100 \text{ m}^3$ of contaminated water processed. The total volume of filter sludges and resins requiring disposal is estimated to be 42 m^3 . This volume is solidified and packaged in 19 steel cask liners. Each of these containers is assumed to have a maximum surface radiation dose rate of 1 R/hr and an average radioactivity content of 12 Ci.

H.5.2 Time and Manpower Requirements for System Decontamination and Draining and Radioactive Liquid Handling

This section contains information concerning the time and manpower requirements for system decontamination and draining and radioactive liquid handling. Discrete tasks are defined for system decontamination and draining, which also include the handling requirements for the radioactive liquids from these tasks. After the time and shiftwise manpower requirements for these tasks are defined, they are integrated into each decommissioning mode's schedule to determine their sequence and schedule and their total manpower requirements for each mode. The resultant task schedules and total manpower requirements are not shown here, but are shown as integral parts of the schedules and manpower requirements presented in Appendices I, J, and K.

H.5.2.1 Time Requirements

The estimated minimum times required to complete the system decontamination and draining tasks are presented in Table H.5-5. The tasks are defined based on the building in which they are performed, their procedural and operational requirements, and their prerequisites. They are listed in Table H.5-5 according to the first two definition bases. The time requirements are determined based on the following assumptions:

- Tasks are performed 24 hours per day, 7 days per week.
- Tasks include all activities required to set up and operate the system, as well as to drain and process the radioactive liquids.
- Recirculatory chemical decontamination takes 2 days of loop operation at the proper temperature (353 K) to achieve the desired DF.
- Single-pass chemical decontamination solutions are neutralized prior to further processing.
- Water-jet cleaning proceeds simultaneously with the other portion of the task to which it is assigned, except in the case of the condenser where it is performed as surfaces become exposed during the dismantling process.
- Draining times and processing times are based on the processing rates of the in-house liquid radwaste systems and the volumes of liquid presented in the previous section.

- The equipment drain (radioactive) and floor drain (radioactive) subsystems are unavailable for about 5 hours per week of operation for filter back-washes and demineralizer resin changeouts.
- The miscellaneous drain (radioactive) subsystem is unavailable for 2 days per week of operation for maintenance of the concentrators.

TABLE H.5-5. Time Requirements for System Decontamination and Draining and Radioactive Liquid Handling

<u>Building</u>	<u>Task</u>	<u>Estimated Minimum Times Required (days)(a)</u>
<u>Reactor Building/Primary Containment</u>		
	Chemical Decon Reactor Water Recirculation and Cleanup Systems	22
	Chemical Decon Residual Heat Removal, Low- and High-Pressure Core Spray Systems	21
	Chemical Decon Fuel Pool Cooling and Cleanup System	8
	Chemical Decon Drain Systems	10
	Drain Suppression Pool to Radwaste; Water-Jet Clean	96
	Drain Reactor Well Pool to Radwaste; Water-Jet Clean	20
	Drain Dryer and Separator Pool to Radwaste; Water-Jet Clean	28
	Drain Spent Fuel Pool to Radwaste; Water-Jet Clean	34
	Drain Contaminated Systems to Radwaste	10
<u>Turbine Generator Building</u>		
	Drain Contaminated Systems to Radwaste; Water-Jet Clean	81
	Condensate Storage Tanks	47
	Drain Condenser to Radwaste; Water-Jet Clean	5
	Chemical Decon Drain Systems	5
<u>Radwaste and Control Building</u>		
	Chemical Decon Equipment; Water-Jet Clean	50
	Chemical Decon Drain Systems	23

(a)Includes time for system set-up, operation, and draining; radioactive liquid handling; and system unavailability; based on the processing capabilities of the in-house liquid and solid radwaste systems.

H.5.2.2 Manpower Requirements

The estimated manpower requirements for the general decontamination, draining, and radioactive liquid handling tasks are shown in Table H.5-6. Many work segments for the various tasks are similar in nature; however, the general tasks are sufficiently different from each other that the optimum task crews vary in both makeup and size.

TABLE H.5-6. Manpower Requirements for System Decontamination and Draining and Radioactive Liquid Handling^(a)

General Task	<u>Estimated Manpower Requirements (number/shift)^(b)</u>				
	<u>Crew Leader</u>	<u>Utility Operator</u>	<u>Laborer</u>	<u>Craftsman</u>	<u>Health Physics Technician</u>
Chemical Decon Systems	1	2	-- ^(c)	1	1
Chemical Decon Equipment	1	2	--	1 ^(d)	1
Chemical Decon Drain Systems	--	4	4	2	--
Water-Jet Clean	1	2	--	--	1
Drain to Radwaste ^(e)	--	1	--	1	1
Radioactive Liquid Handling	1	4	--	1	1

(a) Those required to perform the given task, but not to satisfy the occupational dose commitment for that task; may not be required full-time.

(b) Three shifts per day, 7 days per week; a shift engineer is assumed available on regular shifts and on-call during off-shifts and weekends for emergencies.

(c) Indicates "not required."

(d) Requires the part-time services of craftsmen from different crafts.

(e) Staffed by personnel from the radioactive liquid handling crew.

H.5.3 Costs for System Decontamination and Draining and Radioactive Liquid Handling

The costs for system decontamination and draining and radioactive liquid handling that are independent of the particular decommissioning mode are presented in this section. These are costs for materials, special equipment, energy, and disposal of wet solid wastes. Manpower costs depend on the schedule and manpower loading for a particular decommissioning mode and are included in Appendices I, J, or K as they pertain to the various decommissioning modes.

H.5.3.1 Costs for Materials

The materials that are required for system decontamination and draining and radioactive liquid handling include chemicals and neutralizing agent for chemical decontamination and resins for contaminated water handling. Table H.5-7 lists the costs for these materials. The costs for solidification materials for the wet solid wastes are included later as part of the costs for disposal of these wastes.

TABLE H.5-7. Costs for Materials for System Decontamination and Draining and Radioactive Liquid Handling

<u>Material</u>	<u>Estimated Amounts Required</u>	<u>Total Costs (\$) (a)</u>
EDTA/Citrox Acid	48 830 kg ^(b)	309 600
Phosphoric Acid	5 900 kg ^(c)	22 830
Sodium Hydroxide	13.5 m ³ ^(d)	2 430
Filter/Demineralizer Resins	7 750 kg bead/ 4 130 kg Powdex ^(e)	89 040
Total Cost		423 900

(a)Based on Section M.6 of Appendix M; rounded to the nearest \$10.

(b)Calculated at 5-wt% concentration in 978 m³ of water.

(c)Calculated at 10-wt% concentration in 61 m³ of water.

(d)Calculated at one part sodium hydroxide to five parts phosphoric acid.

(e)Based on information supplied by Washington Public Power Supply System.

H.5.3.2 Costs for Special Equipment

Table H.5-8 shows the costs for special equipment that is assumed required for system decontamination and draining. No special equipment is required for radioactive liquid handling.

TABLE H.5-8. Costs for Special Equipment for System Decontamination and Draining

<u>Special Equipment</u>	<u>Estimated Number Required</u>	<u>Estimated Costs (\$) (a)</u>
Piping Jumper	4	2 500
Mobile Chemical Mixing and Heating Unit	4	10 000
Mobile Chemical Decontamination Unit	5	100 000
High-Pressure Water Jet	2	20 000
<u>Submersible Pump with Disposable Filter</u>	5	<u>5 000</u>
Total Cost		137 500

H.5.3.3 Costs for Energy

The costs for the energy used during system decontamination and draining and radioactive liquid handling are given in Table H.5-9. Electricity and fuel oil are the two energy forms used. In the reference BWR, fuel oil is used in the auxiliary boiler to generate steam for use in the concentrators in the miscellaneous waste (radioactive) subsystem.

TABLE H.5-9. Costs for Energy for System Decontamination and Draining and Radioactive Liquid Handling

<u>Energy Form</u> <u>General Task</u>	<u>Estimated</u> <u>Energy Usage</u> ^(a)	<u>Total</u> <u>Costs (\$)</u> ^(b)
<u>Electricity</u>	<u>(MWh)</u>	
Chemical Decon Systems	1 540	23 100
Chemical Decon Equipment	50	750
Chemical Decon Drain Systems	20	300
Water-Jet Clean	70	1 050
Drain to Radwaste	400	6 000
<u>Radioactive Liquid Handling</u>	<u>510</u>	<u>7 650</u>
<u>Subtotals, Electricity</u>	<u>2 590</u>	<u>38 850</u>
<u>Fuel Oil</u>	<u>(m³)</u>	
<u>Radioactive Liquid Handling</u>	<u>195</u>	<u>25 790</u>
Total Cost		64 640

(a)Based on a system-wise analysis of specific requirements; rounded to the nearest 10 MWh for electricity and to the nearest whole m³ for fuel oil.

(b)Based on Section M.6 of Appendix M.

H.5.3.4 Costs for Disposal of Wet Solid Wastes

Table H.5-10 gives a breakdown of the disposal costs for all of the wet wastes generated during system decontamination and draining and radioactive liquid handling. The wet solid wastes include the concentrator bottoms, the neutralized drain decontamination solutions, and the filter sludges and spent demineralizer resins discussed in Section H.5.1.4. An estimated total of 286 truckloads are shipped to a shallow-land burial site. The wet solid wastes occupy an estimated 814 m³ of space at the burial site. The total disposal cost for the wet solid wastes is estimated to be \$1,073,440.

TABLE H.5-10. Costs for Disposal of Wet Solid Wastes

Material	Assumed Surface Radiation Dose Rates (R/hr)	Number of Disposable Containers(a)	Container and Solidification Costs (\$)(b)	Cask Rental Costs (\$)(c)	Number of Shipments	Transportation Costs (\$)(d)	Handling Costs (\$)(e)	Burial Volumes (m ³)(f)	Burial Costs (\$)(g)	Liner Surcharges (\$)(e)	Total Disposal Costs (\$)(h)
Concentrator Bottoms	50	52	39 730	26 000	52	99 840	10 400	148	13 850	24 700	214 520
	1	108	82 510	54 000	108	207 360	21 600	307	28 730	5 400	399 600
	0.2	65	9 660	32 500	65	124 800	13 000	185	17 310	0	237 770
Neutralized Drain Decontamination Solutions	0.2	42	32 090	21 000	42	80 640	8 400	120	11 230	0	153 360
Filter Sludge and Spent Demineralizer Resins	1	19	12 900	9 500	19	36 480	3 800	54	5 060	950	68 690
Totals (h)		286	216 890	143 000	286	549 120	57 200	814	76 180	31 050	1 073 440

(a) Assumed to be 2.84 m³ steel cask liners

(b) Based on Table M.2-1 in Appendix M, includes solidification costs based on Section M.2.2, rounded to next highest \$10

(c) Based on Table M.3-1 in Appendix M, assumed used for all shipments for public safety

(d) Based on Table M.4-4 in Appendix M

(e) Based on Table M.5-1 in Appendix M

(f) Includes the disposable container, rounded to next whole m³

(g) Based on Table M.5-1 in Appendix M, rounded to next highest \$10

(h) The number of figures shown is for computational completeness and does not imply accuracy to that many significant figures

H.5.4 External Occupational Radiation Dose for System Decontamination and Draining and Radioactive Liquid Handling

Detailed estimates are made of the external occupational radiation doses that are accumulated by the decommissioning workers during system decontamination and draining and radioactive liquid handling tasks. These estimates are based on a task-by-task analysis of personnel exposure hours and expected radiation dose rates associated with each task. The results of these analyses are presented in Table H.5-11. Since the schedule and sequence of these tasks are not presented in this section, the doses in Table H.5-11 are not corrected for radioactive decay.

As can be seen from the table, one of the largest radiation doses is received by craftsmen during the ancillary task titled "Operate Radwaste Systems." This 21.6-man-rem dose is estimated for maintenance on the decontamination solution concentrators (EKN 386). Based on operating experience, the availability for concentrations in radwaste treatment systems has averaged in the range of 60 to 80%.⁽¹⁾ The unavailability is attributed to scaling, fouling of surfaces, instrumentation failures, corrosion, and other occasional upsets that result in high carry-overs and require system cleaning.

H.6 EQUIPMENT DEACTIVATION AND CONTAMINATED AREA ISOLATION

When equipment and systems are no longer needed during decommissioning, they are placed in a condition that provides maximum safety with minimum maintenance. When possible, equipment is left in a condition that permits startup or salvage at a later date. Deactivation and isolation techniques include closing and securing installed valves, installing blank flanges as required, and disconnecting electrical power and other utilities. Equipment deactivation procedures are coordinated with system decontamination and draining operations. In some instances decontamination must be carried out before equipment deactivation, while in others the opposite approach may be necessary. A safety audit of all systems is performed to ensure that all flammable and other potentially hazardous materials are removed. All deactivated equipment and systems are tagged out.

TABLE H.5-11. Estimated External Occupational Radiation Doses for System Decontamination and Draining and Radioactive Liquid Handling

Location	Task	Supervisors (a)		Utility Operators and Laborers		Craftsmen		Health Physics Technicians		Task Totals (b)	
		Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)
Reactor Building/Primary Containment											
	Chemical Decon Reactor Water Recirculation and Cleanup Systems	4	0.200	572	5.143	8	0.760	36	1.836	620	7.939
	Chemical Decon Residual Heat Removal, Low- and High-Pressure Core Spray Systems	12	0.360	259	4.403	20	1.100	118	3.540	409	9.403
	Chemical Decon Fuel Pool Cooling and Cleanup System	2	0.010	52	0.312	2	0.020	18	0.108	74	0.450
	Chemical Decon Drain Systems	--(c)	--	431	0.431	108	0.108	--	--	539	0.539
	Drain Suppression Pool to Radwaste; Water-Jet Clean	27	0.189	322	1.932	27	0.135	27	0.135	403	2.391
	Drain Reactor Well Pool to Radwaste; Water-Jet Clean	3	0.015	55	0.275	5	0.025	5	0.025	68	0.340
	Drain Dryer and Separator Pool to Radwaste; Water-Jet Clean	9	0.045	97	0.485	8	0.040	8	0.040	122	0.610
	Drain Spent Fuel Pool to Radwaste; Water-Jet Clean	9	0.045	109	0.545	9	0.045	9	0.045	136	0.680
	Drain Contaminated Systems to Radwaste	--	--	12	0.180	4	0.060	2	0.030	18	0.270
Turbine Generator Building											
	Drain Contaminated Systems to Radwaste; Water-Jet Clean Condensate Storage Tanks	8	0.008	51	0.051	--	--	5	0.005	64	0.064
	Drain Condenser to Radwaste; Water-Jet Clean	1	0.001	10	0.010	--	--	1	0.001	12	0.012
	Chemical Decon Drain Systems	--	--	305	0.305	76	0.076	--	--	381	0.381
Radwaste and Control Building											
	Chemical Decon Equipment; Water-Jet Clean	45	0.195	360	21.505	--	--	45	0.915	450	22.615
	Chemical Decon Drain Systems	--	--	435	0.435	109	0.109	--	--	544	0.544
Ancillary											
	Operate Radwaste Systems	--	--	286	1.430	144	21.600	36	0.495	466	23.525
Totals (b)		120	1.068	3 356	37.442	520	24.078	310	7.175	4 306	69.763

(a)Includes shift engineers, crew leaders, craft supervisors, and senior health physics technicians.

(b)The number of significant figures shown is for computational convenience and does not imply precision of that degree.

(c)Indicates that category of decommissioning staff is not involved in the particular task.

The particular method used to deactivate each system or piece of equipment is identified during the planning and preparation phase. In general, all systems not necessary to prevent the spread of contamination are deactivated when empty. All equipment, valves, circuit breakers, etc., are tagged when deactivated. These tags identify the piece of equipment, its system, and its condition.

The first step in equipment deactivation is a safety audit of all pumps and pipes used for chemicals or radioactive materials to ensure that all hazardous or corrosive materials have been removed. Electrical service is disconnected from all equipment not required for operation later in the decommissioning process.

Systems inside the Reactor Building, the Turbine Generator Building, and the Radwaste and Control Building are deactivated by a variety of methods. Many piping systems are isolated using the installed valves, with handles or valve operators removed. Other pipes that have contained contaminated materials are blanked where flanges are readily accessible. Other systems are drained and left open to the atmosphere. Cranes and other electrical equipment not required later in decommissioning are disabled by disconnecting electrical service to prevent their unauthorized use.

For passive safe storage, portions of the facility that contain significant amounts of radioactivity are isolated by the installation of tamper-proof barriers. Indirect access routes, however unlikely, are determined from as-built drawings and sealed. Such routes may include, but are not limited to, access through large vessels, tanks, or large-diameter pipes. Barriers are constructed by welding or bolting and sealing steel plates to block potential pathways of unauthorized entry or contamination migration. Extensive use is made of polysulfide rubber as a sealant because of its durability and flexibility. In the HVAC systems servicing these isolated areas, vents with HEPA filters are installed to allow for changes in air pressure and temperature; however, the systems themselves are deactivated.

REFERENCES

1. U.S. NRC, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code), NUREG-0016, p. 2-39, April 1976.*

* Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.

APPENDIX I

IMMEDIATE DISMANTLEMENT DETAILS

Immediate dismantlement is the decommissioning mode used to remove the radioactive materials from a nuclear power plant immediately following final shutdown. After the radioactive materials are shipped from the site, the facility can be released for unrestricted use. In the reference BWR, radioactive materials are present in the Reactor Building (and Primary Containment), in the Turbine Generator Building, and in the Radwaste and Control Building at the end of the operating life.

Details of immediate dismantlement of the reference BWR are discussed in this appendix, including disassembly methods, schedules and manpower requirements, costs, and external occupational radiation doses. The facility descriptions given in Appendices C, D, and E provide the basic information for the development of the tasks, schedules, manpower loadings, costs, and occupational radiation exposure estimates. Additional design details pertinent to dismantlement of specific equipment items come from engineering drawings, manufacturer's data, and onsite inspection and consultation with construction and operating personnel of Washington Public Power Supply System.

The information in this appendix forms the bases for the activities, costs, and occupational radiation doses for immediate dismantlement presented in Chapters 9, 10, and 11 in Volume 1, respectively.

I.1 DISASSEMBLY METHODS

Following completion of planning and preparation (see Section H.2 of Appendix H) and final plant shutdown, a comprehensive radiation survey of each of the three buildings is completed. These surveys are required to finalize plans for decontaminating contaminated process systems and for installing temporary shielding for personnel protection during the dismantlement of piping and equipment. Next, general cleanup is accomplished, temporary shielding is provided where needed, and staging is erected where it is first used in the dismantlement effort.

At reactor shutdown, the reactor cooling and cleanup systems contain the normal operating inventory of radioactively contaminated water. Throughout the dismantlement of the reference BWR, the contaminated water inventories are processed through the radwaste systems as described in Section H.5 of Appendix H. Selected systems and components with high levels of radioactive contamination are chemically decontaminated before dismantlement, with the decontamination solutions being processed in the radwaste systems for disposal.

Contamination control envelopes in the form of portable ventilation enclosures, temporary greenhouses, or HEPA-filtered vacuum systems are provided where contaminated piping and equipment is being cut up or where contaminated concrete is being removed.

All dismantlement activities are conducted in accordance with the decommissioning quality assurance plan (see Section G.5 of Appendix G) and are checked by the utility quality assurance staff. Environmental monitoring (see Section G.6 of Appendix G) during immediate dismantlement is performed by a specialty contractor.

The disassembly methods proposed for immediate dismantlement of the reference BWR employ techniques that have been used successfully and are described generically in Appendix G. Disassembly methods are discussed in the following subsections for each of the three buildings containing radioactive materials. Generic descriptions for the dismantlement of equipment located in all three buildings are not repeated after they are first discussed (i.e., for piping and equipment, contaminated concrete surfaces, and HVAC and electrical equipment).

I.1.1 Reactor Building/Primary Containment

All of the neutron-activated materials and some of the radioactively contaminated materials in the reference BWR are located in the Primary Containment. Most of the reactor vessel internals and portions of the reactor vessel and the sacrificial shield contain the neutron-activated materials. Neutron-activated components are cut into pieces that will fit steel liners for the B3 shielded shipping cask or other suitable shielded shipping cask; contaminated

components are cut up and packaged in standard shipping boxes, 1.2 m x 1.2 m x 2.4 m, or in specially made boxes (see descriptions in Section G.3 of Appendix G).

In preparation for discharging the fuel, the drywell head and the reactor vessel head are removed, the reactor vessel and the reactor well are filled with water, and the steam dryer and the steam separator are removed from the reactor vessel and placed in the dryer and separator storage pool. While the reactor is being defueled, the steam dryer and the steam separator are cut up under water into packageable segments using a plasma-arc torch and an arc saw. Since the steam dryer and the upper portion of the steam separator contain no neutron-activated material, they are packaged and disposed of as contaminated material. However, the lower portion of the steam separator (i.e., the risers and shroud head plate) is activated and is packaged in steel cask liners.

All of the internals are removed from the reactor vessel with the vessel and the reactor well filled with water. In addition, contamination control envelopes are used for all underwater cutting activities.

Following defueling, the fuel support pieces, the control rods, the control rod guide tubes, the in-core instrument strings, the top fuel guide, and the core support plate are removed from the reactor vessel using standard maintenance procedures. They are cut up in the dryer and separator storage pool using an arc saw or a plasma-arc torch, with all except the control rods and the in-core instrument strings being cut for packaging in B3 steel cask liners. Additional lead shielding is required in the cask liners containing segments of the top fuel guide. Two opposite blades of a cruciform control rod are cut off longitudinally so that the segmented control rod will lie flat. The segmented control rods and the in-core instrument strings are packaged in specially built steel containers that will fit into truck-mounted, NAC-1 spent fuel casks for shipment to the disposal site.

The feedwater inlet, core spray, and residual heat removal piping are next cut from inside the reactor vessel and cut up in the dryer and separator storage pool. These components are assumed not activated; therefore, they are packaged in standard shipping boxes as contaminated material.

The jet pump nozzle/throat sections (see Figure C.3-2 in Appendix C) are then removed using standard maintenance procedures and cut up in the dryer and separator pool for B3 cask shipment. Additional lead shielding is also required.

A plasma-arc torch is used to cut the core shroud into rings of packageable height. The plasma-arc torch is guided by a remotely controlled manipulator (see Section G.2.1 of Appendix G) that is installed in the reactor vessel just before core shroud removal is undertaken. Figure I.1-1 shows the positions at which the core shroud is cut circumferentially. The core shroud is the most radioactive component to be removed from the reference BWR. The rings are moved to the dryer and separator storage pool where they are cut up into segments using a plasma-arc torch. In order to meet the dose rate limits for radioactive shipments, extra shielding is needed for the central core shroud segments (between vessel internals cuts 6 and 12 shown in Figure I.1-1) in addition to that provided by the shielded shipping cask. As an economy measure, pieces of the midsection (most of the activated portion) of the reactor vessel wall are arranged in the steel cask liners to surround the core shroud segments and provide the necessary additional shielding.

The jet pump inlet riser pipes, the jet pump diffusers, and the shroud support plate are removed from the reactor vessel and cut up in the dryer and separator storage pool for packaging in standard shipping boxes as contaminated material.

The reactor vessel is remotely cut into rings of packageable height for removal to the dryer and separator storage pool. Circumferential cuts of the vessel wall are made in air using an oxyacetylene torch installed on the manipulator. Before each cut is made, the torch is positioned, the ring to be removed is rigged to the Reactor Building crane, and the water level in the reactor vessel is lowered to a level just below the circumferential cut line. Locations of the cuts for dismantling the reactor vessel are also shown in Figure I.1-1. The top four rings and the ring just above the bottom head of the vessel are not activated and are cut up under water in the dryer and separator storage pool into segments that are packaged in standard shipping boxes. The bottom head is not removed until the inner steel skin of the sacrificial shield is cut and the manipulator is removed. The rings from the center

FIGURE I.1-1. Cutting Pattern for Removal of the Reactor Vessel Internals and the Reactor Vessel

section of the reactor vessel (between reactor vessel cuts 4 and 12 shown in Figure I.1-1, and containing most of the activated portion of the reactor vessel) are cut into pieces that are used to provide shielding for the central core shroud segments. Although the ring removed by reactor vessel cut 13 is also activated, it is too thick to use as shielding and is cut up and packaged by itself in steel cask liners.

Contaminated piping and equipment inside Primary Containment is removed, segmented, and packaged in standard or specially constructed shipping boxes for disposal. Cutting is accomplished with remote oxyacetylene cutting torches or portable power hacksaws.

The inner steel skin of the sacrificial shield is cut using an oxyacetylene torch guided by the manipulator mounted on the reactor vessel bottom head. The inner skin is cut after it is exposed by removal of rings of the reactor vessel. Cuts are made along lines that will result in packageable segments of the concrete shield when it is cut with explosives along the cuts in the skin. Matching cuts in the outer skin are made with an oxyacetylene torch. Holes for placement of explosive are drilled in the concrete along the lines of the cuts in the outer skin of the shield. The radial beams are cut with oxyacetylene torches and removed after the segments of sacrificial shield above their level are removed.

Contaminated concrete surfaces in the Reactor Building/Primary Containment are removed using the concrete spaller described in Section G.1.3.2 of Appendix G. A surface layer about 50 mm thick is assumed to be removed. The rubble is packaged in standard shipping boxes for disposal.

The contaminated HVAC systems and electrical equipment in the Reactor Building/Primary Containment are cut up into segments that are packaged in standard shipping boxes for disposal. Oxyacetylene torches are used for cutting.

Contaminated piping in the Reactor Building is cut into pieces that are packaged in standard shipping boxes. Cutting is accomplished with oxyacetylene torches or portable hacksaws.

Small contaminated equipment in the Reactor Building is removed and packaged in standard shipping boxes. Large contaminated equipment with no external smearable contamination and weighing less than 20 Mg is sealed by welding steel plates over all of the openings and shipped to the burial ground without additional packaging. Contaminated equipment too large or too heavy to be shipped as a unit is cut up, depending on its complexity, either into segments that fit into standard boxes (as is the case with the pool liners) or into segments that are sealed with welded steel plates and thus use the shell as the contamination barrier (as is the case with the residual heat removal [RHK] heat exchangers).

The stainless steel liners of the spent fuel storage pool, the dryer and separator storage pool, and the reactor well are cut up in air using a plasma-arc torch into segments that will fit in standard boxes. The spent fuel storage racks are packaged in specially built steel boxes for shipment to the burial ground.

I.1.2 Turbine Generator Building

Contamination of piping and equipment in the Turbine Generator Building results from contact with the reactor steam and steam condensate. Compared to the Primary Containment, contamination in the Turbine Generator Building is light, but fairly extensive.

The main turbine is a very large machine that has to be dismantled and disposed of as contaminated material. Dimensions of the high-pressure turbine and the three low-pressure turbines are given in Table C.3-5 in Appendix C. Dismantlement of the high-pressure turbine is accomplished in the following sequence: 1) the cover is unbolted, removed, and segmented; 2) the top blade ring halves are removed and segmented; 3) the rotor is removed and segmented; 4) the bottom blade ring halves are removed and segmented; and 5) the base is removed and segmented. The removal sequence for each low-pressure turbine is: 1) outer cover, 2) No. 1 inner cover, 3) No. 2 inner cover, 4) top-half blade rings, 5) rotor, 6) bottom-half blade rings, 7) No. 2 inner base, 8) No. 1 inner base, and 9) outer base. Oxyacetylene torches are used for cutting the turbine components.

Low-pressure feedwater heaters IA, IB, and IC and associated piping that are located in the steam inlets to the main condenser are water-jet cleaned to reduce the external contamination, then removed and packaged as required.

The condenser internal surfaces and the exposed external surfaces of the tube bundles are water-jet cleaned to reduce the radiation dose rate and the airborne contamination potential. The main condenser is shown in Figure C.3-7 in Appendix C. In each of the three banks of tubes there are two sections. The tubes in each section are supported by 14 tube support plates. The tubes are cut from between every other pair of support plates (e.g., between the first and second, third and fourth, and so on) leaving the tubes in place between the end tube sheet and the first support plate, and between the second and third support plates, the fourth and fifth support plates, and so on. The cut tube segments are packaged in standard shipping boxes. The tube sheets and support plates are cut to form segments of the tube bundle that fit in standard shipping boxes. After complete gutting, the condenser shell is segmented and packaged in standard shipping boxes.

The feedwater heaters and the moisture separator reheaters in the Turbine Generator Building are too large to transport to a burial ground in one piece. Since the external surfaces of these equipment pieces either have no smearable contamination or can be cleaned to that state, the shells of these pieces of equipment, after capping all openings, are used as shipping and burial containers. An example is the dismantlement of the two high-pressure feedwater heaters, which are 2.4 m in diameter and 12.5 m in length. Each high-pressure feedwater heater is cut into four sections with a maximum weight of 20 Mg each. The shell is cut circumferentially at the first designated cut line with an oxacetylene torch, and the shell section is moved back enough to make working room for cutting the tube-bundle at the same cut line. After cutting, the tube bundle section is slid into the shell section, then a steel plate is welded over the cut end of the shell section. Steel plates are likewise welded over all nozzles on the shell section. The remaining shell and tube bundle are cut at two more locations in a similar fashion, and the cut ends and the nozzles of the shell sections are sealed with welded steel plates.

Contaminated piping, concrete surfaces, and HVAC and electrical equipment are removed as described previously for the Reactor Building/Primary Containment.

I.1.3 Radwaste and Control Building

Contaminated piping and equipment in the Radwaste and Control Building are primarily parts of the radwaste systems. Much of the liquid and solid radwastes from dismantlement of the Reactor Building and the Turbine Generator Building are processed in the radwaste systems before the systems are dismantled. A temporary, portable radwaste system is provided by a specialty contractor and is used to process liquid waste generated after dismantlement of the in-house liquid radwaste systems is started.

The techniques used in dismantling the piping and equipment in the Radwaste and Control Building are the same as those described previously for the Reactor Building and the Turbine Generator Building.

I.1.4 Special Tools and Equipment

Many of the tools and equipment needed for immediate dismantlement of the reference BWR are not needed during the plant's operating lifetime. The more unique devices used in dismantlement are described in Sections G.1, G.2, and G.4 of Appendix G. A listing of the special tools and equipment for immediate dismantlement and their functions is given in Table I.1-1.

I.1.5 Summary of Disassembly Methods

A summary of the dismantlement data for the reference BWR is given in Table I.1-2. Dismantlement tasks that result in the removal of materials that contain neutron-activation products or are radioactively contaminated are identified, cutting data are presented, and removal conditions and methods are given.

I.2 SCHEDULES AND MANPOWER REQUIREMENTS

Development of the immediate dismantlement schedules and manpower requirements requires several steps to arrive at reasonably optimum results. The sequence in which the various systems must be decontaminated and dismantled is determined. Next, the task time requirements and the numbers and types of

TABLE I.1-1. Special Tools and Equipment for Immediate Dismantlement

Item	Number Required ^(a)	Function
Underwater Manipulator	1	Positioning and Movement of Underwater Cutting Devices Used in Reactor Vessel
Underwater Plasma-Arc Torch	2	Sectioning Vessel Internals
Underwater Oxyacetylene Torch	2	Sectioning Reactor Vessel
Arc Saw	1	Sectioning Materials from Reactor Vessel and Internals
Portable Plasma-Arc Torch	2	Cutting Stainless Steel Equipment, Piping, and Pool Liners
Portable Oxyacetylene Torch	10	Cutting Carbon Steel Equipment and Piping; Welding Carbon Steel
Remote-Controlled Oxyacetylene Torch	4	Remote Cutting of Carbon Steel Equipment and Piping
Guillotine Pipe Saw	10	Cutting Piping
Power-Operated Reciprocating Hacksaw	10	Sectioning Piping and Equipment
Closed-Circuit, High-Resolution Television	2	Observation of Remote or Underwater Operations
Underwater Lights and Periscopes	As Required	Illuminating and Observing Underwater Operations
Underwater Tools (e.g., Impact Wrenches, Bolt Cutters, Tongs)	As Required	Underwater Disassembly, Handling, and Packaging of Activated and Highly Contaminated Materials
Submersible Pump with Disposable Filter	5	Draining of Suppression Chamber and Spent Fuel Storage Pools
High-Pressure Water Jet	2	Surface Decontamination of Suppression Chamber, Tanks, Main Condenser, Pool Liners
Pipe Jumper	4	Chemical Decontamination of Reactor Piping Systems
Mobile Chemical Decontamination Unit	5	Chemical Decontamination of Liquid and Solid Radwaste Equipment
Mobile Chemical Mixing and Heating Unit	4	Chemical Decontamination of Drain Systems
Scaffolding	200	Safe Access to Heights
Safety Nets	As Required	Protect Personnel Working on Elevated Equipment and Structures
Power Operated, Mobile, Scissors-Type Manlift	6	Safe Access to Heights
Power Operated, Mobile, Articulated-Arm Manlift	3	Safe Access to Heights
9100-Kg Mobile Hydraulic Crane	3	Removal and Packaging of Contaminated Piping and Equipment
9100-Kg Forklift	6	Packaging of Contaminated Materials, and Loading of Trucks
Rigging Materials (e.g., Chokers, Grapples, Winches, etc.)	As Required	Handling of Equipment and Piping
Concrete Drill with HEPA-Filtered Dust Collecting System	4	Drilling Holes in Concrete as Required for Blasting or Surface Spalling
Concrete Surface Spaller	4	Removal of Contaminated Concrete Surfaces
Front-End Loader (Light Duty)	3	Cleanup and Packaging of Concrete Rubble
Vacuum Cleaner (HEPA-Filtered)	3	Cleanup of Concrete Rubble Fines
Portable Ventilation Enclosure	10	Contamination Control During Cutting of Contaminated Material
Supplied-Air Plastic Suit	250	Provide Personnel with Maximum Respiratory and Body-Surface Protection from Radioactive Contaminants
Filtered-Exhaust Fan Unit	4	Contamination Control and Personnel Comfort During Main Condenser Removal
Polyurethane Foam Generator	2	Contamination Control During HVAC Removal

(a)Based on a taskwise analysis of the immediate dismantlement schedule, plus spares as required.

TABLE I.1-2. Immediate Dismantlement Removal Data and Methods

Location	Task		Estimated Cutting Length (m)	Nominal Material Thickness (m)	Estimated Cutting Time (hr)	Strategy and Work Conditions(a)	Methods						
Reactor Building/Primary Containment													
4	Remove Dryer and Separator	(Activated)	9 0 x 10 ²	1 2 x 10 ⁻³	3 2 x 10 ¹	(1,2,5,7)	Remove to dryer and separator pool and segment	Package					
		(Contaminated)	5 3 x 10 ²	1 4 x 10 ⁻³	2 6 x 10 ¹	(1,2,5,7)							
6	Remove Reactor Vessel Internals	(Activated)	2 5 x 10 ³	5 0 x 10 ⁻²	5 2 x 10 ²	(1,2,5,7)	Cut in vessel	Remove to dryer and separator pool and segment	Package				
		(Contaminated)	1 6 x 10 ²		5 5 x 10 ²	(1,2,5,7)							
12	Remove Reactor Vessel	(Activated)	1 7 x 10 ²	1 8 x 10 ⁻¹	3 9 x 10 ¹	(3,6,7)	Support vessel section	Adjust water level	Cut section	Transfer to dryer and separator pool			
		(Contaminated)	5 1 x 10 ²	2 0 x 10 ⁻¹	9 4 x 10 ¹	(3,6,7)							
		Segment (Activated)	4 0 x 10 ²	1 8 x 10 ⁻¹	9 2 x 10 ¹	(3,5,7)					Segment vessel sections in dryer and separator pool	Package	
		(Contaminated)	2 6 x 10 ²	2 0 x 10 ⁻¹	4 7 x 10 ¹	(3,5,7)							
14	Remove Primary Containment Piping	Average O.D. (mm)					Decontaminate as required	Cut into 2 l-m segments	Include small valves	Cut off large valves	Package		
		60	4 7 x 10 ²	(Schedule 80 Pipe)	3 4 x 10 ²	(3,4,6,7)							
		152	4 5 x 10 ²		3 2 x 10 ²	(3,4,6,7)							
		356	2 3 x 10 ³		1 6 x 10 ²	(3,4,6,7)							
		533	1 4 x 10 ³		9 7 x 10 ²	(3,6,7)							
		660	1 0 x 10 ²		7 5 x 10 ¹	(3,6,7)							
		Whip Restraints and Hangers	1 4 x 10 ²	5 0 x 10 ⁻³	9 5 x 10 ¹	(3,7)							
15	Remove Sacrificial Shield and Radial Beams	Cutting (Activated)	2 0 x 10 ²	7 0 x 10 ⁻²	1 3 x 10 ¹	(3,6,7)	Cut outer skin and inner skin to achieve segment size						
		(Contaminated)	7 1 x 10 ¹	7 0 x 10 ⁻³	4 7 x 10 ¹	(3,7)							
		Drilling and Blasting	(8 9 x 10 ¹ holes)	(4 3 x 10 ⁻³ m ³ waste volume)	1 5 x 10 ¹	(7)					Drill holes	Blast into segments	Package
		Radial Beams	7 3 x 10 ²	2 5 x 10 ⁻²	2 5 x 10 ²	(3,7)							
16	Remove Contaminated Concrete from Primary Containment	Drilling	(2 6 x 10 ⁴ holes)	(1 0 x 10 ⁻¹ m ³ waste volume)	3 8 x 10 ³	(7)	Drill holes for removal of 0.051-m contamination depth						
		Spalling	(2 2 x 10 ³ m ² surface area)	(1 8 x 10 ⁻² m ³ rubble volume)	6 6 x 10 ³	(7)					Spall surface layer with hydraulic spaller	Package	
17	Remove HVAC and Electrical from Primary Containment	HVAC	2 0 x 10 ³	6 0 x 10 ⁻³	1 1 x 10 ²	(3)	Requires scaffolding	Cut ductwork, fans, and hangers	Package				
		Electrical	7 6 x 10 ²	1 8 x 10 ⁻²	1 2 x 10 ²	(3)					Cut trays, braces, and conduit into 2 l-m segments	Package	
20	Remove Reactor Building Piping	Average O.D. (mm)					Decontaminate as required	Cut into 2 l-m segments	Include small valves	Cut off large valves	Package		
		60	9 6 x 10 ²	(Schedule 80 Pipe)	6 8 x 10 ²	(1,3,4,6,7)							
		152	1 4 x 10 ²		1 0 x 10 ²	(1,3,4,6,7)							
		356	4 4 x 10 ²		3 2 x 10 ²	(1,3,4,6,7)							
		533	1 2 x 10 ³		8 5 x 10 ²	(3,6,7)							
		660	9 8 x 10 ¹		7 0 x 10 ¹	(3,6,7)							
25	Remove Reactor Building Equipment(b)		2 0 x 10 ³	1 0 x 10 ⁻²	3 1 x 10 ²	(3,7)	Decontaminate as required	Segment and cap-weld large heat exchangers	Cut tanks, pumps, and other equipment as required	Package			
26	Remove Liners from Spent Fuel Pool, Reactor Well, and Dryer and Separator Pool		2 2 x 10 ³	6 0 x 10 ⁻³	1 5 x 10 ²	(3,7)	Decontaminate as required	Cut into 1.3-m x 2 l-m segments	Package				
27	Remove Reactor Building Contaminated Concrete	Drilling	(3 5 x 10 ⁴ holes)	(1 4 x 10 ⁻¹ m ³ waste volume)	1 9 x 10 ³	(7)	Drill holes for removal of 0.051-m contamination depth						
		Spalling	(2 9 x 10 ³ m ² surface area)	(1 5 x 10 ⁻² m ³ rubble volume)	1 1 x 10 ³	(7)					Spall surface layer with hydraulic spaller	Package	
28	Remove HVAC and Electrical from Reactor Building	HVAC	7 0 x 10 ³	6 0 x 10 ⁻³	3 7 x 10 ²	(3)	Requires scaffolding	Cut ductwork, fans, and hangers	Package				
		Electrical	1 6 x 10 ²	1 8 x 10 ⁻²	2 4 x 10 ¹	(3)					Cut trays and braces into 2 l-m segments	Package	
Turbine Generator Building													
4	Remove Turbine		2 4 x 10 ³	5 6 x 10 ⁻²	2 3 x 10 ²	(3,7)	HP turbine	remove and cut cover, remove rings, cut rotor shaft, and cut base	Package	LP turbines	unbolt and separate cover, cut cover sections, inner covers, rings, inner base, rotor, and base sections	Package	
6	Remove Condenser		2 6 x 10 ⁴	5 2 x 10 ⁻³	4 0 x 10 ³	(3,7)	Remove end water boxes	Segment internal steel and tube bundles into packageable sections	Set scaffolding	Cut shell into 1.2-m x 2 l-m segments	Package		
9	Remove Piping	Average O.D. (mm)					Decontaminate as required	Cut into 2 l-m segments	Include small valves	Cut off large valves	Package		
		60	5 0 x 10 ²	(Schedule 80 Pipe)	4 2 x 10 ²	(3,4,7)							
		152	1 5 x 10 ³		1 1 x 10 ²	(3,4,7)							
		356	1 6 x 10 ³		9 0 x 10 ²	(3,4,7)							
		533	1 7 x 10 ²		1 4 x 10 ³	(3,7)							
		660	7 4 x 10 ³		4 1 x 10 ²	(3,7)							
		1 220	1 5 x 10 ³		6 0 x 10 ²	(3,7)							
10	Remove Equipment	Low-Pressure Feedwater Heaters	6 7 x 10 ³	1 8 x 10 ⁻³	2 9 x 10 ²	(3,7)}	Segment shell and tube bundles	Cap-weld openings					
		High-Pressure Feedwater Heaters	2 5 x 10 ³	7 9 x 10 ⁻³	3 4 x 10 ²	(3,7)}							
		Moisture Separator Reheaters	3 9 x 10 ³	2 6 x 10 ⁻³	1 7 x 10 ²	(3,7)}							
		Miscellaneous Equipment	2 8 x 10 ³	1 2 x 10 ⁻²	1 6 x 10 ²	(3,7)}							
11	Remove Contaminated Concrete	Drilling	(2 3 x 10 ⁴ holes)	(9 1 x 10 ⁻² m ³ waste volume)	1 3 x 10 ³	(7)	Drill holes for removal of 0.051-m contamination depth						
		Spalling	(1 9 x 10 ³ m ² surface area)	(1 2 x 10 ⁻² m ³ rubble volume)	7 4 x 10 ²	(7)					Spall concrete from surface with hydraulic spaller	Package	
12	Remove HVAC and Electrical Systems	HVAC	8 4 x 10 ³	6 0 x 10 ⁻³	1 3 x 10 ²	(3)	Requires scaffolding	Cut ductwork, fans, and hangers	Package				
		Electrical	7 4 x 10 ²	5 0 x 10 ⁻³	1 9 x 10 ¹	(3)					Cut trays and braces, into 2 l-m sections	Package	
Radwaste and Control Building													
5	Remove Piping	Average O.D. (mm)					Decontaminate as required	Cut into 2 l-m segments	Include small valves	Cut off large valves	Package		
		60	4 6 x 10 ²	6.0×10^{-3}	4 0 x 10 ²	(3,4,7)							
		152	1 4 x 10 ²		1 0 x 10 ³	(3,4,7)							
		356	2 8 x 10 ¹		1 6 x 10 ²	(3,4,7)							
		533	1 9 x 10 ²		7 0 x 10 ⁰	(3,7)							
		914	1 6 x 10 ²		3 8 x 10 ¹	(3,7)							
7	Remove Equipment		6 2 x 10 ³	1 3 x 10 ⁻²	3 3 x 10 ²	(3,7)	Decontaminate as required	Cut tanks and pumps as required	Package				
8	Remove Contaminated Concrete	Drilling	(2 4 x 10 ⁴ holes)	(9 6 x 10 ⁻² m ³ waste volume)	6 6 x 10 ²	(7)	Drill holes for removal of 0.051-m contamination depth						
		Spalling	(2 0 x 10 ³ m ² surface area)	(1 0 x 10 ⁻² m ³ rubble volume)	3 8 x 10 ²	(7)					Spall concrete from surface with hydraulic spaller	Package	
9	Remove Miscellaneous Steel Structures		2 5 x 10 ²	1 8 x 10 ⁻²	1 3 x 10 ¹	(3)	Remove, segment, and package						
10	Remove HVAC and Electrical Systems	HVAC	9 0 x 10 ³	6 0 x 10 ⁻³	1 4 x 10 ²	(3)	Requires scaffolding	Cut ductwork fans and hangers	Package				
		Electrical	4 6 x 10 ²	1 8 x 10 ⁻²	2 4 x 10 ¹	(3)					Cut trays and braces into 2 l-m pieces	Package	

(a)1 = plasma-arc torch, 2 = arc saw, 3 = oxyacetylene torch, 4 = power hacksaw, 5 = underwater cutting, 6 remote cutting, 7 = contamination control envelope
(b)Includes Primary Containment equipment

decommissioning workers required to accomplish each task in the allotted time are estimated (the functions and organization of the decommissioning staff are discussed in Section H.1 of Appendix H). Based on the estimated dose to accomplish each task, the number of workers needed to complete the radiation zone work in the allotted time and within the assumed radiation dose limits is determined. Whole-body radiation doses to the decommissioning workers are limited in accordance with 10 CFR 20.101. The supervisors, utility operators, and health physics technicians are assumed to be long-time radiation workers whose annual exposure is limited to 5 rem/yr by the formula 5(N-18) of 10 CFR 20.101(b)(2). The craftsmen and laborers are assumed to have had little prior radiation exposure and, therefore, under 10 CFR 20.101(b)(1) and (2) may receive up to 3 rem per quarter, within the limitation of the formula 5(N-18). In those instances where the manpower estimated for physically accomplishing a task results in an average dose for each person in excess of these limits, additional people are assigned to the tasks to keep the individual occupational doses below these limits. In the manpower tables presented in this section, the manpower shown is adequate both to accomplish the task and to meet these occupational dose limits.

Task schedules and decommissioning worker requirements for immediate dismantlement of the Reactor Building/Primary Containment, the Turbine Generator Building, and the Radwaste and Control Building are discussed and presented in the following three subsections. The decommissioning worker requirements given in these subsections do not include the management and support staff (see Figure H.1-1 in Appendix H). A fourth subsection presents the overall task schedule and decommissioning worker requirements and the total staff labor requirements for immediate dismantlement of the reference BWR.

I.2.1 Reactor Building/Primary Containment

Over half the effort of dismantling the reference BWR is involved in the Reactor Building/Primary Containment. All of the neutron-activated materials are in the Primary Containment and much of the material contaminated with fission products and activated corrosion products is in the Primary Containment and the Reactor Building. Based on the dismantlement methods discussed previously in Section I.1 and the occupational dose estimates presented later in Section I.4,

the task schedule and sequence and the decommissioning worker requirements for immediate dismantlement of the Reactor Building/Primary Containment are presented in Figure I.2-1. Immediate dismantlement of the Reactor Building/Primary Containment is estimated to be accomplished in about 30 months following final reactor shutdown and is estimated to require about 2340 man-months of decommissioning worker effort.

I.2.2 Turbine Generator Building

The Turbine Generator Building has less deposited radioactive contamination per unit area than the other two buildings, and, therefore, has the lowest radiation dose rates for dismantlement of the contaminated piping and equipment. In arranging the task schedule for immediate dismantlement of the Turbine Generator Building, the work is planned so that low radiation exposure work in this building is in progress at the same time as much of the high exposure work in the other two buildings, so that by rotating workers between the buildings more efficient utilization of the manpower from an occupational dose consideration can be achieved. The task schedule and sequence and the decommissioning worker requirements for immediate dismantlement of the Turbine Generator Building are given in Figure I.2-2. Immediate dismantlement of the Turbine Generator Building is estimated to be completed 27 months following final reactor shutdown and is estimated to require approximately 640 man-months of decommissioning worker effort.

I.2.3 Radwaste and Control Building

The liquid and solid radwaste systems located in the Radwaste and Control Building are needed to process most of the contaminated liquids contained in the systems at final reactor shutdown and generated during dismantlement. Consequently, the Radwaste and Control Building is the last to be dismantled. A temporary, portable radwaste system is assumed installed when expedient to process the remainder of these contaminated liquids during dismantlement of the in-house systems. Figure I.2-3 shows the task schedule and sequence and the decommissioning worker requirements for immediate dismantlement of the Radwaste and Control Building. Dismantlement of the Radwaste and Control Building starts in month 17 after final reactor shutdown and continues through

FIGURE I.2-1. Task Schedule and Sequence and Decommissioning Worker Requirements for Immediate Dismantlement of the Reactor Building/Primary Containment

FIGURE I.2-2. Task Schedule and Sequence and Decommissioning Worker Requirements for Immediate Dismantlement of the Turbine Generator Building

FIGURE I.2-3. Task Schedule and Sequence and Decommissioning Worker Requirements for Immediate Dismantlement of the Rad-Waste and Control Building

month 42, thus requiring an estimated 25 months to complete. In addition, it is estimated to require nearly 1070 man-months of decommissioning worker effort.

I.2.4 Overall Schedule and Manpower Requirements

The overall task schedule and sequence for immediate dismantlement of the reference BWR is shown in Figure I.2-4. The planning and preparation tasks (see Section H.2 of Appendix H) for the estimated 2-year period preceding reactor shutdown are also included. Dismantlement of the three buildings is estimated to be accomplished in 42 months after final reactor shutdown.

Immediate dismantlement tasks, with a few exceptions, are performed on two 8-hr shifts, 5 days per week. Discharge and shipment of spent fuel, shipment of neutron-activated reactor vessel internals and reactor vessel segments, decontamination and draining of systems and equipment, and operation of the radwaste systems are conducted three shifts per day, 7 days per week, as required. The overall dedicated manpower requirements for each immediate dismantlement task are given in Table I.2-1. The overall decommissioning worker requirements for the period following reactor shutdown are shown in Table I.2-2. Slightly more than 4040 man-months of "hands-on" effort are estimated.

The total staff labor requirements for immediate dismantlement of the reference BWR are given in Table I.2-3. These requirements are given in equivalent man-years for the 2 years before and the 4 years following final reactor shutdown, and include the management and support staff as well as the decommissioning workers. A little more than 610 man-years of effort are estimated for immediate dismantlement of the reference BWR, including approximately 230 man-years for the management and support staff and about 380 man-years for the decommissioning workers. The following assumptions are used in making this estimate:

- The annual management and support staff requirements are nearly identical to those specified in Reference 1, except for the control room operators, the security shift supervisors, and the security patrolmen, whose requirements are developed specifically for this study.

FIGURE I.2-4. Overall Task Schedule and Sequence for Immediate Dismantlement of the Reference BWR

TABLE I.2-1. Dedicated Manpower Requirements for Immediate Dismantlement Tasks

Building	Task	Duration (months)	Dedicated Manpower Requirements (a) (man-months)				Health Physics Technicians
			Crew Leaders	Utility Operators	Laborers	Craftsmen	
Reactor Building/Primary Containment							
1.	Install HEPA Filters(c)	2	2.0	--(d)	--	30.0	--
2.	Comprehensive Radiation Survey	1	--	--	--	--	6.0
3.	Discharge and Ship Fuel	2/15	--	24.0	--	--	--
4.	Remove Dryer and Separator	2	--	8.0	--	8.0	--
5.	Drain Suppression Pool to Radwaste; Water-Jet Clean(e)	4	16.8	68.4	--	--	16.8
6.	Remove Reactor Vessel Internals	6.5	--	39.0	--	26.0	--
7.	Ship Activated Reactor Vessel Internals(f)	13	111.8	166.4	--	--	54.6
8.	Drain Reactor Well Pool to Radwaste; Water-Jet Clean(e)	1	4.2	17.1	--	--	4.2
9.	Chemical Decon Reactor Water Recirculation and Cleanup Systems(e)	1	8.6	25.7	--	--	8.6
10.	Cleanup, Stage and Shield Hot Spots in Primary Containment	1	--	--	6.0	--	--
11.	Enlarge Suppression Chamber Access	0.5	--	--	2.0	1.0	--
12.	Remove Reactor Vessel	3.5	--	14.0	--	14.0	--
13.	Ship Activated Reactor Vessel Segments(f)	5	43.0	64.0	--	--	21.0
14.	Remove Primary Containment Piping and Equipment(g)	8	--	96.0	80.0	64.0	--
15.	Remove Sacrificial Shield and Radial Beams	5	--	--	30.0	40.0	30.0
16.	Remove Contaminated Concrete from Primary Containment	3	--	--	24.0	--	--
17.	Remove HVAC and Electrical Equipment from Primary Containment	1	--	--	4.0	4.0	--
18.	Drain Contaminated Systems to Radwaste(e)	1.5	6.3	25.6	--	--	6.3
19.	Chemical Decon Residual Heat Removal, Low- and High-Pressure Core Spray Systems(e)	0.5	4.3	12.8	--	--	4.3
20.	Remove Reactor Building Piping(g)	10.5	--	147.0	126.0	84.0	--
21.	Drain Dryer and Separator Pool to Radwaste; Water-Jet Clean(e)	1	4.2	17.1	--	--	4.2
22.	Chemical Decon Drain Systems(e)	0.5	4.3	12.8	--	--	4.3
23.	Drain Spent Fuel Pool To Radwaste; Water-Jet Clean(e)	2	8.4	34.2	--	--	8.4
24.	Chemical Decon Fuel Pool Cooling and Cleanup System(e)	0.5	2.1	4.3	--	--	2.1
25.	Remove Reactor Building Equipment	3	--	18.0	30.0	12.0	--
26.	Remove Liners from Spent Fuel Pool, Reactor Well, and Dryer and Separator Pool	1	--	--	8.0	8.0	--
27.	Remove Reactor Building Contaminated Concrete	2.5	--	--	20.0	--	--
28.	Remove HVAC and Electrical Systems from Reactor Building	1	--	--	6.0	6.0	--
29.	Final Radiation Survey(h)	1	--	--	--	--	2.0
Turbine Generator Building							
1.	Install HEPA Filters(c)	5.5	5.5	--	--	82.5	--
2.	Comprehensive Radiation Survey(d)	1	--	--	--	--	2.0
3.	Cleanup and Stage	1	--	--	4.0	--	--
4.	Remove Turbine	5	--	--	20.0	10.0	--
5.	Drain Contaminated Systems to Radwaste; Water-Jet Clean Condensate Storage Tanks(e)	3	12.6	51.3	--	--	12.6
6.	Remove Condenser	4	--	16.0	48.0	48.0	--
7.	Drain Condenser to Radwaste; Water-Jet Clean(e)	3	12.6	51.3	--	--	12.6
8.	Chemical Decon Drain Systems(e)	0.5	4.3	12.8	--	--	2.1
9.	Remove Piping	9	--	--	36.0	108.0	--
10.	Remove Equipment	3	--	12.0	12.0	18.0	--
11.	Remove Contaminated Concrete	1.5	--	--	12.0	--	--
12.	Remove HVAC & Electrical Systems	1	--	--	4.0	2.0	--
13.	Final Radiation Survey(h)	0.5	--	--	--	--	1.0
Radwaste and Control Building							
1.	Comprehensive Radiation Survey and Release of Support Facilities/Comprehensive Radiation Survey of Radwaste & Control Building(h)	2/1	--	--	--	--	10.0
2.	General Cleanup	0.5	--	--	2.0	2.0	--
3.	Chemical Decon Drain Systems(e)	1.5	6.3	25.6	--	6.3	6.3
4.	Chemical Decon Equipment; Water-Jet Clean(e)	3	12.6	51.3	--	12.6	12.6
5.	Remove Piping	8	--	80.0	64.0	128.0	80.0
6.	Install Temporary Radwaste System	0.5	--	--	1.0	1.0	--
7.	Remove Equipment	9	--	90.0	108.0	90.0	90.0
8.	Remove Contaminated Concrete	2	--	--	16.0	--	--
9.	Remove Miscellaneous Steel Structures	1	--	--	4.0	4.0	--
10.	Remove HVAC and Electrical Systems	1	--	--	8.0	8.0	--
11.	Final Radiation Survey	1	--	--	--	--	2.0

(a)Non-dedicated personnel per shift (two-shift-per-day basis) include one shift engineer, one crew leader, one craft supervisor, and one senior health physics technician, as well as one packing and support crew (one utility operator and three craftsmen) and one health physics technician for each building undergoing active dismantlement.
(b)All tasks are conducted on two-shift-per-day basis, 5 days per week, except as noted otherwise.
(c)Performed primarily before reactor shutdown, but included here for consistency's sake.
(d)Indicates either no dedicated personnel required or sufficient non-dedicated personnel available to complete the task.
(e)Conducted on a three-shift-per-day, 7-days-per-week basis; includes operation of radwaste system.
(f)Conducted on a three-shift-per-day, 7-days-per-week basis.
(g)Radiation exposure limitations require multiple task assignments per shift for crew members.
(h)Requires assignment of additional senior health physics technician(s).

SUMMARY/MANPOWER		SUMMARY MAN-MONTHS PER WORKING MONTH																																										
LABOR GRADE	TOTAL M/M	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	37	38	39	40	41	42	
SHIFT ENGINEER	84.0	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	
CREW LEADER	338.1	6.2	12.7	14.8	14.8	14.8	14.8	14.9	10.6	12.7	19.1	19.1	14.8	14.8	14.8	17	17	10.6	12.7	14.8	10.4	10.4	8.3	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	—
UTILITY OPERATOR	1265.4	35.1	41.4	53.9	53.9	53.9	53.9	49.6	34.8	40.4	51.8	52.7	47.9	51.9	51.9	54.2	50.2	28.8	38.4	45.9	50.2	52.2	43.4	32	24	14	14	13	12	12	10	10	10	10	10	10	10	10	12	12	2	2	—	
LABORER	651.0	—	4	16	16	16	16	16	12	15	11	10	22	24	24	24	12	12	14	13	12	20	22	24	36	38	30	22	15	11	8	12	12	12	12	12	12	12	20	20	4	8	—	
CRAFT SUPERVISOR	82.0	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	—		
CRAFTSMAN	998.0	12	18	28	28	28	28	28	20	18	13	14	34	44	44	40	28	28	33	29	28	36	36	52	48	46	31	32	27	23	16	10	10	10	10	10	10	10	10	10	8	—		
SENIOR HEALTH PHYSICS TECHNICIAN	94.0	6	4	2	2	2	2	2	2	2	2	2	2	2	2	2	2	4	2	2	2	2	2	2	2	2	2	3	2	2	3	2	2	2	2	2	2	2	2	2	2	2		
HEALTH PHYSICS TECHNICIAN	532.4	18.2	14.3	12.4	12.4	12.4	12.4	12.5	6.2	8.3	15.3	16.7	12.4	12.4	12.4	14.5	14.6	10.2	12.3	12.4	12.4	12.4	10.3	22	22	20	15	15	13	13	12	13	13	13	13	13	13	13	13	13	2	2	2	
TOTAL	4044.9	81.5	98.4	131.1	131.1	131.1	131.1	127	89.6	100.4	116.2	118.5	137.1	153.1	153.1	155.7	127.8	97.6	116.4	121.1	119	137	126	138	138	126	98	91	75	67	55	53	53	53	53	53	53	63	63	26	28	6		

TABLE I.2-2. Overall Decommissioning Worker Requirements for Immediate Dismantlement of the Reference BWR

TABLE I.2-3. Staff Labor Requirements for Immediate Dismantlement

Position	Time Relative to Final Reactor Shutdown (year)						Total Staff Labor Required (man-years)
	-2	-1	1	2	3	4	
	Annual	Staff	Labor	Requirement	(man-years)	(a)	
Management and Support Staff							
Decommissioning Superintendent	0.3	1.0	1.0	1.0	1.0	0.8 ^(b)	5.1
Secretary	1.0	2.0	3.0	3.0	3.0	1.6 ^(b)	13.6
Clerk	0	1.0	2.0	2.0	2.0	0.5	7.5
Decommissioning Engineer	1.0	1.0	1.0	1.0	1.0	0.8 ^(b)	5.8
Assistant Decommissioning Engineer	1.0	1.0	1.0	1.0	1.0	0.4	5.4
Radioactive Shipment Specialist	0	1.0	1.0	1.0	1.0	0.4	4.4
Procurement Specialist	0.3	1.0	1.0	1.0	1.0	0.4	4.7
Tool Crib Attendant	0	0	2.0	2.0	2.0	0.8	6.8
Control Room Operator ^(c)	0	0	5.0	5.0	5.0	2.1	17.1
Security Supervisor	0	0	1.0	1.0	1.0	0.5	3.5
Security Shift Supervisor ^(d)	0	0	5.0	5.0	5.0	2.5	17.5
Security Patrolman ^(d)	0	0	39.0	28.0	13.0	6.5 ^(b)	86.5
Contracts and Accounting Specialist	0.3	1.0	1.0	1.0	1.0	0.8 ^(b)	5.1
Health and Safety Supervisor	1.0	1.0	1.0	1.0	1.0	0.8 ^(b)	5.8
Health Physicist	0	0.5	1.0	1.0	1.0	0.5	4.0
Protective Equipment Attendant	0	0	2.0	2.0	2.0	1.0	7.0
Industrial Safety Specialist	0.3	1.0	1.0	1.0	1.0	0.5	4.8
Quality Assurance Supervisor	0.3	1.0	1.0	1.0	1.0	0.8 ^(b)	5.1
Quality Assurance Engineer	0.5	2.0	1.0	1.0	1.0	0.5	6.0
Quality Assurance Technician	0	0.5	4.0	4.0	4.0	2.0	14.5
Consultant (Safety Review)	0.5	0.5	0.5	0.5	0.5	0.3	2.8
Subtotals, Management and Support Staff	6.5	15.5	74.5	63.5	48.5	24.5	233.0
Decommissioning Workers^(e)							
Shift Engineer	1.0	2.0	2.0	2.0	2.0	1.0	10.0
Crew Leader ^(f)	0	1.0	15.7	12.0	2.0	0.8	31.5
Utility Operator ^(g)	0	3.2	55.0	49.6	12.4	3.3	123.5
Laborer	0	0	12.9	19.8	16.3	5.3	54.3
Craft Supervisor	0	0.6	2.0	2.0	2.0	0.8	7.4
Craftsman	0	10.0	22.4	37.2	19.6	4.0	93.2
Senior Health Physics Technician	0	1.0	2.5	2.2	2.1	1.0	8.8
Health Physics Technician ^(h)	0	3.0	14.3	14.9	14.0	3.8	50.0
Subtotals, Decommissioning Workers	1.0	20.8	126.8	139.7	70.4	20.0	378.7
Totals	7.5	36.3	201.3	203.2	118.9	44.5	611.7

(a) Rounded to the next higher 0.1 man-year.

(b) Includes an additional 4 months following active decommissioning in order to complete the documentation and other unspecified license and contract termination requirements.

(c) Based on one operator per shift in the control room, three shifts per day, 7 days per week.

(d) Based on 10 CFR Part 73 and information obtained from Washington Public Power Supply System; includes both response and access-control personnel on a three-shift, 7-day-week basis.

(e) Requirements during the 4 years following reactor shutdown are based on Table I.2-2, unless otherwise noted; individual man-month requirements in Table I.2-2 are rounded to the next higher 1.0 man-month in calculating these requirements.

(f) Based on a constant loading of 15 people through month 19 following reactor shutdown, with additional personnel added as required to meet schedule demands during that period, and diminishing thereafter as the schedule allows.

(g) Based on a constant loading of 55 people through month 21 following reactor shutdown, diminishing as the schedule allows, except during months 30 through 37 when 12 trained personnel are maintained to meet the requirements during months 38 and 39.

(h) Based on a constant loading of 13 people through month 39 following reactor shutdown, with additional personnel added as required to meet schedule demands during that period.

- Supervisors, including the shift engineers, the craft supervisors, and the senior health physics technicians, are staffed one per shift on a two-shift, 5-day-week basis, except for additional senior health physics technicians as required for comprehensive and final radiation surveys.
- A basic force of trained crew leaders is augmented by locally available craft foremen as required.
- Utility operators are specially trained for their functions and are not readily available on the open job market; therefore, a group large enough to meet the peak manpower requirements during active decommissioning is trained during the planning and preparation phase and maintained until the schedule allows a permanent reduction.
- Laborers and craftsmen are readily available from the open job market and are hired as required.
- A basic force of health physics technicians is augmented by special contract service health physics technicians to meet schedule peaking requirements. Salary costs for these extra health physics technicians are assumed the same as for the regular ones.

The total staff labor requirements shown in Table I.2-3 for the decommissioning workers during the 4 years following reactor shutdown exceed the overall decommissioning worker requirements presented in Table I.2-2 because of the above assumptions. This "excess" manpower is assumed used for the numerous, small, unspecified work items that accompany an activity of this magnitude.

I.3 COSTS FOR IMMEDIATE DISMANTLEMENT

The costs for immediate dismantlement of the reference BWR are presented in this section. These costs are all adjusted to early 1978 to provide consistency within this series of NRC nuclear facility decommissioning studies.⁽²⁾ The estimated costs and the other possible costs for immediate dismantlement of the reference BWR are summarized and totaled in Table I.3-1.

The total cost for immediate dismantlement is estimated at about \$43.6 million, including a 25% contingency. Costs for disposal of radioactive materials, staff labor, and energy (at approximately 25%, 50%, and 10% of the total,

TABLE I.3-1. Summary of Estimated Costs for Immediate Dismantlement

Cost Category	Estimated Costs (\$ millions)(a,b)	Percent of Total(c)
Disposal of Radioactive Materials		
Neutron-Activated Materials	2.300	
Contaminated Materials	4.909	
Radioactive Wastes(d)	1.469	
Total Disposal Costs	8.678	24.9
Staff Labor	17.561	50.4
Energy	3.519	10.1
Special Tools and Equipment	2.016	5.8
Miscellaneous Supplies	1.859	5.3
Specialty Contractors(e)	0.356	1.0
Nuclear Insurance	0.800	2.3
License Fees	0.051	0.1
Subtotal	34.840	100.0
Contingency (25%)	8.710	
Total, Immediate Dismantlement Costs	43.550	
Other Possible Costs		
Spent Fuel Shipment	3.788(f)	
Facility Demolition and Site Restoration	13.244	
Deep Geologic Disposal of Highly Activated Materials	0.848(g)	
Fuel Channel Disposal	0.617(h)	
Subtotal	18.497	
Contingency (25%)	4.624	
Total, Other Possible Costs	23.121	

(a) Costs adjusted to early 1978.

(b) The number of significant figures shown is for computational completeness and does not imply accuracy to the nearest \$1000.

(c) Individually rounded to the nearest 0.1%.

(d) Includes both wet solid wastes and dry solid wastes.

(e) Includes explosives, temporary radwaste, and environmental monitoring services.

(f) Does not include costs for handling at the reactor or costs for handling and storage at the repository; if required, shipment by special train costs an additional estimated \$2.451 million, maximum.

(g) Incremental cost in addition to the cost for shallow-land disposal of these materials; the maximum additional cost for shipment by special train is estimated at \$1.254 million.

(h) Alternate deep geologic disposal costs an estimated \$1.047 million; shipment by special train to the deep geologic disposal facility increases this cost by an estimated maximum cost of \$0.456 million.

respectively) are the major contributors to the total cost for immediate dismantlement. Combined costs for special tools and equipment and for miscellaneous supplies make up an additional 11% of the total; combined costs for specialty contractors, nuclear insurance, and license fees contribute about 4% of the total.

The total cost for other possible immediate dismantlement requirements (i.e., spent fuel shipment, facility demolition and site restoration, deep geological disposal of highly activated materials, and fuel channel disposal) is estimated at approximately \$23 million, including a 25% contingency. This total cost could go as high as nearly \$29 million depending on the additional requirements specified in the footnotes to Table I.3-1.

Detailed cost data for the individual cost categories shown in Table I.3-1 are presented and discussed in the following subsections.

I.3.1 Costs for Disposal of Radioactive Materials

Three distinct types of radioactive materials in the reference BWR require disposal during immediate dismantlement: 1) neutron-activated materials, 2) contaminated materials, and 3) radioactive wastes. The total cost for disposal of these materials is about \$8.7 million and is approximately 25% of the total immediate dismantlement cost. The disposal cost includes the container, transportation, and burial costs, but does not include the direct labor costs for removing and packaging these materials. Labor costs are incurred regardless of the decommissioning activity and are discussed later (Section I.3.2). Table I.3-2 contains a summary of the number of shipments, the burial volumes, and the disposal costs for the radioactive materials.

The disposal costs of the three types of radioactive materials are discussed in more detail in the following subsections.

TABLE I.3-2. Summary Information Regarding Disposal of Radioactive Materials from Immediate Dismantlement

<u>Radioactive Material Category</u>	<u>Number of Shipments</u>	<u>Burial Volumes (m³)(a)</u>	<u>Disposal Costs (\$ millions)(b)</u>
Neutron-Activated Materials			
Metal	303	138	2.174 ^(c)
Concrete	14	90	0.126
Contaminated Materials			
Metal	676	15,543	4.319
Concrete	130	1,676	0.590
Radioactive Wastes			
Wet Solid Wastes	286	814	1.073
Dry Solid Wastes	86	678	0.396
Totals	1,495	18,939	8.678

(a) Includes disposable containers if required.

(b) Adjusted to early 1978, the number of figures shown is for computational completeness and does not imply accuracy to the nearest \$1000.

(c) An additional cost of \$0.848 million is required if deep geologic disposal of highly activated materials is necessary.

I.3.1.1 Neutron-Activated Materials

All of the neutron-activated materials are contained in the reactor vessel, the vessel internals, and the sacrificial shield (see Section E.1 of Appendix E), and are located inside Primary Containment. Table I.3-3 gives a detailed breakdown of the masses, the radioactivity contents, the number of pieces after segmenting, the number of packaging containers, and the disposal costs for the neutron-activated components (except the fuel channels) in the reference BWR.

The total radioactivity expected to be present in the neutron-activated materials is approximately 6.6 million curies. The packaged materials require an estimated 317 overweight truck shipments to a shallow-land burial facility and occupy an estimated 228 m³ of space at the burial facility. The total estimated cost for disposal of the neutron-activated materials in a shallow-land burial facility is about \$2.3 million.

There is a possibility that the more highly activated materials may require disposal in a deep geologic disposal facility. This alternative is discussed in Section I.3.9.3.

The fuel channels are assumed to be shipped together with the spent reactor fuel to an offsite repository. However, in the event that this is not

TABLE I.3-3. Costs for Disposal of Neutron-Activated Materials

Component ^(a)	Estimated Mass (kg) ^(b)	Estimated Radioactivity Content (Ci)	No. of Pieces	Number of Disposable Containers ^(c)	Container Costs (\$) ^(d)	Cask Rental Costs (\$) ^(e)	Number of Shipments	Transport Costs (\$) ^(f)	Handling Surcharges (\$) ^(g)	Burial Volumes (m ³) ^(h)	Burial Costs (\$) ⁽ⁱ⁾	Liner Sur- charges (\$) ^(j)	Curie Sur- charges (\$) ^(k)	Total Disposal Costs (\$) ^(m)
Steam Separator Assembly ⁽ⁿ⁾	9 500	9 600	951	28p	12 600	14 000	14	26 880	5 600	10	940	36 400 ^(o)	7 320	103 740
Fuel Support Pieces	5 420	700	740	14p ^(p)	6 300	7 000	7	13 440	2 800 ^(o)	5	470	7 700 ^(o)	0	37 710
Control Rods and In-Core Instruments ⁽ⁿ⁾	19 810	189 000	610	8p ^(p)	48 000	27 600	8	15 360	12 310 ^(o)	15	1 400	42 810 ^(o)	11 780	159 260
Control Rod Guide Tubes ⁽ⁿ⁾	3 500	100	370	12p	5 400 ^(q)	6 000	6	11 520	2 400	4	380	3 900 ^(o)	0	29 600
Jet Pump Assemblies ⁽ⁿ⁾	6 000	20 000	110	40p	38 400 ^(q)	20 000	20	38 400	8 000	14	1 310	85 000 ^(o)	6 700	197 810
Top Fuel Guide	2 300	30 100	408	72p	111 600 ^(q)	36 000	72	138 240	14 400	24	2 250	153 000 ^(o)	22 030	477 520
Core Support Plate	18 500	650	258	31p	13 950	15 500	16	30 720	6 200	11	1 030	10 080 ^(o)	0	77 480
Core Shroud	32 000	6 300 000	558	13v + 127c	63 000	70 000	140	268 800	28 000	47	4 400	228 850 ^(o)	356 510	1 019 560
Reactor Vessel Wall ^(n,r)	157 700	2 160	576	22p ^(s)	9 900	11 000	20	38 400	4 400	8	680	7 150	0	71 530
Sacrificial Shield ⁽ⁿ⁾	272 000	170	14	14w ^(p)	84 000	0	14	26 880	5 600	90	9 530	0	0	126 010
Totals ^(m)	526 730	6 552 480	4 595	381	393 150	207 100	317	608 640	89 710	228	22 390	574 890	404 340	2 300 220

- (a) Listed in order of removal.
- (b) Estimated from volumes presented in Table E.1-6 in Appendix E.
- (c) Assumed to be 0.33-m³ steel cask liners, unless otherwise indicated; estimated on the following bases: p = packaging arrangement; v = volume at 60% packaging fraction; c = curie limit of 50,000 Ci per container; w = maximum weight per shipment.
- (d) Based on information in Section M.2 of Appendix M.
- (e) Based on Table M.3-1; assumes 5 days per shipment.
- (f) Based on Table M.4-4, for overweight shipments.
- (g) Based on cask handling fee in Table M.5-1, except as otherwise noted.
- (h) Includes the disposable container; rounded to nearest whole m³.
- (i) Based on Table M.5-1; based on volume only; rounded to nearest \$10.
- (j) Based on Table M.5-1, except as otherwise noted; based on the estimated dose rate at the container surface.
- (k) Based on Table M.5-1 and a per-shipment curie inventory.
- (m) The number of figures shown is for computational completeness only and does not imply accuracy to that many significant figures.
- (n) Includes only the neutron-activated portion of the component, rest shipped as contaminated material (see Table I.3-4).
- (o) Based on cost estimate information supplied by Nuclear Engineering Co., Louisville, KY (minus 30% for escalation since September 1977).
- (p) Specially constructed steel boxes.
- (q) Includes costs for lead shielding.
- (r) Includes additional mass to provide sufficient shielding for the core shroud packaging arrangement.
- (s) Includes segments of the neutron-activated portion of the reactor vessel that cannot be used as shielding for the core shroud.

TABLE I.3-3. Costs for Disposal of Neutron-Activated Materials

possible, a presentation of the disposal costs for these components is contained in Section I.3.9.4.

I.3.1.2 Contaminated Materials

Contaminated materials in the reference BWR are assumed to include a lot of the piping and equipment located in the Reactor Building/Primary Containment, the Turbine Generator Building, and the Radwaste and Control Building. In addition, many concrete surfaces in these three buildings are assumed to be contaminated and require surface removal to a depth of about 50 mm. Breakdowns of the disposal costs for contaminated materials are given in Table I.3-4.

Approximately 8600 Ci (see Section E.2 of Appendix E) of radioactivity are removed with the contaminated materials. These materials require an estimated 806 truck shipments to and an estimated 17,219 m³ of space (including the disposable containers, as required) at a shallow-land burial site. The total disposal cost for contaminated materials from the reference BWR is estimated to be about \$4.9 million.

Twenty large, contaminated heat exchangers in the reference BWR require special segmenting for shipping and burial. These are the two RHR heat exchangers in the Reactor Building and the 14 low-pressure feedwater heaters, the two moisture separator reheaters, and the two high-pressure feedwater heaters in the Turbine Generator Building. To comply with both overweight shipping limits using standard vehicles and burial site handling capabilities (see Sections M.4 and M.5 of Appendix M), these heat exchangers require segmenting into 4, 42, 26, and 8 individually contained packages, respectively, as shown in Table I.3-4. An alternative to this, in which special equipment is used to transport the maximum allowed weights over the public highways as well as handle them at the burial site, was analyzed by William N. Lampson of Neil F. Lampson, Inc., Kennewick, Washington. The heat exchangers were assumed segmented, as required, into 35 individual packages to meet the weight limits. However, the estimated total cost for transporting and handling, but not packaging, the 35 maximum-weight packages (ranging from \$437,500 to \$507,500 in 1979 dollars) is approximately twice the combined packaging (container), transportation, and handling costs (\$238,930 in 1978 dollars) shown in Table I.3-4 for these heat exchangers.

TABLE I.3-4. Costs for Disposal of Contaminated Materials

Location	Estimated Mass (kg)	Number of Disposable Containers	Container Costs (\$)	Number of Shipments	Transport Costs (\$)	Handling Costs (\$)	Burial Volumes (m ³)	Burial Costs (\$)	Total Disposal Costs (\$)
Component(a)	(b)	(c)	(d)	(e)	(f)	(g)	(h)	(i)	(j)
Primary Containment									
Steam Dryer	38 560	11	4 400	3	5 760	0	40	3 750	13 910
Steam Separator Assembly (k)	42 000	26	10 400	3	5 760	0	95	8 890	25 050
Misc. Reactor Vessel Internals (k)	69 600	26	10 400	4	7 680	0	95	8 890	26 970
Reactor Vessel									
Top Head	90 700	23	9 200	5	9 600	0	84	7 860	26 660
Refueling Bellows	32 480	10	4 000	2	3 840	0	37	3 460	11 300
Wall (k)	357 300	82	32 800	20	38 400	0	299	27 980	99 180
Bottom Head	81 900	20	8 000	4	7 680	0	73	6 830	22 510
Piping (and Valves) (m)									
Mirror Insulation	19 500	19	7 600	2	3 840	0	70	6 550	17 990
Carbon Steel	794 500	277	118 000	50	96 000	0	1 081	101 160	315 960
Stainless Steel	44 240	20	29 600	10	19 200	0	270	25 270	74 070
Whip Restraints and Hangers (m)	148 330	74	29 600	10	19 200	0	270	25 270	74 070
Equipment									
Control Rod Drive (k)	100 960	44	17 600	8	15 360	0	161	15 070	48 030
Reactor Water Recirculation	87 180	1 + 5 (n)	5 600	5	9 600	3 740	88	8 240	27 180
Sacrificial Shield (k)									
Radial Beams (and Floor Gratings) (m)	704 400	0 + 38 (n)	17 480	38	72 960	14 960	310	29 020	133 520
Contaminated Concrete	348 070	138	55 200	19	36 480	0	503	47 070	138 750
	374 300	86	34 400	22	42 240	0	313	29 290	105 930
HVAC Equipment (m)									
Electrical Equipment (m)	26 550	54	21 600	3	5 760	0	197	18 430	45 790
	254 010	60	24 000	15	28 800	0	218	20 400	73 200
Subtotals, Primary Containment (j)	3 614 580	971 + 43 (n)	411 080	213	408 960	17 800	3 934	368 160	1 206 000
Reactor Building									
Piping (and Valves)									
Aluminum	1 020	1							
Carbon Steel	322 750	154	66 000	28	53 760	0	601	56 240	176 000
Stainless Steel	25 370	10							
Equipment									
Containment Atmosphere Control	10 890	0 + 2 (n)	2 820	1 (o1)	1 060 (o1)	150	48	4 490	8 520
Condensate (Nuclear Steam)	3 640	2	800	1 (o1)	360 (o1)	0	8	750	1 910
Control Rod Drive	71 210	16 + 2 (n)	7 720	4	7 680	0	72	6 740	22 140
Equipment Drain (Radioactive)									
Floor Drain (Radioactive)	1 330	1	400	1 (o1)	170 (o1)	0	4	380	950
Fuel Pool Cooling and Cleanup	3 400	2	800	1 (o1)	330 (o1)	0	8	750	1 880
	12 880	2 + 2 (p)	1 200	1	1 920	0	11	1 030	4 150
High-Pressure Core Spray									
Low-Pressure Core Spray	27 840	1 + 2 (n)	1 650	2	3 840	510	17	1 590	7 590
MSIV Leakage Control	10 060	1 + 1 (n)	1 050	1 (o2)	930 (o2)	170	10	940	3 090
	410	-- (q)	--	--	--	--	--	--	--
Reactor Building Closed Cooling									
Reactor Core Isolation Cooling	27 200	3 + 3 (p)	1 800	2 (o2)	3 840	360	32	3 000	9 000
Reactor Refueling	8 830	2 + 1 (n)	1 480		820 (o2)	100	13	1 220	3 620
	30 160	14	5 600	2	3 840	0	51	4 780	14 220
Reactor Water Cleanup									
Residual Heat Removal	21 750	1 + 5 (p)	1 400	1	1 920	0	22	2 060	5 380
Traversing In-Core Probe	82 250	0+3 (n)+4 (p)	3 450	5	9 600	1 490	62	5 810	20 350
	1 810	2	800	1 (o2)	170 (o2)	0	8	750	1 720
Pool Liners (and Spent Fuel Racks) (r)									
Contaminated Concrete	346 700	22 + 15 (p)	13 760	18	34 560	5 610	381	35 650	89 580
HVAC Equipment (s)	495 700	92 + 16 (n)	48 000	28	53 760	1 440	434	40 610	143 810
Electrical Equipment (t)	73 030	152	60 800	9	17 280	0	554	51 840	129 920
	50 720	22	8 800	2	3 840	0	80	7 490	20 130
Subtotals, Reactor Building (j)	1 628 950	500 + 56 (n,p)	228 330	104	199 680	9 830	2 416	226 120	663 960

Location	Estimated Mass (kg)	Number of Disposable Containers	Container Costs (\$)	Number of Shipments	Transport Costs (\$)	Handling Costs (\$)	Burial Volumes (m ³)	Burial Costs (\$)	Total Disposal Costs (\$)
Component(a)	(b)	(c)	(d)	(e)	(f)	(g)	(h)	(i)	(j)
Turbine Generator Building									
Turbine (u)									
Condenser (u)	1 662 780	220 + 58 (n)	156 550	82	157 440	14 480	1 406	131 560	460 030
	1 570 000	500	200 000	125	240 000	0	1 820	170 300	610 300
Piping (and Valves)									
Aluminum	37 880	40							
Carbon Steel	1 168 180	426 + 27 (n)	213 520	83	159 360	0	2 013	188 360	561 240
Stainless Steel	7 700	3							
Equipment									
Air Removal	6 350	2	800	1 (o3)	880 (o3)	0	8	750	2 430
Condensate (Nuclear Steam)	219 770	32+9 (n)+3 (p)	25 300	13	24 960	2 210	363	33 970	86 440
Low-Pressure Feedwater Heaters	738 000	2 + 42 (p)	7 100	42	80 640	15 270	737	68 970	171 980
Condenser Off Gas Treatment									
Equipment Drain (Radioactive)	2 080	0 + 4 (p)	300	1 (o3)	290 (o3)	0	8	750	1 340
Floor Drain (Radioactive)	2 340	1	400	1 (o3)	320 (o3)	0	4	380	1 100
	1 940	1	400	1	270 (o3)	0	4	380	1 050
Heater Drain									
Main Steam	34 860	0 + 12 (p)	900	2	3 840	0	27	2 530	7 270
Moisture Separator Reheaters	32 220	0 + 3 (p)	600	2	3 840 (v)	560	71	6 650	11 650
	417 310	0 + 26 (p)	32 000	26	62 920 (v)	9 360	715	66 900	171 180
Miscellaneous Drain and Vent									
Reactor Feedwater Pumps	1 140	0 + 2 (p)	300	1 (o3)	160 (o3)	0	14	1 310	1 770
High-Pressure Feedwater Heaters	109 770	10 + 10 (n)	16 520	6	11 520	1 200	194	18 250	47 490
	146 960	0 + 8 (p)	3 600	8	15 360	3 000	121	11 320	33 280
Contaminated Concrete									
HVAC Equipment (s)	255 120	59	23 600	15	28 800	0	215	20 120	72 520
Electrical Equipment (t)	64 450	192	76 800	10	19 200	0	699	65 410	161 410
	36 760	15	6 000	2	3 840	0	55	5 150	14 990
Subtotals, Turbine Generator Building (j)	6 516 610	1 503 + 204 (n,p)	764 690	417	813 640	46 080	8 464	793 060	2 417 470
Radwaste and Control Building									
Piping (and Valves)									
Aluminum	19 050								
Carbon Steel	149 260	146	58 400	13	24 960	0	530	49 600	132 960
Stainless Steel	16 670								
Equipment									
Condensate Filter Demineralizer	47 340								
Condenser Off Gas Treatment	66 490								
Equipment Drain (Radioactive)	47 980								
Floor Drain (Radioactive)									
Fuel Pool Cooling and Cleanup	28 090	117 + 22 (n)	63 520	17	32 640	0	576	53 900	150 060
Misc. Waste (Radioactive)	3 530								
	49 800								
Process Waste (Radioactive)									
Reactor Water Cleanup	8 400								
	12 570								
Contaminated Concrete									
Miscellaneous Steel Structures (w)	488 160	104 + 7 (n)	44 900	27	51 840	0	404	37 810	134 550
HVAC Equipment (s)	19 440	11	4 400	1	1 920 (x)	0	40	3 750	10 070
Electrical Equipment (t)	69 510	208	83 200	12	14 160 (x)	0	757	70 840	168 200
	63 210	27	10 800	3	5 760	0	98	9 170	25 730
Subtotals, Radwaste and Control Building (j)	1 089 500	613 + 29 (n)	265 220	72	131 280	0	2 405	225 070	621 570
Totals (from all locations) (j)	12 849 640	3 587 + 332 (n,p)	1 669 320	806	1 553 560	73 710	17 219	1 612 410	4 909 000

(a) Listed in order of removal (i.e., main headings only) from each location.
(b) Obtained from Appendix C, or estimated from information in Appendices C and D or from data supplied by Washington Public Power Supply System.
(c) Assumed to be 1.2-m by 1.2-m by 2.4-m plywood boxes, unless otherwise indicated.
(d) Based on information in Section M.2 of Appendix M.
(e) Assumed to be overweight shipments, unless otherwise indicated.
(f) Based on Table M.4-4.
(g) Based on Table M.5-1.
(h) Includes the disposable container where required; rounded to the next whole m³.
(i) Based on Table M.5-1, based on an assumed container surface dose rate of <0.20 R/hr, rounded to next highest \$10.
(j) The number of figures shown is for computational completeness and does not imply accuracy to that many significant figures.
(k) Includes only the contaminated portion of the component, rest shipped as neutron-activated material (see Table I.3-3).
(l) Includes all such equipment inside Primary Containment.
(m) Specially fabricated plywood boxes.
(n) Equipment that is assumed grouped together in a single overweight shipment (three such shipments designated as o₁, o₂, and o₃), individual transportation cost calculated based on percentage of total shipment mass.
(p) Equipment or segments of equipment on which external openings or surfaces are capped or covered and seal-welded to form self-contained packages.
(q) Indicates equipment packaged and shipped with other equipment.
(r) Includes all the spent fuel racks, even though those not required for the last core load are assumed removed during the planning and preparation phase.
(s) Includes all exhaust equipment plus HEPA filters installed during the planning and preparation phase, where applicable.
(t) Assumed to be 5% of the total cable trays and hangers.
(u) Includes noncontaminated material that is handled as salvage.
(v) Overweight-oversize shipments.
(w) Includes steel stairs, gratings, and ladders assumed to be contaminated.
(x) Legal-weight shipments.

TABLE I.3-4. Costs for Disposal of Contaminated Materials

I.3.1.3 Radioactive Wastes

While not a prime "product," radioactive wastes (radwastes) result directly from the immediate dismantlement of the reference BWR. Two categories of radioactive wastes are defined: wet solid wastes and dry solid wastes.

Wet solid wastes result from the processing of chemical decontamination solutions and contaminated water volumes. These wastes include concentrator bottoms, filter sludges, and spent demineralizer resins, as well as the neutralized chemical solutions from decontamination of the contaminated drain systems. These wastes are assumed mixed with a cement solidifying agent and encapsulated in a steel cask liner prior to shipping to a shallow-land burial facility. The disposal cost data for the wet solid wastes generated during immediate dismantlement are presented in Table H.5-10 in Appendix H and are not repeated in detail here. An estimated 286 truck shipments, 816 m³ burial space, and \$1,073,400 are required for disposal of the wet solid wastes.

Dry solid wastes include discarded contaminated materials such as plastic sheeting, rags, and anticontamination clothing. They are expected to occur as a result of most of the tasks specified in Section I.2 and are estimated on a taskwise basis. The dry solid wastes are compacted as much as possible to reduce their volume.

Large volumes of dry solid wastes are expected to be produced during immediate dismantlement. Uncompacted volumes of 1500 m³ from the Reactor Building, 1012 m³ from the Turbine Building, and 874 m³ from the Radwaste and Control Building are estimated to be collected. These volumes are assumed to be compacted five-fold and packaged in 1428, 964, and 832 0.21-m³ standard steel drums, respectively. Of the total number of drums, 1000 are assumed to require shielding during shipment, with the remainder shipped unshielded in closed trucks. The two types are shipped separately, since, because of the ⁶⁰Co content, those requiring shielding also require Type B packaging and shipping. Table I.3-5 gives a breakdown of the disposal costs for the dry solid wastes. An estimated 86 overweight truck shipments (72 shielded, 14 unshielded) are required to transport the compacted, packaged, dry solid wastes to a shallow-land burial facility, where they occupy an estimated 678 m³ of space. The total disposal

TABLE I.3-5. Costs for Disposal of Dry Solid Wastes

Source (Building)	Number of Disposable Containers (a)	Container Costs (\$) (b)	Estimated No. Requiring Shielding	Cask Rental Costs (\$) (c)	Number of Shipments Shielded (d) / Unshielded	Trans- portation Costs (\$) (e)	Handling Costs (\$) (f)	Burial Volumes (m ³) (g)	Burial Costs (\$) (h)	Total Disposal Costs (\$) (i)
Reactor	1 428	28 560	443	31 500	32/6	72 960	12 600	300	29 210	174 830
Turbine Generator	964	19 280	299	21 500	21/4	48 000	8 600	203	19 720	117 100
Radwaste and Control	832	16 640	258	18 500	19/4	44 160	7 400	175	17 020	103 720
Totals (i)	3 224	64 480	1 000	71 500	72/14	165 120	28 600	678	65 950	395 650

(a)Based on a 5:1 compaction of estimated waste volumes into standard 0.21-m³ steel drums; estimated on a taskwise assessment of expected dry solid waste generation rates.

(b)Based on Table M.2-1 in Appendix M.

(c)Based on Table M.3-1; assumes a maximum of seven containers per cask and 5 days per shipment.

(d)Assumes two cask loads per shipment.

(e)Based on Table M.4-4 for overweight shipments.

(f)Based on Table M.5-1, cask handling fee.

(g)Includes the disposable container; rounded to next whole m³.

(h)Based on Table M.5-1; surface dose rates assumed to be 0.21 to 1.00 R/hr for those drums requiring shielding during shipment, <0.2 R/hr for all others; rounded to the next highest \$10.

(i)The number of figures shown is for computational completeness only and does not imply accuracy to that many significant figures.

cost for the dry solid wastes from the immediate dismantlement of the reference BWR is estimated at \$395,650.

The sum of the disposal costs for the wet solid wastes and the dry solid wastes is \$1.469 million, which represents the disposal cost for the radioactive wastes.

I.3.2 Costs for Staff Labor

The costs for staff labor during immediate dismantlement are shown in detail in Table I.3-6. More than 50% of the total immediate dismantlement cost is associated with the staff labor requirements (see Table I.2-3). A total staff labor cost of about \$17.6 million is estimated for immediately dismantling the reference BWR. Specialty contractor labor is not included in this total.

The dedicated manpower costs for the immediate dismantlement tasks are given in Table I.3-7. These costs are attributed to manpower that is specifically assigned to the tasks, and do not include either non-dedicated personnel or management and support staff (see Figure H.1-1 in Appendix H).

I.3.3 Costs for Energy

The costs for energy during immediate dismantlement are presented in Table I.3-8, together with the estimated usage of both electricity and fuel oil. The usage of both energy forms is estimated on a detailed analysis of the requirements for the essential systems presented in Table H.4-1 in Appendix H and the immediate dismantlement tasks and schedule.

A total of 106,400 MWh of electricity, costing \$1,590,000, and 14,570 m³ of fuel oil, costing \$1,923,290, are estimated to be used during immediate dismantlement. The total cost for energy is about \$3.5 million and represents about 10% of the total immediate dismantlement cost.

I.3.4 Costs for Special Tools and Equipment

Based on the information presented in Table I.1-1, the estimated costs for the special tools and equipment that are required for immediate dismantlement of the reference BWR are presented in Table I.3-9. The estimated total cost for special tools and equipment is approximately \$2.0 million and is approximately 6% of the total immediate dismantlement cost.

TABLE I.3-6. Costs for Staff Labor During Immediate Dismantlement

Position	Time Relative to Final Reactor Shutdown (year)						Total Staff Labor Costs ^(b) (\$ thousands)
	-2	-1	1	2	3	4	
	Annual Staff Labor Costs (\$ thousands)(a,b)						
<u>Management and Support Staff</u>							
Decommissioning Superintendent	22.4	74.5	74.5	74.5	74.5	59.6	380.0
Secretary	20.3	40.6	60.9	60.9	60.9	32.5	276.1
Clerk	0	20.3	40.6	40.6	40.6	10.2	152.3
Decommissioning Engineer	63.5	63.5	63.5	63.5	63.5	50.8	368.3
Assistant Decommissioning Engineer	43.8	43.8	43.8	43.8	43.8	17.6	236.6
Radioactive Shipment Specialist	0	32.9	32.9	32.9	32.9	13.2	144.8
Procurement Specialist	9.2	30.7	30.7	30.7	30.7	12.3	144.3
Tool Crib Attendant	0	0	46.4	46.4	46.4	18.6	157.8
Control Room Operator	0	0	145.0	145.0	145.0	60.9	495.9
Security Supervisor	0	0	32.9	32.9	32.9	16.5	115.2
Security Shift Supervisor	0	0	153.5	153.5	153.5	76.8	537.3
Security Patrolman	0	0	830.7	596.4	276.9	138.5	1 842.5
Contracts and Accounting Specialist	11.5	38.4	38.4	38.4	38.4	30.8	195.9
Health and Safety Supervisor	50.4	50.4	50.4	50.4	50.4	40.4	292.4
Health Physicist	0	19.7	39.4	39.4	39.4	19.7	157.6
Protective Equipment Attendant	0	0	46.4	46.4	46.4	23.2	162.4
Industrial Safety Specialist	13.1	43.8	43.8	43.8	43.8	21.9	210.2
Quality Assurance Supervisor	13.1	43.8	43.8	43.8	43.8	35.1	223.4
Quality Assurance Engineer	19.7	78.8	39.4	39.4	39.4	19.7	236.4
Quality Assurance Technician	0	11.6	92.8	92.8	92.8	46.4	336.4
Consultant (Safety Review)	33.5	33.5	33.5	33.5	33.5	20.1	187.6
<u>Subtotals, Management and Support Staff</u>	<u>300.5</u>	<u>626.3</u>	<u>1 983.3</u>	<u>1 749.0</u>	<u>1 429.5</u>	<u>764.8</u>	<u>6 853.4</u>
<u>Decommissioning Workers</u>							
Shift Engineer	43.8	87.6	87.6	87.6	87.6	43.8	438.0
Crew Leader	0	37.3	585.7	447.6	74.6	7.5	1 152.7
Utility Operator	0	86.8	1 490.5	1 344.2	336.1	89.5	3 347.1
Laborer	0	0	334.2	512.9	422.2	137.3	1 406.6
Craft Supervisor	0	24.8	78.8	78.8	78.8	31.6	292.8
Craftsman	0	271.0	607.1	1 008.2	531.2	108.4	2 525.9
Senior Health Physics Technician	0	32.9	82.3	72.4	69.1	32.9	289.6
Health Physics Technician	0	75.3	359.0	374.0	351.4	95.4	1 255.1
<u>Subtotals, Decommissioning Workers</u>	<u>43.8</u>	<u>615.7</u>	<u>3 625.2</u>	<u>3 925.7</u>	<u>1 951.0</u>	<u>546.4</u>	<u>10 707.8</u>
<u>Totals</u>	<u>344.3</u>	<u>1 242.0</u>	<u>5 608.5</u>	<u>5 674.7</u>	<u>3 380.5</u>	<u>1 311.2</u>	<u>17 561.2</u>

(a) Calculated as the product of the data given in Table I.2-3 and the corresponding data given in Table M.1-1 in Appendix M; rounded to the next higher \$100.

(b) The number of significant figures shown is for computational completeness and does not imply accuracy to the nearest \$100.

TABLE I.3-7. Dedicated Manpower Costs for Immediate Dismantlement

Building	Task	Dedicated Manpower Costs \$(a)
<u>Reactor Building/Primary Containment</u>		
1	Install HEPA Filters	74 320
2	Comprehensive Radiation Survey	12 550
3	Discharge and Ship Fuel	54 200
4	Remove Dryer and Separator	36 130
5	Drain Suppression Pool to Radwaste, Water-Jet Clean	241 790
6	Remove Reactor Vessel Internals	146 770
7	Ship Activated Reactor Vessel Internals	837 370
8	Drain Reactor Well Pool to Radwaste, Water-Jet Clean	60 450
9	Chemical Decon Reactor Water Recirculation and Cleanup Systems	102 740
10	Cleanup, Stage and Shield Hot Spots in Primary Containment	12 950
11	Enlarge Suppression Chamber Access	6 570
12	Remove Reactor Vessel	63 220
13	Ship Activated Reactor Vessel Segments	322 070
14	Remove Primary Containment Piping and Equipment	533 920
15	Remove Sacrificial Shield and Radial Beams	217 790
16	Remove Contaminated Concrete from Primary Containment	51 790
17	Remove HVAC and Electrical Equipment from Primary Containment	17 660
18	Drain Contaminated Systems to Radwaste	90 560
19	Chemical Decon Residual Heat Removal, Low-and High-Pressure Core Spray Systems	51 260
20	Remove Reactor Building Piping	793 510
21	Drain Dryer and Separator Pool to Radwaste, Water-Jet Clean	60 450
22	Chemical Decon Drain Systems	51 260
23	Drain Spent Fuel Pool to Radwaste, Water-Jet Clean	120 900
24	Chemical Decon Fuel Pool Cooling and Cleanup System	20 630
25	Remove Reactor Building Equipment	132 480
26	Remove Liners from Spent Fuel Pool, Reactor Well, and Dryer and Separator Pool	35 330
27	Remove Reactor Building Contaminated Concrete	43 160
28	Remove HVAC and Electrical Systems from Reactor Building	26 500
29	Final Radiation Survey	4 180
<u>Subtotal, Reactor Building</u>		<u>4 222 510</u>
<u>Turbine Generator Building</u>		
1	Install HEPA Filters	204 370
2	Comprehensive Radiation Survey	4 180
3	Cleanup and Stage	8 630
4	Remove Turbine	65 740
5	Drain Contaminated Systems to Radwaste, Water-Jet Clean	181 340
6	Condensate Storage Tanks	248 100
7	Remove Condenser	181 340
8	Drain Condenser to Radwaste, Water-Jet Clean	46 660
9	Chemical Decon Drain Systems	321 550
10	Remove Piping	93 640
11	Remove Equipment	25 900
12	Remove Contaminated Concrete	13 150
13	Remove HVAC and Electrical Systems	2 090
13	Final Radiation Survey	2 090
<u>Subtotal, Turbine Generator Building</u>		<u>1 396 690</u>
<u>Radwaste and Control Building</u>		
1	Comprehensive Radiation Survey and Release of Support Facilities, Comprehensive Radiation Survey of Radwaste and Control Building	20 910
2	General Cleanup	8 830
3	Chemical Decon Drain Systems	104 780
4	Chemical Decon Equipment, Water-Jet Clean	209 790
5	Remove Piping	775 060
6	Install Temporary Radwaste System	4 420
7	Remove Equipment	827 690
8	Remove Contaminated Concrete	34 530
9	Remove Miscellaneous Steel Structures	17 660
10	Remove HVAC and Electrical Systems	35 330
11	Final Radiation Survey	4 180
<u>Subtotal, Radwaste and Control Building</u>		<u>2 043 180</u>
Total, Dedicated Manpower Cost		7 662 380

(a) 1) Calculated as the sum of the products of the data presented in Table I 2-1 (converted to man-years) and the corresponding data presented in Table M 1-1 in Appendix M, rounded to the nearest \$10 2) Do not include non-dedicated personnel (defined in Table I 2 1) or management and support staff (defined in Figure H 1-1 in Appendix H) 3) The number of significant figures shown is for computational completeness and does not imply accuracy to the nearest \$10

TABLE I.3-8. Costs for Energy During Immediate Dismantlement

Energy Form System(s) or Equipment	Estimated Energy Usage ^(a)	Costs (\$) ^(b)
Electricity	(MWh)	
HVAC		
Main Buildings ^(c)	52 150	782 250
Control Tower	1 120	16 800
Other Buildings	7 090	106 350
Lighting	9 600	144 000
Communications	370	5 550
Security	2 460	36 900
Fire Protection	130	1 950
Compressed Air	2 300	34 500
Cranes and Hoists	1 150	17 250
Domestic Water	15 180	227 700
Contaminated Water ^(d)	9 780	146 700
Decommissioning Equipment	5 070 ^(e)	76 050
Subtotals, Electricity	106 400	1 596 000
Fuel Oil	(m ³)	
Building Heating	14 375 ^(f)	1 897 500
Decontamination Solution Concentrator	195 ^(g)	25 790 ^(g)
Subtotals, Fuel Oil	14 570	1 923 290
Total Energy Cost		3 519 290

(a)Based on a system/component-wise analysis of specific requirements, unless otherwise noted; rounded to the nearest 10 MWh for electricity and to the nearest whole m³ for fuel oil.

(b)Based on Section M.6 of Appendix M.

(c)Includes the Reactor Building, the Turbine Generator Building, and the radwaste area of the Radwaste and Control Building.

(d)Includes all such systems listed in Appendix C for all anticipated operations (decontamination, draining, cooling, clean-up, radwaste processing, etc.); includes appropriate information from Table H.5-9 in Appendix H.

(e)Estimated at about 5% of the total.

(f)Assumes four heating periods during immediate dismantlement.

(g)From Table H.5-9.

**TABLE I.3-9. Costs for Special Tools and Equipment
for Immediate Dismantlement**

Item	Estimated Number Required ^(a)	Estimated Costs ^(b) (\$ thousands)
Underwater Manipulator	1	1 000
Underwater Plasma-Arc Torch	2	40
Underwater Oxyacetylene Torch	2	10
Arc Saw	1	100
Portable Plasma-Arc Torch	2	40
Portable Oxyacetylene Torch	10	10
Remote-Controlled Oxyacetylene Torch	4	20
Guillotine Pipe Saw	10	40
Power-Operated Reciprocating Hacksaw	10	5
Closed-Circuit, High-Resolution Television	2	15 ^(c)
Underwater Lights and Periscopes	As Required	5 Total
Underwater Tools (e.g., Impact Wrenches, Bolt Cutters, Tongs)	As Required	25 Total
Submersible Pump with Disposable Filter	5	5
High-Pressure Water Jet	2	40
Pipe Jumper	4	3
Mobile Chemical Decontamination Unit	5	100
Mobile Chemical Mixing and Heating Unit	4	10
Scaffolding	200	10
Safety Nets	As Required	25 Total
Power-Operated, Mobile, Scissors-Type Manlift	6	192
Power-Operated, Mobile, Articulated-Arm Manlift	3	27
9100-kg Mobile Hydraulic Crane	3	60
9100-kg Forklift	6	80 ^(d)
Rigging Materials (e.g., Chokers, Grapples, Winches)	As Required	25 Total
Concrete Drill with HEPA-Filtered Dust Collecting System	4	8
Concrete Surface Spaller	4	20
Front-End Loader (Highly Maneuverable, Light-Duty)	3	30
Vacuum Cleaner (HEPA-Filtered)	3	15
Portable Ventilation Enclosure	10	15
Supplied-Air Plastic Suit	250	15
Filtered-Exhaust Fan Unit	4	16
<u>Polyurethane Foam Generator</u>	2	<u>10</u>
Total Cost		2 016

(a)Based on Table I.1-1.

(b)Costs adjusted to early 1978.

(c)Estimated for modifications of existing systems.

(d)Assumes the availability of two forklifts from plant operations.

I.3.5 Costs for Miscellaneous Supplies

A variety of supplies are used during immediate dismantlement. These include expendable glass-fiber and HEPA filters, anticontamination clothing, cleaning and contamination control supplies (chemical agents, sweeping compounds, rags, mops, and plastic bags and sheeting), expendable hand tools, cutting and welding supplies (saw blades, torch gas, and welding rod), decontamination chemicals, and filter/demineralizer resins. The estimated costs for these items are given in Table I.3-10. The total estimated cost for miscellaneous supplies during immediate dismantlement of the reference BWR is about \$1.9 million and represents about 5% of the total immediate dismantlement cost.

TABLE I.3-10. Costs for Miscellaneous Supplies
During Immediate Dismantlement

<u>Item</u>	<u>Estimated Costs (\$ thousands)</u>
Filters	
Reactor Building and Turbine Generator Building HVAC Exhausts ^(a)	450
All Other Requirements	137
Anticontamination Clothing ^(b)	323
Cleaning and Contamination Control Supplies	345
Hand Tools	120
Cutting and Welding Supplies	60
Decontamination Chemicals ^(c)	335
<u>Filter/Demineralizer Resins^(c)</u>	<u>89</u>
Total	1 859

(a)Includes filters, housings, and ducting; see Section H.2.8 of Appendix H.

(b)Estimated at four changes per day per decommissioning worker. See Reference 3.

(c)From Table H.5-7, rounded to nearest \$1000.

I.3.6 Costs for Specialty Contractors

The estimated costs for specialty contractors are given in Table I.3-11. As discussed in Section H.3 of Appendix H, these specialty contractors perform explosives work, temporary radwaste handling, and environmental monitoring. The costs for a hauling contractor are not shown in this table, but are shown as "transportation costs" in Section I.3.1 for disposal of radioactive wastes.

The total cost for specialty contractors during immediate dismantlement, excluding the hauling contractor, is \$356,000, which is about 1% of the total immediate dismantlement cost.

TABLE I.3-11. Costs for Specialty Contractors
During Immediate Dismantlement

<u>Specialty Contractor^(a)</u>	<u>Cost Estimating Basis (\$/unit)</u>	<u>Estimated Costs (\$)</u>
Explosives	21 250/job ^(b)	21 250
Temporary Radwaste Handling	10 000/mo ^(c)	200 000
<u>Environmental Monitoring</u>	38 500/yr ^(d)	<u>134 750</u>
Total Cost		356 000

(a) Does not include the hauling contractor, which is shown as "transportation costs" in Section I.3.1 with a total cost of approximately \$2.9 million.

(b) Based on information supplied by Mr. Jerry Curry of JC Drilling and Contracting, Bellevue, Washington; rounded to the next higher \$50.

(c) Based on information presented on Page 10-4 of Reference 1 and corroborated with Mr. Ed Brooks of Chem-Nuclear Systems, Inc., Bellevue, Washington.

(d) Based on information presented on Page G-37 of Reference 1.

I.3.7 Costs for Nuclear Insurance

The costs for nuclear liability insurance during immediate dismantlement are estimated for an assumed policy limit of \$125 million carried through the dismantlement period and are shown in Table I.3-12. The total estimated cost for nuclear insurance is \$800,000, which represents a little more than 2% of the total immediate dismantlement cost.

TABLE I.3-12. Costs for Nuclear Liability Insurance
During Immediate Dismantlement(a)

<u>Year After Shutdown</u>	<u>Estimated Premium (\$ thousands)</u>
1	240
2	210
3	210
<u>4</u>	<u>140</u>
Total	800

(a)Based on information presented on
Page 10-8 of Reference 1.

I.3.8 Costs for Licensing Fees

The fees charged for licensing services performed by the NRC are delineated in 10 CFR Part 170.⁽⁴⁾ The costs of licensing fees during immediate dismantlement of the reference BWR are shown in Table I.3-13. The total cost for licensing services is \$50,800, which is approximately 0.1% of the total immediate dismantlement cost.

TABLE I.3-13. Costs for Licensing Fees During
Immediate Dismantlement(a)

<u>Licensing Service Category</u>	<u>Fee (\$)</u>	<u>Total Costs (\$)</u>
Facility License Amendment (Class IV)	12 300	24 600 ^(b)
Health, Safety, and Environmental Inspection	650/year	2 600 ^(c)
<u>Safeguards Inspection</u>	11 800/year	<u>23 600^(d)</u>
Total		50 800

(a)From Reference 4.

(b)Based on two license amendments, one to obtain a possession-only license prior to dismantlement and one to terminate that license following dismantlement.

(c)Based on the time the license is in force (i.e., approximately 4 years for immediate dismantlement.

(d)Based on having spent fuel on site for 19 months (i.e., two yearly fees charged).

I.3.9 Other Possible Costs

Four additional categories of costs could figure into the total immediate dismantlement cost depending on how they are classified. In this study, they are presented separately since they cannot be clearly identified as belonging to immediate dismantlement. The tasks that require these costs are:

- shipment of the spent reactor fuel to an offsite repository
- demolition of the structures and restoration of the site
- alternative disposal of the highly activated materials in a deep geologic disposal facility
- disposal of the fuel channels.

Discussions of the costs associated with these tasks are presented in the following subsections.

I.3.9.1 Spent Fuel Shipment

Since the ultimate disposition of the spent reactor fuel is not known, it is assumed in this study that the 764 fuel assemblies from the final reactor core load are shipped, together with their fuel channels, to a repository located 2400 km by rail from the reference BWR. The assemblies are assumed to be shipped in IF-300 rail casks, using the fuel channels as protection instead of the normal shipping baskets. Each cask shipment is assumed to carry a maximum of 18 fuel assemblies and fuel channels, to be shipped using regular freight methods, and to require 18 days round trip. Thus, 43 cask shipments are necessary to ship all the fuel and channels. The total cask rental cost (see Table M.3-1 in Appendix M) is \$2,709,000, and the total transportation cost (see Section M.4.2 of Appendix M) is \$1,079,300. Therefore, the total estimated cost for shipping the spent fuel and channels to the assumed repository is \$3,788,300. This does not include either handling costs at the reactor or handling and storage costs at the repository. The unencapsulated volume of the spent reactor fuel with the fuel channels is approximately 67 m³.

In order to meet the schedule shown in Section I.2 with the 18-day-round-trip requirement stated above, two casks are assumed used for about 60% of the time, with one cask for the rest. This, however, does not affect the costs associated with spent fuel shipment, whether shipped individually or together.

If special trains are required, an additional cost of \$2,451,000 would be necessary for 43 single-cask shipments (see Section M.4.2). The use of more than one cask per shipment would decrease this cost inversely to the number of casks per shipment.

I.3.9.2 Facility Demolition and Site Restoration

The costs for demolishing the decontaminated and uncontaminated structures of the reference BWR are summarized in Table L.3-1 of Appendix L and are not repeated here. The total cost of \$13,244,000 (without contingency) includes labor, supplies, overheads, and profit, but not extraordinary insurance premiums, bonding, or state sales tax. Details of the cost estimates for this task, which is not required for termination of the facility nuclear license, are given in Section L.3 of Appendix L.

I.3.9.3 Deep Geologic Disposal of Highly Activated Materials

The information presented in Table I.3-3 is based on the assumption that all of the neutron-activated components are placed in a shallow-land disposal site. Because of the large initial amount of radioactivity, the long half-lives of ^{59}Ni and ^{94}Nb , and the potential biological hazard of ^{14}C , it is a possibility that the highly activated materials may require disposal as intermediate-level wastes in a deep geologic disposal facility, similar to the proposed disposal of high-level radioactive wastes and fuel cladding hulls that result from spent fuel reprocessing. An analysis is, therefore, made to estimate the additional cost of placing the highly activated materials in deep geologic storage rather than in shallow-land storage. This analysis is based on the proposed burial canisters (0.76 m dia. by 3.05 m long) and the postulated burial costs for intermediate-level wastes (\$7100/m³) from Reference 5.

The shroud head plate (the lower part of the steam separator assembly), the fuel support pieces, the control rods, the in-core instruments, the jet pump assemblies, the top fuel guide, the core support plate, and the core shroud are considered to be sufficiently activated to require possible deep geologic disposal. The steam separator risers, the control rod guide tubes, the reactor vessel, and the sacrificial shield are considered to be mildly activated and require only shallow-land disposal.

The highly activated components have a calculated mass of 90,600 kg and, with a density of 7860 kg/m^3 , a theoretical volume of about 11.5 m^3 . Assuming a 15% packing density in the burial canister (about 1.2-m^3 internal volume), 64 canisters are required to package these components. The total cost for the canisters (see Table M.2-1 in Appendix M) is \$320,000.

Shipment to the deep geologic disposal facility is assumed to use regular freight methods and to be in IF-300 railroad casks, with each cask carrying a maximum of three canisters. The one-way rail distance is again assumed to be 2400 km, and the round trip is again assumed to take 18 days. A minimum of 22 single-cask shipments are required. The total cask rental cost (see Table M.3-1 in Appendix M) is \$1,386,000, and the total transportation cost (see Section M.4.2 of Appendix M) is \$552,200.

The total cost for placing the 64 loaded canisters (the total volume is 88.9 m^3) in deep geologic disposal, at $\$7100/\text{m}^3$, is \$631,190.

The individual estimated costs for deep geologic disposal of the highly activated materials are summarized and totaled below:

<u>Canister Cost (\$)</u>	<u>Cask Rental Cost (\$)</u>	<u>Transportation Cost (\$)</u>	<u>Disposal Cost (\$)</u>	<u>Total Cost (\$)</u>
320 000	1 386 000	552 200	631 190	2 889 390

For the purposes of this study, it is necessary to know the incremental cost of this alternate disposal method over and above the cost of shallow-land burial. From Table I.3-3, the total cost for disposal of the highly activated components in a shallow-land disposal facility is estimated to be \$2,041,030. Thus, the incremental cost for disposing these materials in deep geologic disposal is \$848,360 more than the shallow-land burial cost. No attempt is made to estimate the effects of this alternate disposal method on the other immediate dismantlement costs that it might affect.

If a special train is required, an additional total estimated cost of \$1,254,000 is necessary for 22 single-cask shipments (see Section M.4.2). The use of more than one cask per shipment would decrease this cost inversely to the number of casks per shipment, but would not affect the disposal costs presented previously.

I.3.9.4 Fuel Channel Disposal

The assumption that the fuel channels are shipped and stored with the spent fuel is based on practicality rather than present practice. Presently, this is not a viable option, and, therefore, an analysis is required to determine the alternative costs for disposing of the fuel channels as activated waste in either a shallow-land disposal facility or a deep geologic disposal facility.

It is recognized that the zircaloy-4 metal that is used for the fuel channels contains trace quantities of natural uranium, and that the irradiated fuel channels could possibly be classed as transuranic waste. However, for this analysis it is assumed that the transuranic radionuclide concentration is low enough to allow disposal of the irradiated fuel channels as non-transuranic waste similar to all the other neutron-activated materials.

No attempt is made in this analysis to determine the effect of this disposal operation on the task schedules presented in Section I.2 or the other costs presented in this section (Section I.3) that might be affected.

Shallow-Land Disposal. For shallow-land disposal, the 764 fuel channels are assumed to be cut lengthwise, packaged in specially fabricated steel boxes, and shipped in a truck-mounted, NAC-1 spent fuel cask. These packaging and shipping methods are the same as those assumed for the control rods in Section I.3.1.

Based on a 35% packing density, an estimated 31 specially fabricated steel boxes are required to package the fuel channel segments, at a total container cost of \$186,000 (see Table M.2-1 in Appendix M). For 31 overweight shipments, at 1600-km, 5-day round trips, the total cask rental cost (see Table M.3-1) is \$106,950, and the total transportation cost (see Table M.4-4) is \$59,520.

The handling surcharge is estimated to be \$47,700, based on a ratio-proportion calculation with the handling surcharge reported in Table I.3-3 for the control rods. The burial volume is estimated on this same basis to be about 58 m³, and the resultant burial cost is about \$5430 (see Table M.5-1).

For estimating the liner and curie surcharges, it is assumed that the surface dose rate and the radioactivity content of each fuel channel package are similar to those estimated for a control rod package, even though it is recognized that the radionuclide inventories in the contents of the two packages are quite different. Thus, the liner surcharge is estimated to be approximately \$165,890, and the curie surcharge is about \$45,650, again based on ratio-proportion calculations with the respective parameter values reported in Table I.3-3 for the control rods.

The individual estimated costs for shallow-land disposal of the fuel channels are summarized and totaled below:

<u>Container Cost (\$)</u>	<u>Cask Rental Cost (\$)</u>	<u>Transport Cost (\$)</u>	<u>Handling Surcharge (\$)</u>	<u>Burial Cost (\$)</u>	<u>Liner Surcharge (\$)</u>	<u>Curie Surcharge (\$)</u>	<u>Total Cost (\$)</u>
186 000	106 950	59 520	47 700	5 430	165 890	45 650	617 140

Deep Geologic Disposal. The fuel channels can also be classed as highly activated materials that may require placing in a deep geologic disposal facility. Therefore, with two exceptions, this analysis to estimate the alternate cost of deep geologic disposal of the fuel channels uses the same methods and assumptions as are used in Section I.3.9.3 in estimating this cost for the other highly activated materials. The packing density is assumed to be 25%, and the volume of the segmented fuel channels is estimated using data obtained from Reference 6.

The packageable volume of the segmented fuel channels is estimated to be 6.9 m³. Twenty-three (23) burial canisters are required, at a total cost of \$115,000. A minimum of 8 single-cask shipments are required, at a total cask rental cost of \$504,000 and a total transportation cost of \$208,000. The total cost for placing the 23 loaded canisters (the total volume is 32 m³) in deep geologic disposal is \$227,200.

The individual estimated costs for deep geologic disposal of the fuel channels are summarized and totaled below:

<u>Canister Cost (\$)</u>	<u>Cask Rental Cost (\$)</u>	<u>Transportation Cost (\$)</u>	<u>Disposal Cost (\$)</u>	<u>Total Cost (\$)</u>
115 000	504 000	208 000	227 200	1 047 000

The incremental cost for this disposal alternative is \$429,860 more than for shallow-land burial.

An additional cost of \$456,000 would be required for shipping the 8 single-cask shipments by special train. Again, the use of more than one cask per shipment would decrease this special train cost inversely to the number of casks per shipment, but would not affect the disposal costs presented previously.

I.4 EXTERNAL OCCUPATIONAL RADIATION DOSES FOR IMMEDIATE DISMANTLEMENT

Detailed estimates are made of the external occupational radiation doses that are accumulated by the decommissioning workers during immediate dismantlement of the reference BWR. The estimates are based on a task-by-task analysis to determine the man-hours of effort required in radiation zone work and the anticipated dose rates associated with each task for all labor categories. Basic assumptions made in developing these estimates are: 1) every effort is made to minimize the personnel exposure to radiation (ALARA philosophy) while accomplishing a task, by the use of temporary shielding and remote handling techniques and by 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 the decontaminated piping and equipment by at least a factor of 10; 3) careful, prompt accounting of radiation doses is maintained to rapidly identify jobs that are causing excessive dose accumulations so corrective action can be taken; and 4) ^{60}Co governs the radioactive decay.

The estimated total dose for each task is corrected for radioactive decay with a decay factor calculated using the half-life of ^{60}Co and the midpoint of the timeline for the given task (shown in Figure I.2-4).

The results of these analyses and corrections are presented in Table I.4-1. The total corrected external occupational radiation dose for immediate dismantlement of the reference BWR is 1845 man-rem. Of the total, 53% results from immediate dismantlement of the Reactor Building/Primary Containment, about 11% from immediate dismantlement of the Turbine Generator Building, nearly 32% from immediate dismantlement of the Radwaste and Control Building, and approximately 4% from ancillary tasks.

TABLE I.4-1. Estimated Occupational Radiation Doses for Immediate Dismantlement

Location		Supervisors (a)		Utility Operators and Laborers		Craftsmen		Health Physics Technicians		Task Totals			Corrected Dose (man-rem)(c)
		Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Decay Factor (b)	
Task													
Reactor Building/Primary Containment													
1	Install HEPA Filters	400	0 400	---	---	2 712	2 712	270	0 270	3 382	3 382	1 000	3 382
2	Comprehensive Radiation Survey	---	---	---	---	---	---	100	0 500	100	0 500	0 995	0 498
3	Discharge and Ship Fuel	941	8 469	6 268	56 412	---	---	2 752	24 768	9 961	89 649	0 904	81 043
4	Remove Dryer and Separator	130	0 650	865	4 325	545	2 725	87	0 435	1 627	8 135	0 989	8 046
5	Drain Suppression Pool to Radwaste, Water-Jet Clean(e)	27	0 189	322	1 932	27	0 135	27	0 135	403	2 391	0 957	2 288
6	Remove Reactor Vessel Internals	735	3 675	3 293	16 466	1 605	8 025	490	2 450	6 123	30 616	0 944	28 902
7	Ship Activated Reactor Vessel Internals Segments	1 865	9 325	12 432	62 160	---	---	1 250	6 250	15 547	77 735	0 921	71 594
8	Drain Reactor Well Pool to Radwaste, Water-Jet Clean(e)	3	0 015	55	0 275	5	0 025	5	0 025	68	0 340	0 906	0 308
9	Chemical Decon Reactor Water Recirculation and Cleanup Systems(e)	4	0 200	572	5 143	8	0 760	36	1 836	620	7 939	0 896	7 113
10	Clean up, Stage, and Shield Hot Spots in Primary Containment	108	1 080	720	7 200	---	---	72	0 720	900	9 000	0 906	8 154
11	Enlarge Suppression Chamber Access	63	0 315	420	2 100	---	---	42	0 210	525	2 625	0 898	2 357
12	Remove Reactor Vessel	300	1 500	1 888	9 440	960	4 800	190	0 950	3 338	16 690	0 874	14 587
13	Ship Activated Reactor Vessel Segments	520	2 600	3 456	17 280	---	---	350	1 750	4 326	21 630	0 835	18 061
14	Remove Primary Containment Piping and Equipment	1 040	20 800	6 905	138 100	2 872	57 440	690	13 800	11 507	230 140	0 829	193 087
15	Remove Sacrificial Shield and Radial Beams	770	10 010	5 080	35 560	2 770	49 860	510	6 630	9 130	102 060	0 781	79 709
16	Remove Contaminated Concrete from Primary Containment	490	2 450	3 272	17 305	---	---	330	1 650	4 092	21 405	0 768	16 439
17	Remove HVAC and Electrical Systems from Primary Containment	65	0 327	220	1 100	216	1 080	44	0 220	545	2 727	0 732	1 996
18	Drain Contaminated Systems to Radwaste(e)	---	---	12	0 180	4	0 060	2	0 030	18	0 270	0 933	0 252
19	Chemical Decon Residual Heat Removal, Low- and High-Pressure Core Spray Systems(e)	12	0 360	259	4 403	20	1 100	118	3 540	409	9 403	0 989	9 300
20	Remove Reactor Building Piping	1 370	27 340	9 150	183 000	3 786	75 720	915	18 300	15 221	304 470	0 896	272 805
21	Drain Dryer and Separator Pool to Radwaste, Water-Jet Clean(e)	9	0 045	97	0 485	8	0 040	8	0 040	122	0 610	0 848	0 517
22	Chemical Decon Drain Systems(e)	---	---	431	0 431	108	0 108	---	---	539	0 539	0 841	0 453
23	Drain Spent Fuel Pool to Radwaste, Water-Jet Clean(e)	9	0 045	109	0 545	9	0 045	9	0 045	136	0 680	0 803	0 546
24	Chemical Decon Fuel Pool Cooling and Cleanup System(e)	2	0 010	52	0 312	2	0 020	18	0 108	74	0 450	0 810	0 364
25	Remove Reactor Building Equipment	2 175	4 350	14 494	28 980	1 106	2 212	1 450	2 900	19 225	38 442	0 849	32 637
26	Remove Liners from Spent Fuel Pool, Reactor Well, and Dryer and Separator Pool	160	0 800	1 057	5 285	925	4 625	106	0 530	2 248	11 240	0 764	8 587
27	Remove Reactor Building Contaminated Concrete	1 000	3 000	6 764	20 797	---	---	680	2 040	8 444	25 837	0 746	19 274
28	Remove HVAC and Electrical Systems from Reactor Building	326	1 630	880	4 400	972	4 860	284	1 420	2 462	12 310	0 724	8 912
29	Final Radiation Survey	---	---	---	---	---	---	100	0 100	100	0 100	0 716	0 072
Subtotals, Reactor Building (c,f)		12 524	100	79 073	623	18 660	216	10 935	92	121 192	1 031		891
Turbine Generator Building													
1	Install HEPA Filters	1 100	1 100	---	---	7 450	7 450	745	0 745	9 295	9 295	1 000	9 295
2	Comprehensive Radiation Survey	---	---	---	---	---	---	100	0 200	100	0 200	0 995	0 199
3	Clean Up and Stage	100	0 100	700	0 700	---	---	70	0 070	870	0 870	0 984	0 856
4	Remove Turbine	407	0 407	2 711	5 422	1 242	2 484	272	0 544	4 632	8 857	0 952	8 432
5	Drain Contaminated Systems to Radwaste, Water-Jet Clean Condensate Storage Tanks(e)	8	0 008	51	0 051	---	---	5	0 005	64	0 064	0 921	0 059
6	Remove Condenser	1 651	1 792	11 011	21 566	7 755	4 538	1 100	2 154	21 517	40 050	0 867	34 723
7	Drain Condenser to Radwaste, Water-Jet Clean(e)	1	0 001	10	0 010	---	---	1	0 001	12	0 012	0 872	0 010
8	Chemical Decon Drain Systems(e)	---	---	305	0 305	76	0 076	---	---	381	0 381	0 855	0 326
9	Remove Piping	2 530	15 180	2 484	14 904	14 411	86 466	1 440	8 640	20 865	125 190	0 835	104 534
10	Remove Equipment	990	4 950	4 026	20 130	2 591	12 955	660	3 300	8 267	41 335	0 781	32 283
11	Remove Contaminated Concrete	300	0 300	2 039	2 039	---	---	204	0 204	2 543	2 543	0 762	1 938
12	Remove HVAC and Electrical Systems	53	0 053	315	0 315	167	0 167	36	0 036	571	0 571	0 744	0 425
13	Final Radiation Survey	---	---	---	---	---	---	50	0 050	50	0 050	0 746	0 037
Subtotals, Turbine Generator Building (c,f)		7 140	24	23 652	65	33 437	124	4 683	16	69 167	229		193
Radwaste and Control Building													
1	Comprehensive Radiation Survey	---	---	---	---	---	---	80	0 800	80	0 800	0 995	0 796
2	General Cleanup	15	0 075	180	2 700	---	---	15	0 075	210	2 850	0 828	2 360
3	Chemical Decon Drain Systems (e)	---	---	435	0 435	109	0 109	---	---	544	0 544	0 810	0 441
4	Chemical Decon Equipment, Water-Jet Clean(e)	45	0 195	360	21 505	---	---	45	0 915	450	22 615	0 797	18 024
5	Remove Piping	2 707	13 001	9 845	115 696	19 688	163 848	7 382	38 239	39 622	330 774	0 752	248 742
6	Install Temporary Radwaste System	2	0 004	16	0 032	24	0 048	2	0 004	44	0 088	0 786	0 069
7	Remove Equipment	3 048	7 621	6 508	171 001	4 976	124 638	4 035	52 138	18 567	355 398	0 685	243 448
8	Remove Contaminated Concrete	144	0 402	2 640	8 640	690	1 950	546	1 134	4 020	12 126	0 659	7 991
9	Remove Miscellaneous Steel Structures	16	0 032	312	0 624	312	0 624	39	0 078	679	1 358	0 648	0 880
10	Remove HVAC and Electrical Systems	361	0 900	888	4 200	1 012	4 460	428	1 069	2 689	10 629	0 642	6 824
11	Final Radiation Survey	---	---	---	---	---	---	80	0 080	80	0 080	0 635	0 051
Subtotals, Radwaste & Control Bldg (c,f)		6 338	22	21 184	325	26 811	296	12 652	95	66 985	737		530
Ancillaries													
1	Operate Radwaste Systems (e)	---	---	286	1 430	144	21 600	36	0 495	466	23 525	0 886	20 843
2	Routine Radiation Surveys	---	---	---	---	---	---	1 950	7 950	1 950	7 950	0 799	6 352
3	Package Dry Solid Wastes	---	---	2 134	46 962	66	1 600	---	---	2 200	48 562	0 799	38 801
4	Miscellaneous(g)	---	---	---	---	---	---	---	---	---	---	---	165
Subtotals, Ancillaries (c,f)		---	---	2 420	48	210	23	1 986	8	4 616	80		231
Totals (c)		26 002	146	126 329	1 061	79 118	695	30 256	211	261 960	2 077		1 845

(a)Includes shift engineers, crew leaders, craft supervisors, and senior health physics technicians
(b)Based on the half-life of ⁶⁰Co, calculated at the midpoint of the task timelines shown in Figure 1 2-4
(c)The number of significant figures shown is for computational convenience and does not imply accuracy to the nearest millirem
(d)Indicates that category of decommissioning staff is not involved in the particular task
(e)Based on data presented in Table H 5-11 in Appendix H
(f)Dose totals are rounded to the nearest whole number
(g)Consists of an allowance of up to 1 rem per year for selected management and support staff whose radiation doses are not specifically estimated, together with an allowance of 5% of the total explicitly estimated task radiation dose, to account for any omissions and uncertainties in the analyses

Two other tasks not shown in Table I.4-1 provide the potential for external radiation dose and occur as a direct result of immediate dismantlement: 1) transporting the radioactive materials to the disposal site, and 2) handling these materials at the disposal site. A total of 1,495 shipments of radioactive materials must be transported to, and handled at, the disposal site. The external radiation dose from transportation is discussed in Section N.5 of Appendix N. The external radiation dose from handling these materials at the disposal site is beyond the scope of this study since it is considered under another nuclear license.

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* Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and the National Technical Information Service, Springfield, Virginia 22161.

** Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.

APPENDIX J

PASSIVE SAFE STORAGE, CONTINUING CARE, AND DEFERRED DISMANTLEMENT DETAILS

This appendix contains the details of passive safe storage for the reference BWR. Besides the preparations made during active decommissioning following reactor shutdown, passive safe storage includes continuing care to meet the remaining requirements of the nuclear license and probable deferred dismantlement to terminate that license.

The information presented in this appendix includes the following:

- selection of passive safe storage
- passive safe storage supporting information
- preparations for passive safe storage
- passive safe storage schedules and manpower estimates
- costs for passive safe storage
- estimated external occupational radiation doses for passive safe storage
- deferred dismantlement

J.1 SELECTION OF PASSIVE SAFE STORAGE

All categories of safe storage--custodial, passive, and hardened--are open-ended, in that all require positive action at the conclusion of the period of continuing care to release the property for unrestricted use and to terminate the possession-only license.

In hardened safe storage, temporary, hardened barriers around areas containing significant quantities of residual radioactivity must be installed and may eventually be removed; thus, this category requires maximum costs for preparations and possible deferred dismantlement but minimum costs for continuing care. Custodial safe storage requires minimal preparations but requires full-time, onsite, surveillance crews during continuing care to maintain the structure, the operating equipment, and the security of the property. Passive safe storage is assumed to optimize the costs for preparations, continuing care, and possible deferred dismantlement, keeping

the combined cost at a minimum. Thus, passive safe storage is selected in this study as the safe storage category for decommissioning the reference BWR.

The planning and preparation activities for the preparations for passive safe storage are described generically in Section H.2 of Appendix H. Work plans are developed for decommissioning the Reactor Building (including the Primary Containment), the Turbine Generator Building, and the Radwaste and Control Building. The procedures discussed in this appendix are postulated to be used to decontaminate, deactivate, and secure radioactive materials and contaminated areas within these buildings, to achieve a condition that provides protection to the public and the environment, with limited maintenance and surveillance.

Areas of the facility that must be accessible during continuing care are decontaminated to unrestricted use levels. Noncombustible, contaminated materials, which are not removed from the facility, may be placed in the drained spent fuel pool, dryer separator pool, and/or pressure suppression pool. These areas are isolated from the remainder of the facility by structurally substantial physical barriers. The remaining portion of the site may be released for unrestricted use during the continuing care period.

The continuing care period begins when the preparations for passive safe storage are completed. This period is characterized by surveillance and maintenance activities designed to ensure that the facility remains in a condition that poses minimum risks to the public. The activities included in this phase are limited to environmental and facility radiation monitoring and to inspecting and repairing, as necessary, the physical barriers, the structures and the instrumentation. Security is provided by the fence around the immediate facility site (about 0.06 square kilometers), by high-security locks on entrance doors, and by electronic alarms.

It is assumed that approximately 18 months before final plant shut-down are devoted to planning and preparation. Based on detailed work plans, about 30 months (from the time of initiation of physical decommissioning tasks) are estimated to be required to prepare the facility for the period of continuing

care. It is assumed that the facility will remain in the passive safe storage condition until deferred dismantlement takes place.

J.2 PASSIVE SAFE STORAGE SUPPORTING INFORMATION

Decommissioning methods, special tools and equipment, and essential systems and services used in preparing the reference BWR for passive safe storage are discussed in this section. These methods and considerations, in order of their presentation in the following subsections, are:

- decontamination, deactivation, and sealing methods
- spray painting
- transfer of contaminated equipment and materials
- decontamination and isolation procedure
- special tools and equipment
- essential systems and services

J.2.1 Decontamination, Deactivation, and Sealing Methods

Initially, the comprehensive decommissioning radiation survey is completed. Systems draining and decontamination tasks described in Section H.5 of Appendix H begin immediately after the fuel is discharged from the reactor. Selected decontamination tasks may proceed concurrently. The primary concern is to ensure that no recontamination of clean areas occurs and that air leaving a given area flows through a filter system or, in the case of liquid effluents, through the existing contaminated waste systems.

The particular method used to decontaminate, deactivate, and seal each system or piece of equipment is identified during the planning phase. In general, all systems not necessary to prevent the spread of contamination are deactivated. Equipment deactivation, isolation of contaminated areas, and sealing techniques are described in Section H.6 of Appendix H. Generic decontamination methods used in the preparations for passive safe storage are described in Appendix G.

The contaminated drains are decontaminated as described in Section H.5 of Appendix H. Building sumps are decontaminated and secured. In some cases,

after the sump pumps are decontaminated and/or removed; steel plates are welded in place to cover the sump area.

Spray painting of contaminated materials and equipment and transferring the contaminated materials and equipment are assumed to be widely used in the preparations for passive safe storage. These techniques are described in subsequent subsections.

J.2.2. Spray Painting

After the loose, readily removable contamination is removed by the physical cleaning methods described in Section G.1 of Appendix G, the rooms or areas and their associated equipment are thoroughly spray painted before isolation or removal procedures begin. When possible, all contaminated surfaces, both inside and outside, are coated to prevent the entrainment of radioactivity in the air during the active decommissioning tasks or during subsequent surveillance and maintenance activities.

In general, if the contamination on a surface cannot be removed by wiping or washing using standard decontamination solutions, it is painted to fix the contamination in place. An example is a concrete surface that has been penetrated by contaminated liquids. While the surface might be cleaned initially, the subsurface contamination can migrate to the surface and be dispersed by air movement and/or foot traffic. On protected, interior surfaces with essentially no traffic or adverse environment, the paint coatings can be expected to last almost indefinitely. Part of the surveillance program is to monitor painted areas for deterioration of the coatings and to recoat them as necessary.

J.2.3 Transfer of Contaminated Equipment and Materials

Unsalvageable, contaminated equipment and other miscellaneous noncombustible items may be transferred to other secured, onsite retrievable storage areas, as described in Section J.1. It is anticipated that before transferring small equipment items, the items are carefully spray painted to immobilize any contamination. Freshly exposed surfaces are immediately painted to prevent dispersal of contamination. The disconnected items are carefully bagged and transferred to a retrievable storage area. The equipment and ductwork remaining

in the work area is physically decontaminated as described in Section G.1 of Appendix G and spray painted as previously described.

J.2.4 Decontamination and Isolation Procedure

The 13-point procedure given below is postulated to be used to prepare the contaminated areas throughout the Reactor Building, the Turbine Generator Building, and the Radwaste and Control Building for passive safe storage:

1. Conduct initial radiation survey.
2. Vacuum interior surface areas.
3. Deactivate nonessential systems and equipment.
4. Clean interior surface areas and exposed surfaces of equipment and piping.
5. Clean remaining hot spots.
6. Apply protective paint.
7. Transfer, as feasible, contaminated equipment and materials.
8. Decontaminate and seal vent systems.
9. Install HEPA-filtered vents.
10. Deactivate remaining nonessential systems and equipment.
11. Install intrusion alarms; provide for offsite readout for intrusion, fire, and radiation detection systems.
12. Conduct final radiation survey.
13. Secure the structure.

J.2.5 Special Tools and Equipment

Fewer special tools and equipment are required for the preparations for passive safe storage than for immediate dismantlement. No dismantlement of highly activated material or equipment is necessary, thus eliminating the need for and expense of special remote handling equipment.

A list of special tools and equipment postulated for use in preparations for passive safe storage, together with their functions, is given in Table J.2-1.

TABLE J.2-1. Special Tools and Equipment for Preparations for Passive Safe Storage

Item	Estimated Number Required (a)	Function
Portable Oxyacetylene Torch	2	Cutting carbon steel equipment and piping, welding carbon steel
Guillotine Pipe Saw	2	Cutting piping
Power-Operated Reciprocating Hacksaw	2	Cutting piping
Closed-Circuit, High-Resolution Television	1	Remote observation of tasks
Submersible Pump with Disposable Filter	5	Draining of suppression chamber and spent fuel storage pools
High-Pressure Water Jet	{ 2	Surface decontamination of suppression chamber, tanks, main condenser, pool liners
Low-Pressure Water Jet		
Pipe Jumper	4	Chemical decontamination of reactor piping systems
Mobile Chemical Decontamination Unit	5	Chemical decontamination of liquid and solid radwaste equipment
Mobile Chemical Mixing and Heating Unit	4	Chemical decontamination of drain systems
Scaffolding	50	Safe access to heights
Safety Nets	As Required	Protect personnel working on elevated equipment and structures
Power-Operated, Mobile, Scissors-Type Manlift	2	Safe access to heights
Power-Operated, Mobile, Articulated-Arm Manlift	1	Safe access to heights
9100-kg Capacity Forklift	1	Packaging of contaminated materials, and loading of trucks
Rigging Materials (e.g., chockers, grapples, winches)	As Required	Handling of equipment and piping
Concrete Drill with HEPA-filtered Dust Collecting System	1	Drilling holes in concrete as required for surface spalling
Concrete Surface Spaller	1	Removal of contaminated concrete surfaces
Front-End Loader (highly maneuverable, light-duty)	1	Cleanup and packaging tasks
Vacuum Cleaner (HEPA-filtered)	3	Cleanup tasks
Portable Ventilation Enclosure	3	Contamination control during cutting of contaminated material
Supplied-Air Plastic Suit	100	Provide personnel with maximum respiratory and surface protection from radioactive contaminants
Polyurethane Foam Generator	2	Contamination control during HVAC isolation tasks
Paint Sprayer	4	Immobilization of contamination

(a)Based on a taskwise analysis of the schedule, plus spares, as required

J.2.6 Essential Systems and Services

During preparations for passive safe storage, certain facility systems and services must remain in place and in service for contamination control, for industrial or personnel safety, and to aid in the completion of decommissioning tasks. These systems are the same as those described in Section H.4 of Appendix H for immediate dismantlement and are not repeated here.

As areas within the facility are secured for continuing care, the essential systems and services in these areas are deactivated as described previously in Section J.2.1. Continuous service to the remaining work areas is maintained as required.

After placing the facility in passive safe storage, certain systems and services are required during the continuing care period. The systems and services postulated to be required during continuing care are listed in Table J.2-2, together with the justification for retaining each. The equipment in these systems is inspected and renovated to ensure adequate reliability before the surveillance and maintenance period begins. In addition, the intrusion alarm systems within the facility and on the perimeter fence are modified to provide surveillance capability by a commercial security agency.

TABLE J.2-2. Systems and Services Required During the Continuing Care Period

<u>Systems and Services</u>	<u>Justification</u>
Electrical Power	Operation of electrical equipment, including lighting, surveillance monitoring, and radiation monitoring systems and alarms
Fire Protection System	Health and safety
In-Plant Communications System (telephone)	Personnel safety considerations
Security Systems	Public safety and plant protection considerations

J.3 PREPARATIONS FOR PASSIVE SAFE STORAGE

After reactor defueling operations, draining of reactor pools and piping systems that contain radioactively contaminated water begins. This contaminated water inventory is scheduled for processing through the radwaste systems as described in Section H.5 of Appendix H. Chemical decontamination of selected systems, also described in Section H.5, is accomplished before these systems are isolated and deactivated. The chemical decontamination solutions also are processed through the existing radwaste systems.

The Reactor Building, the Turbine Generator Building, and the Radwaste and Control Building are given a comprehensive radiation survey to accurately determine radiation and contamination levels before they are prepared for continuing care. The site and support facilities are given a comprehensive radiation survey and, for purposes of this study, are assumed to be released for unrestricted use without further effort. A discussion of the postulated preparations for passive safe storage of the reference BWR is presented in the following subsections.

J.3.1 Preparation of the Reactor Building for Passive Safe Storage

Large quantities of neutron-activated materials and many of the radioactively contaminated areas are located in the Primary Containment Vessel (PCV). For this reason, several task steps are taken in this section that are designed to improve the isolation of this significant substructure within the Reactor Building. These steps involve sealing all major accessways into the PCV, as well as providing barriers around high-radiation-level piping areas just outside of the PCV.

At reactor shutdown, the operations proceed in the usual manner for discharging the existing fuel load. The drywell and reactor vessel heads are removed, followed by the steam dryer and steam separator. The fuel is stored and cooled as necessary and then transferred to fuel shipment casks (GE IF-300 or equivalent) for shipment to a repository. The fuel shipment operations maintain an inventory of the fuel material for accountability purposes. Rail shipments convey the nuclear fuel to its appropriate repository. Components of the reactor vessel and PCV removed during refueling are returned to their normal positions and secured as normally provided for.

Fluids from the suppression pool and reactor well pool cavities are sequenced into the radwaste system, followed by chemical decontamination of the reactor water recirculating systems and of piping and equipment intertied with other buildings undergoing preparation for passive safe storage. Valves within the PCV are closed at this time in preparation for the structure's eventual isolation. The spent fuel pool and dryer separator pool are drained, cleaned, covered, and sealed in sequence. The three major equipment and personnel hatches are sealed as described in Appendix K.

All equipment, gratings, ladders, stairs, HVAC, electrical, concrete, and other miscellaneous items requiring decontamination are cleaned. Long-lasting protective paint coatings are applied to the cleaned areas to immobilize any residual radioactivity.

The various areas of the building are isolated and sealed, where appropriate, by closing all piping valves, sealing interconnecting HVAC air passageways, and sealing rooms containing significant quantities of radiation. Both the steam tunnel and the rail tunnel are sealed and sectioned off.

A permanent seal is applied to the drywell top head of the Reactor Vessel/PCV as a final act of isolation of this structure. Nonessential utilities are deactivated, including fans, electrical units, motors, cranes, and pumps.

Waste materials from the preparation for passive safe storage are packaged and shipped to the appropriate disposal sites.

J.3.2 Preparation of the Turbine Generator Building for Passive Safe Storage

The tasks associated with this mode of decommissioning for the Turbine Generator Building are rather straight-forward because of the low level of radioactive contamination expected to be present. Following a general cleanup, the contaminated systems are drained and the condensate storage tanks are cleaned. After the condenser hot well is drained, areas accessible to personnel, which represent about 25% of the total condenser surfaces, are cleaned using the water-jet technique.

Zones of contamination found on concrete surfaces and certain metal structures are water-jet cleaned where possible. Other areas, including the HVAC system, electrical and mechanical equipment, and other poorly accessible surfaces, are manually decontaminated. Protective coats of paint are applied to the cleaned surfaces by spray painting or by manual application where necessary. Tasks associated with isolation and sealing of equipment, piping, rooms, and HVAC ducts are handled similar to those described for the Reactor Building/PCV.

J.3.3 Radwaste and Control Building

Much of the contaminated water in the Reactor Building and the Turbine Generator Building is processed in the liquid and solid radwaste systems

located in the Radwaste and Control Building. Once decommissioning of the radwaste systems is started, a temporary, portable radwaste system, provided by a specialty contractor, is used for final radwaste processing.

The decommissioning techniques postulated for use in the Radwaste and Control Building are the same as those described for the Reactor Building and the Turbine Generator Building (see Section J.2 for generic methods).

J.3.4 Site and Support Facilities

The final active decommissioning task is the comprehensive radiation survey of the reference BWR site and support facilities. The existing security fence that surrounds the entire site (see Figure C.1-1 in Appendix C) is left intact and is assumed to be maintained during the continuing care period. A well-labeled security fence, equipped with intrusion detection and alarm provisions, is constructed within the existing fence to form an easily monitored exclusion area around the Reactor Building, the Turbine Generator Building, and the Radwaste and Control Building.

Electricity requirements at the facility are nominal after the preparations for passive safe storage are completed. It is assumed that the plant's large electrical service is changed to a small electrical service near the end of the active decommissioning period to provide for the needs of the essential systems and services described in Section J.2.6.

J.4 PASSIVE SAFE STORAGE SCHEDULES AND MANPOWER ESTIMATES

Developing a reasonably optimum schedule and manpower requirements for preparations for passive safe storage requires several iterations. Initially, the sequence in which the various systems must be decontaminated, drained, and secured is determined. Similarly, the sequence for decontamination, contamination immobilization, and isolation of the various rooms and areas within the Reactor Building, the Turbine Generator Building, and the Radwaste and Control Building is determined. Then, the number and types of decommissioning workers required to accomplish each task in the allotted time is estimated. The organization of the decommissioning staff is the same as that for immediate dismantlement, as shown in Figure H.1-1 in Appendix H. Based on the radiation dose estimated to

accomplish each task, the number of workers needed to complete the radiation zone work in the allotted time and within the assumed radiation dose limits is determined. Whole-body radiation doses to the decommissioning workers are limited in accordance with 10 CFR 20.101. The supervisors, utility operators, and health physics technicians are assumed to be long-time radiation workers whose annual exposure is limited to 5 rem/yr by the formula $5(N-18)$ of 10 CFR 20.101(b)(2). The craftsmen and laborers are assumed to have had little prior radiation exposure and, therefore, under 10 CFR 20.101(b)(1) and (2) may receive up to 3 rem per quarter, within the limitation of the formula $5(N-18)$. In those instances where the manpower estimated for physically accomplishing a task results in an average dose for each person in excess of the limits above, additional manpower is assigned to the tasks to keep the occupational dose below these limits. In the manpower table presented in this section, the manpower shown is adequate both to accomplish the task and to meet the occupational dose limits.

The decommissioning worker requirements given in this section include neither the management and support staff (see Figure H.1-1 in Appendix H) nor the specialty contractor personnel (see Section H.3 of Appendix H).

All decommissioning tasks are conducted in accordance with the decommissioning quality assurance plan (see Section G.5 of Appendix G) and are checked by the utility quality assurance staff. Environmental monitoring (see Section G.6 of Appendix G) during preparations for passive safe storage is performed by a specialty contractor.

J.4.1 Reactor Building/Primary Containment

A considerable effort is necessary in the preparation for passive safe storage of this structure. Special care is taken to isolate the large amounts of radionuclides from unauthorized public access and eliminate the need for authorized inspections. The manpower required to accomplish this task is estimated to exceed 20,000 man-hours. Based on the task description discussed in Section H.3.1 and the occupational dose estimates discussed in Section J.6, the schedule and decommissioning worker requirements for passive safe storage of this building are presented in Figure J.4-1. The preparations for passive

FIGURE J.4-1. Task Schedule and Sequence of the Preparations for Passive Safe Storage

safe storage are estimated to be completed about 30 months after final reactor shutdown.

J.4.2 Turbine Generator Building

The Turbine Generator Building is a rather large structure. However, the manpower required to place it in the passive safe storage mode is not proportional to its size. It is expected that considerably smaller quantities of radioactive deposition and contamination will exist in this building at reactor shutdown than in the other structures treated in this study. Much of the time necessary to decommission this building is taken up by fluid processing in the radwaste facility rather than by actual decommissioning staff activities. Except for the fluid processing, the low dose rate activities associated with this building are readily coordinated within the time frame necessary for the other two major buildings.

The schedule and decommissioning worker requirements for passive safe storage of this building are presented in Figure J.4-1. The passive safe storage tasks are completed about 25 months following reactor shutdown.

J.4.3 Radwaste and Control Building

The liquid and solid radwaste systems located in the Radwaste and Control Building are needed to process the contaminated liquids present in the systems at reactor shutdown and the contaminated liquid wastes generated during decommissioning. Consequently, the Radwaste and Control Building is the last contaminated structure prepared for passive safe storage. A temporary, portable radwaste system is installed to process liquid wastes during decommissioning of this building. Preparations for passive safe storage of the Radwaste and Control Building are started in the 14th month after final reactor shutdown and continue through the 30th month.

J.4.4 Site and Support Facilities

The structures and site surfaces outside of the exclusion area (shown in Figure 9.2-1 in Chapter 9) but within the site perimeter fence are surveyed and assumed to be releasable for unrestricted use without further effort. However, it is recognized that extensive radiation surveys and analyses of

soil surfaces and paved areas are required to demonstrate that the site is releasable. Determination of the level of effort actually necessary for cleanup of the site is beyond the scope of this study because of the wide range of possible conditions. However, for this study, these activities are assumed to be accomplished starting in the 29th month following final reactor shutdown and are estimated to require 8 man-months of decommissioning worker effort.

J.4.5 Summary of Schedule and Manpower Estimates for Preparations for Passive Safe Storage

The overall task schedule and sequence for preparing the reference BWR for passive safe storage is shown in Figure J.4-1, together with the total decommissioning worker requirements for the period following final reactor shutdown. The planning and preparation tasks (see Section H.2 of Appendix H and Section J.1.1) for the 18-month period preceding reactor shutdown are also included. The preparations for passive safe storage of the Reactor Building/Primary Containment, the Turbine Generator Building, and the Radwaste and Control Building are estimated to be accomplished during the 30-month period following final reactor shutdown. This time frame includes a 2-month radiation survey of the site and support facilities starting in the 29th month after reactor shutdown (see Section J.4.4).

Decommissioning tasks, with a few exceptions, are accomplished on two shifts, 5 days per week. The discharge and shipment of spent fuel and the operating of the radwaste systems are conducted on three shifts per day, 7 days per week, as required. The total decommissioning worker requirements for the period following reactor shutdown are shown in Figure J.4-1. A total of 1676.3 man-months of "hands-on" effort is estimated to be required.

The total staff labor requirements for preparing the reference BWR for passive safe storage are given in Table J.4-1. The requirements are given in equivalent man-years for the 1.5 years before and the 2.5 years following final reactor shutdown, and include the management and support staff as well as the decommissioning workers. A total of 383.4 man-years is estimated to be required to place the reference BWR in passive safe storage, including 176.9 man-years for the management and support staff and 206.5 man-years for

TABLE J.4-1. Staff Labor Requirements for Preparations for Passive Safe Storage

Position	Time Relative to Final Reactor Shutdown (year)					Total Staff Labor Required (man-years)
	-2	-1	1	2	3	
	Annual Staff Labor Requirement (man-years)(a)					
Management and Support Staff						
Decommissioning Superintendent	0.5	1.0	1.0	1.0	0.8 ^(b)	4.3
Secretary	0.5	2.0	3.0	3.0	1.6 ^(b)	10.1
Clerk	0	1.0	2.0	2.0	0.5	5.5
Decommissioning Engineer	0.5	1.0	1.0	1.0	0.8 ^(b)	4.3
Assistant Decommissioning Engineer	0.5	1.0	1.0	1.0	0.4	3.9
Radioactive Shipment Specialist	0	1.0	1.0	1.0	0.4	3.4
Procurement Specialist	0.2	1.0	1.0	1.0	0.4	3.6
Tool Crib Attendant	0	0	2.0	2.0	0.8	4.8
Control Room Operator ^(c)	0	0	5.0	5.0	2.1	12.1
Security Supervisor	0	0	1.0	1.0	0.5	2.5
Security Shift Supervisor ^(d)	0	0	5.0	5.0	2.5	12.5
Security Patrolman ^(d)	0	0	39.0	28.0	6.5 ^(b)	73.5
Contracts and Accounting Specialist	0.2	1.0	1.0	1.0	0.8 ^(b)	4.0
Health and Safety Supervisor	0.5	1.0	1.0	1.0	0.8 ^(b)	4.3
Health Physicist	0	0.5	1.0	1.0	0.5	3.0
Protective Equipment Attendant	0	0	2.0	2.0	1.0	5.0
Industrial Safety Specialist	0.2	1.0	1.0	1.0	0.5	3.7
Quality Assurance Supervisor	0.2	1.0	1.0	1.0	0.8 ^(b)	4.0
Quality Assurance Engineer	0.3	2.0	1.0	1.0	0.5	4.8
Quality Assurance Technician	0	0.5	2.0	2.0	1.0	5.5
Consultant (Safety Review)	0.3	0.5	0.5	0.5	0.3	2.1
Subtotals, Management and Support Staff	3.9	15.5	72.5	61.5	23.5	176.9
Decommissioning Workers^(e)						
Shift Engineer	0.5	2.0	2.0	2.0	1.0	7.5
Crew Leader ^(f)	0	1.0	15.0	13.3	0.9	30.2
Utility Operator ^(g)	0	3.0	45.0	32.1	1.0	81.1
Laborer	0	0	0.6	3.0	8.0	11.6
Craft Supervisor	0	0.6	1.0	2.0	0.9	4.5
Craftsman	0	10.0	5.0	8.0	14.0	37.0
Senior Health Physics Technician	0	1.0	2.2	2.2	1.4	6.8
Health Physics Technician ^(h)	0	3.0	13.0	9.1	2.7	27.8
Subtotals, Decommissioning Workers	0.5	20.6	83.8	71.7	29.9	206.5
Totals	4.4	36.1	156.3	133.2	53.4	383.4

(a) Rounded to the next higher 0.1 man-year.

(b) Includes an additional 4 months following active decommissioning in order to complete the documentation and other unspecified license and contract termination requirements.

(c) Based on one operator per shift in the control room, three shifts per day, 7 days per week.

(d) Based on 10 CFR Part 73 and information obtained from Washington Public Power Supply System; includes both response and access control personnel on a three-shift, 7-day-week basis.

(e) Requirements during the 3 years following final reactor shutdown are based on Figure J.4-1, unless otherwise noted; individual man-month requirements in Figure J.4-1 are rounded to the next higher 1.0 man-month in calculating these requirements.

(f) Based on a constant loading of 15 people through month 22 following reactor shutdown, with additional personnel added as required to meet schedule demands during that period, and diminishing thereafter as the schedule allows.

(g) Based on constant loadings of 45 people through month 12 following final reactor shutdown and 35 people from month 13 through month 23, and diminishing thereafter as the schedule allows.

(h) Based on a constant loading of 13 people through month 18 following reactor shutdown, with additional personnel added as required to meet schedule demands during that period, and diminishing thereafter as the schedule allows except during months 29 and 30 when 10 trained people are required to meet the schedule requirements.

the decommissioning workers. The assumptions used in developing this estimate are the same as those used for immediate dismantlement (see Section I.2-4 of Appendix I).

The staff labor requirements shown in Table J.4-1 for the decommissioning workers during the 2.5 years following final reactor shutdown exceed the total manpower presented in Figure J.4-1 because of the aforementioned assumptions. This "excess" manpower is assumed to be utilized for the numerous, small, unspecified tasks that accompany an activity of this magnitude.

J.4.6 Manpower Estimates for the Continuing Care Period

The staff organization shown in Figure J.4-2 is postulated to take control of all surveillance, maintenance, and security activities for the duration of the continuing care period. The estimated manpower requirements for continuing care of the reference BWR are summarized in Table J.4-2. A total of about 1.3 man-years per year of continuing care is estimated to adequately and safely satisfy anticipated passive safe storage requirements.

FIGURE J.4-2. Staff Organization for the Continuing Care Period

TABLE J.4-2. Estimated Annual Manpower Requirements for the Continuing Care Period

<u>Labor</u>	<u>Estimated Man-Years/Year Labor Requirement</u>
Surveillance and Maintenance Representative	0.25
Secretary	0.25
Repairman	0.1
Security	0.3
Third-Party Inspection	0.06
Environmental and Radiological Monitoring Program Personnel	0.35
<u>Quality Assurance Specialist</u>	<u>0.03</u>
Total	1.34

J.5 COSTS FOR PASSIVE SAFE STORAGE

The estimated costs for preparing the reference BWR for passive safe storage and for continuing care are presented in this section; costs for possible deferred dismantlement are presented later in Section J.7. These costs are all adjusted to early 1978 conditions.

J.5.1 Costs for Preparations for Passive Safe Storage

The estimated costs for preparations for passive safe storage are summarized and totaled in Table J.5-1. The total cost for preparations for passive safe storage is estimated at approximately \$21.3 million. The costs for disposal of radioactive materials represent about 7% of this total. The major contributors to the total cost are staff labor and energy (at approximately 66% and 13%, respectively). The combined costs for special tools and equipment and for miscellaneous supplies make up about 10% of the total cost. The remaining 4% of the total cost is contributed by specialty contractors, nuclear insurance, and license fees.

The total cost for other possible requirements (i.e., spent fuel shipment and fuel channel disposal) during preparations for passive safe storage is

estimated at about \$5.5 million, including a 25% contingency. This cost could go as high as \$9.5 million, depending on the possible additional requirements listed in the footnotes to Table J.5-1.

TABLE J.5-1. Summary of the Estimated Costs for Preparations for Passive Safe Storage

<u>Cost Category</u>	<u>Estimated Costs (\$ millions)(a,b)</u>	<u>Percent of Total(c)</u>
Disposal of Radioactive Materials (Radioactive Wastes)(d)	1.216	7.1
Staff Labor	11.254	66.1
Energy	2.122	12.5
Special Tools and Equipment	0.351	2.1
Miscellaneous Supplies	1.361	8.0
Specialty Contractors(e)	0.196	1.2
Nuclear Insurance	0.500	2.9
License Fees	0.038	0.2
Subtotal	17.038	100.0
Contingency (25%)	4.260	
Total, Preparations for Passive Safe Storage Costs	21.298	
Other Possible Costs		
Spent Fuel Shipment	3.788(f)	
Fuel Channel Disposal	0.617(g)	
Subtotal	4.405	
Contingency (25%)	1.101	
Total, Other Possible Costs	5.506	

(a) Costs adjusted to early 1978.

(b) The number of significant figures shown is for computational completeness and does not imply accuracy to the nearest \$1000.

(c) Individually rounded to the nearest 0.1%.

(d) Includes both wet and dry solid wastes.

(e) Includes temporary radwaste, environmental monitoring services, and security preparations.

(f) If required, shipment by special train would cost an additional estimated \$2.451 million, maximum.

(g) Deep geologic disposal costs (\$1.047 million) and use of special trains (\$0.456 million) could add \$1.5 million to this cost.

J.5.1.1 Costs for Disposal of Radioactive Materials

Only radioactive wastes (wet solid wastes and dry solid wastes) in the reference BWR require disposal during preparations for passive safe storage. The total cost for disposal of these materials is about \$1.2 million and is approximately 7% of the total cost of preparations for passive safe storage. The disposal cost includes the container, transportation, and burial costs, but does not include the direct labor costs for removing and packaging these materials. Labor costs are discussed later (Section J.5.1.2). Table J.5-2 contains a summary and a total of the number of shipments, the burial volumes, and the disposal costs for the radioactive waste materials.

TABLE J.5-2. Summary of Shipments, Volumes, and Costs for Disposal of Radioactive Materials During Preparations for Passive Safe Storage

<u>Radioactive Material</u>	<u>Number of Shipments</u>	<u>Burial Volume (m³)(a)</u>	<u>Disposal Costs (\$ millions)(b)</u>
Radioactive Wastes			
Wet Solid Wastes	286	814	1.073
<u>Dry Solid Wastes</u>	<u>32</u>	<u>241</u>	<u>0.143</u>
Totals	318	1 055	1.216

(a)Includes disposable containers.

(b)Adjusted to early 1978 conditions; the number of figures shown is for computational completeness and does not imply accuracy to the nearest \$1000.

Radioactive wastes (radwastes) result directly from preparing the reference BWR for passive safe storage. Two categories of radwastes are defined: wet solid wastes and dry solid wastes.

Wet solid wastes result from the processing of chemical decontamination solutions and contaminated water volumes and include concentrator bottoms, filter sludges, and spent demineralizer resins, as well as neutralized chemical solutions from decontamination of the contaminated drain systems. These wastes are assumed mixed with a cement solidifying agent and encapsulated in a steel cask liner prior to shipment to a shallow-land burial facility. The disposal cost data for

wet solid wastes generated during preparations for passive safe storage are presented in Table H.5-9 in Appendix H and are not repeated in detail here. An estimated 286 truck shipments, 814 m³ of burial space, and \$1,073,400 are required to dispose of the wet solid wastes.

Dry solid wastes include discarded contaminated materials such as plastic sheeting, rags, and anticontamination clothing. They are expected to occur as a result of most of the tasks specified in Section J.4 and are estimated on a taskwise basis. The dry solid wastes are compacted as much as possible to reduce their volume prior to shipment to a disposal facility.

Dry solid wastes in uncompacted volumes of 365 m³ from the Reactor Building, 478 m³ from the Turbine Generator Building, and 351 m³ from the Radwaste and Control Building are estimated to be collected during the preparations for passive safe storage. These volumes are assumed to be compacted five-fold and packaged in 348, 456, and 335 standard 0.21-m³ steel drums, respectively. Of the total number of drums, 352 are assumed to require shielding during shipment, with the remainder shipped unshielded in closed trucks. The two types are shipped separately, since, by virtue of the ⁶⁰Co content, those requiring shielding also require Type B packaging. Table J.5-3 gives a breakdown of the disposal costs for the dry solid wastes. An estimated 32 overweight truck shipments (26 shielded, 6 unshielded) are required to transport the compacted, packaged, dry solid wastes to a shallow-land burial facility, where they occupy an estimated 241 m³ of space. The total disposal cost for the dry solid wastes resulting from preparing the reference BWR for passive safe storage is estimated to be \$142,670.

The sum of the disposal costs for the wet solid wastes and the dry solid wastes is \$1,216,070 and represents the total disposal cost for the radioactive materials.

J.5.1.2 Costs for Staff Labor

The costs for staff labor during preparations for passive safe storage are shown in Table J.5-4. Approximately 66% of the total preparations cost is associated with the staff labor requirements (presented in Table J.4-1). A total staff labor cost of about \$11.3 million is estimated for preparing the

TABLE J.5-3. Costs for Disposal of Dry Solid Wastes

Source (Building)	Number of Disposable Containers (a)	Container Costs (\$) (b)	Estimated No. Requiring Shielding	Cask Rental Costs (\$) (c)	Number of Shipments Shielded (d) / Unshielded	Trans- portation Costs (\$) (e)	Handling Costs (\$) (f)	Burial Volumes (m ³) (g)	Burial Costs (\$) (h)	Total Disposal Costs (\$) (i)
Reactor	348	6 960	108	7 700	8/2	19 200	3 000	3	7 120	43 980
Turbine Generator	456	9 120	140	10 000	10/2	23 100	4 000	96	9 310	55 530
Radwaste and Control	<u>335</u>	<u>6 700</u>	<u>104</u>	<u>7 400</u>	<u>8/2</u>	<u>19 200</u>	<u>3 000</u>	<u>71</u>	<u>6 860</u>	<u>43 160</u>
Totals	1 139	22 780	352	25 100	26/6	61 500	10 000	241	23 290	142 670

(a)Based on a 5:1 compaction of estimated waste volumes into standard 0.21-m³ steel drums; estimated on a taskwise assessment of expected dry solid wastes generation rates.

(b)Based on Table M.2-1 in Appendix M.

(c)Based on Table M.3-1; assumes a maximum of seven containers per cask and 5 days per shipment.

(d)Assumes two casks per shipment.

(e)Based on Table M.4-4 for overweight shipments.

(f)Based on Table M.5-1, cask handling fee.

(g)Includes the disposable container; rounded to next whole m³.

(h)Based on Table M.5-1; surface dose rates assumed to be 0.21 to 1.00 R/hr for those drums requiring shielding during shipment, <0.2 R/hr for all others; rounded to the next highest \$10.

(i)The number of figures shown is for computational completeness only and does not imply accuracy to that many significant figures.

TABLE J.5-4. Costs for Staff Labor During Preparations for Passive Safe Storage

Position	Time Relative to Final Reactor Shutdown (year)					Total Staff Labor Costs(b) (\$ thousands)
	-2	-1	1	2	3	
	Annual	Staff	Labor	Costs	(\$ thousands)(a,b)	
Management and Support Staff						
Decommissioning Superintendent	37.3	74.5	74.5	74.5	59.6	320.4
Secretary	10.2	40.6	60.9	60.9	32.5	205.1
Clerk	0	20.3	40.6	40.6	10.2	111.7
Decommissioning Engineer	31.8	63.5	63.5	63.5	50.8	273.1
Assistant Decommissioning Engineer	21.9	43.8	43.8	43.8	17.6	170.9
Radioactive Shipment Specialist	0	32.9	32.9	32.9	13.2	111.9
Procurement Specialist	6.2	30.7	30.7	30.7	12.3	110.6
Tool Crib Attendant	0	0	46.4	46.4	18.6	111.4
Control Room Operator	0	0	145.0	145.0	60.4	350.9
Security Supervisor	0	0	32.9	32.9	16.5	82.3
Security Shift Supervisor	0	0	153.5	153.5	76.8	383.8
Security Patrolman	0	0	830.7	596.4	138.5	565.6
Contracts and Accounting Specialist	7.7	38.4	38.4	38.4	30.8	153.7
Health and Safety Supervisor	25.2	50.4	50.4	50.4	40.4	216.8
Health Physicist	0	19.7	39.4	39.4	19.7	118.2
Protective Equipment Attendant	0	0	46.4	46.4	23.2	116.0
Industrial Safety Specialist	8.8	43.8	43.8	43.8	21.9	162.1
Quality Assurance Supervisor	8.8	43.8	43.8	43.8	35.1	175.3
Quality Assurance Engineer	11.9	78.8	39.4	39.4	19.7	189.2
Quality Assurance Technician	0	11.6	46.4	46.4	23.2	127.6
Consultant (Safety Review)	20.1	33.5	33.5	33.5	20.1	140.7
Subtotals, Management and Support Staff	189.9	626.3	1 935.9	1 702.6	741.6	5 197.3
Decommissioning Workers						
Shift Engineer	21.9	87.6	87.6	87.6	43.8	328.5
Crew Leader	0	37.3	559.5	496.1	33.6	1 126.5
Utility Operator	0	81.3	1 219.5	869.9	27.1	2 197.3
Laborer	0	0	15.6	77.7	207.2	300.5
Craft Supervisor	0	24.8	39.4	78.8	35.5	178.5
Craftsman	0	271.0	135.5	216.8	379.4	1 002.7
Senior Health Physics Technician	0	32.9	72.4	72.4	46.1	223.8
Health Physics Technician	0	75.3	326.3	228.5	67.8	697.9
Subtotals, Decommissioning Workers	21.9	610.2	2 455.8	2 127.8	840.5	6 056.2
Totals	211.8	1 236.5	4 392.7	3 830.4	1 582.1	11 253.5

(a) Calculated as the product of the data given in Table J.4-1 and the corresponding data given in Table M.1-1; rounded to the next higher \$100.

(b) The number of significant figures shown is for computational completeness and does not imply accuracy to the nearest \$100.

reference BWR for passive safe storage. Specialty contractor labor is not included in this total.

J.5.1.3 Costs for Energy

The costs for energy during preparations for passive safe storage are presented in Table J.5-5, together with the estimated usage of both electricity

TABLE J.5-5. Costs for Energy During Preparations
for Passive Safe Storage

<u>Energy Form</u> <u>System(s) or Equipment</u>	<u>Estimated</u> <u>Energy</u> <u>Usage(a)</u> <u>(MWh)</u>	<u>Costs (\$)(b)</u>
<u>Electricity</u>		
HVAC	36 700	500 500
Main Buildings(c)	36 700	550 500
Control Tower	2 860	42 900
Other Buildings	4 850	72 750
Lighting	6 810	102 150
Communications	260	3 900
Security	1 760	26 400
Fire Protection	90	1 350
Compressed Air	810	12 150
Compressed Air	1 590	23 850
Cranes and Hoists	810	12 150
Domestic Water	9 150	137 250
Contaminated Water(d)	9 780	146 700
Decommissioning Equipment	1 850(e)	27 750
Subtotals, Electricity	76 510	1 147 650
<u>Fuel Oil</u>		
	<u>(m³)</u>	
Building Heating	7 190(f)	949 080
Decontamination Solution Concentrator	195(g)	25 740(g)
Subtotals, Fuel Oil	7 385	924 820
Total Energy Cost		2 122 470

(a)Based on a system/component-wise analysis of specific requirements, unless otherwise noted; rounded to the nearest 10 MWh for electricity and to the nearest whole m³ for fuel oil.

(b)Based on Section M.6 of Appendix M.

(c)Includes the Reactor Building, the Turbine Generator Building, and the radwaste area of the Radwaste and Control Building.

(d)Includes all such systems listed in Appendix C for all anticipated operations; includes appropriate information from Table H.5-9 in Appendix H.

(e)Estimated at about 2% of the total.

(f)Assumes two heating periods during the preparations for passive safe storage.

(g)From Table H.5-9.

and fuel oil. The usage of both energy forms is estimated based on a detailed analysis of the requirements for the essential systems presented in Table H.4-1 in Appendix H and the tasks and schedules during preparations for passive safe storage.

Totals of 76,510 MWh of electricity, costing \$1,147,650, and 7385 m³ of fuel oil, costing \$974,820, are estimated to be used during the preparations for passive safe storage. The total cost for energy is about \$2.1 million and is about 13% of the total cost for preparations for passive safe storage.

J.5.1.4 Costs for Special Tools and Equipment

Based on the information presented previously in Table J.2-1, the estimated costs for the special tools and equipment that are required for preparations for passive safe storage of the reference BWR are presented in Table J.5-6.

TABLE J.5-6. Costs for Special Tools and Equipment for Preparations for Passive Safe Storage

Item	Estimated Number Required ^(a)	Estimated Cost (\$ thousands) ^(b)
Portable Oxyacetylene Torch	2	1
Guillotine Pipe Saw	2	8
Power-Operated Reciprocating Hacksaw	2	1
Closed-Circuit, High-Resolution Television	1	7.5 ^(c)
Submersible Pump with Disposable Filter	5	5
High-Pressure Water Jet	2	40
Low-Pressure Water Jet	1	15
Pipe Jumper	4	3
Mobile Chemical Decontamination Unit	5	100
Mobile Chemical Mixing and Heating Unit	4	10
Scaffolding	50	2.5
Safety Nets	As Required	10
Power-Operated, Mobile, Scissors-Type Manlift	2	64
Power-Operated, Mobile, Articulated-Arm Manlift	1	9
9100-kg-Capacity Forklift	1	13.3
Rigging Materials (e.g., Chockers, Grapples, Winches)	As Required	5
Concrete Drill with HEPA-Filtered Dust Collecting System	1	2
Concrete Surface Spaller	1	5
Front-End Loader (Highly Maneuverable, Light Duty)	1	10
Vacuum Cleaner (HEPA-Filtered)	3	15
Portable Ventilation Enclosure	3	4.5
Supplied-Air Plastic Suit	100	6
Polyurethane Foam Generator	2	10
Paint Sprayer	4	4
Total Cost		350.8

(a)Based on Table J.2-1.

(b)Costs adjusted to early 1978.

(c)Estimated for modifications of existing systems.

The estimated total cost for special tools and equipment is \$350,800, which represents approximately 2% of the total cost of preparations for passive safe storage.

J.5.1.5 Costs for Miscellaneous Supplies

A variety of supplies are used during preparations for passive safe storage. These include expendable glass-fiber and HEPA filters, anticontamination clothing, cleaning and contamination control supplies (chemical agents, sweeping compounds, rags, mops, and plastic bags and sheeting), expendable hand tools, cutting and welding supplies (saw blades, torch gas, and welding rod), carbon steel plates, decontamination chemicals, and filter/demineralizer resins. The estimated costs for these items are given in Table J.5-7. The total estimated cost for miscellaneous supplies used while preparing the reference BWR for passive safe storage is about \$1.4 million, 8% of the total cost for preparations for passive safe storage.

TABLE J.5-7. Costs for Miscellaneous Supplies During Preparations for Passive Safe Storage

Item	Estimated Costs (\$ thousands)
Filters	
Reactor Building and Turbine Generator Building HVAC Exhausts ^(a)	450
Reactor Building, Turbine Generator Building, and Radwaste and Control Building Isolation HEPA Filters ^(b)	37
All Other Requirements	57
Anticontamination Clothing ^(c)	207
Cleaning and Contamination Control Supplies	120
Hand Tools	42
Cutting and Welding Supplies	5
Carbon Steel Plates	19
Decontamination Chemicals ^(d)	335
Filter/Demineralizer Resins ^(d)	89
Total	1 361

(a) Includes filters, housings, and ducting; see Section H.2.8 of Appendix H.

(b) Includes isolation filters, housings, and ducting; see Section J.2-4.

(c) Estimated at four changes per day per decommissioning worker; see Reference 1.

(d) From Table H.5-7, rounded to nearest \$1000.

J.5.1.6 Costs for Specialty Contractors

The estimated costs for specialty contractors are given in Table J.5-8. As discussed in Section H.3 of Appendix H, specialty contractors perform temporary radwaste handling and environmental monitoring during preparations for passive safe storage. In addition, a specialty contractor, who is responsible for security during the continuing care period, begins work during the preparations period, including making a site-security survey, reducing the size of the security area, and procuring and installing the necessary remote-readout security equipment. The costs for a hauling contractor are not shown in Table J.5-8, but are shown as "transportation costs" in Section J.5.1.1 for disposal of radioactive wastes.

TABLE J.5-8. Costs for Specialty Contractors During Preparations for Passive Safe Storage

<u>Specialty Contractor^(a)</u>	<u>Cost Estimating Basis (\$/unit)</u>	<u>Estimated Costs (\$)</u>
Temporary Radwaste Handling	10 000/mo ^(b)	20 000
Environmental Monitoring	38 500/yr ^(c)	89 700
<u>Security</u>	<u>85 800/job^(d)</u>	<u>85 800</u>
Total Cost		195 500

(a) Does not include the hauling contractor, which is shown as "transportation costs" in Section J.5.1.1, with a total cost of \$610,620.

(b) Based on information presented on page 10-4 of Reference 2 and corroborated with Mr. Ed Brooks of Chem-Nuclear Systems, Inc., Bellevue, Washington.

(c) Based on information presented on page H-20 of Reference 2.

(d) Based on information presented as Option C in Table H.4-3 in Reference 2.

The total cost for specialty contractors during preparations for passive safe storage, excluding the hauling contractor, is \$195,500, which is about 1% of the total cost for preparations for passive safe storage.

J.5.1.7 Costs for Nuclear Insurance

The costs for nuclear liability insurance during preparations for passive safe storage are estimated for an assumed policy limit of \$125 million carried through the active decommissioning period. These costs are shown in Table J.5-9. The total estimated cost for nuclear insurance is \$500,000, which represents about 3% of the total cost for preparations for passive safe storage.

TABLE J.5-9. Costs for Nuclear Liability Insurance
During Preparations for Passive
Safe Storage^(a)

<u>Year After Shutdown</u>	<u>Estimated Premium (\$ thousands)</u>
1	240
2	210
<u>3</u>	<u>50</u>
Total	500

(a)Based on information
presented on Page 10-8
of Reference 2.

J.5.1.8 Costs for Licensing Fees

The fees charged for licensing services performed by the NRC are delineated in 10 CFR Part 170.⁽³⁾ The costs for licensing fees while preparing the reference BWR for passive safe storage are shown in Table J.5-10. The total cost for licensing services is \$37,850, which is approximately 0.2% of the total cost for preparations for passive safe storage.

J.5.1.9 Other Possible Costs

Two additional categories of costs could figure into the total cost for preparations for passive safe storage, depending on how they are classified. In this study, they are presented separately, since they cannot be clearly identified as decommissioning costs. The tasks that result in these costs are:

TABLE J.5-10. Costs for Licensing Fees During Preparations for Passive Safe Storage^(a)

<u>Licensing Service Category</u>	<u>Fee (\$)</u>	<u>Total Costs (\$)</u>
Facility License Amendment (Class IV)	12 300	12 300
Health, Safety, and Environmental Inspection	650/year	1 950 ^(c)
<u>Safeguards Inspection</u>	<u>11 800/year</u>	<u>23 600^(d)</u>
Total Cost		37 850

(a) From Reference 3.

(b) Based on one license amendment (i.e., to obtain a possession-only license prior to the start of decommissioning tasks).

(c) Based on the time the license is in force (i.e., approximately 3 years during active decommissioning).

(d) Based on having spent fuel on site for 19 months (i.e., two yearly fees charged).

1) shipment of the spent reactor fuel to an offsite repository and 2) disposal of the fuel channels.

Discussions of the costs associated with these tasks are presented in the following subsections.

Spent Fuel Shipment. Since the ultimate disposition of the spent reactor fuel is not known, it is assumed in this study that the 764 fuel assemblies from the final reactor core load are shipped by rail, together with their fuel channels, to a repository located 2400 km from the reference BWR. The resulting costs for preparations for passive safe storage are the same as those calculated for immediate dismantlement in Section I.3.9.1 of Appendix I.

The total estimated cost for shipping the spent fuel and channels to the assumed repository is about \$3.8 million. If special trains are required, the cost is increased by up to about \$2.5 million.

Fuel Channel Disposal. The assumption that the fuel channels are shipped and stored with the spent fuel is based on practicality rather than on present practice. Presently, this is not a viable option and, therefore, an analysis is performed (in Section I.3.9.4 of Appendix I) to determine the alternative

costs for disposing of the fuel channels as activated wastes in either a shallow-land disposal facility or a deep geologic disposal facility.

The total cost for shallow-land burial of the fuel channels is estimated to be \$617,140. Deep geologic disposal is estimated to cost \$1,047,000, or \$429,860 more than shallow-land disposal. Shipment of the fuel channels by special train would cost an additional \$456,000, maximum.

J.5.2 Costs for Continuing Care During Passive Safe Storage

The costs for continuing care during passive safe storage of the reference BWR are developed in this section. Also included are details of security during continuing care.

The organization shown in Figure J.4-2 of Section J.4.6 is postulated to take over the surveillance, maintenance, and security of the reference BWR for the duration of the continuing care period. Based on the estimated labor requirements given in Table J.4-2 and on the miscellaneous costs anticipated, the total annual costs for continuing care are tabulated in Table J.5-11.

Ongoing Environmental Monitoring Program. The estimated costs of the ongoing environmental monitoring program during the continuing care period are given in Table J.5-12. Identification of the labor-intensive tasks for the various positions are included in Table J.5-12 for completeness.

Security Costs. The information contained in this subsection forms the bases for the estimated security costs during continuing care. It is assumed that at the start of the continuing care period the guard force is disbanded and replaced by private security forces and electronic intrusion systems.

The costs of a private security agency using remote surveillance equipment are given in Table J.5-13. Consideration is 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 areas

TABLE J.5-11. Estimated Annual Costs for Surveillance, Maintenance, and Security During Continuing Care

<u>Labor</u>	<u>Estimated Annual Cost (\$)</u> ^(a)
Surveillance and Maintenance Representative	6 500
Secretary	5 075
Repairman	2 710
Security	8 800 ^(b)
Third-Party Inspection	7 500 ^(c)
Environmental Radiological Monitoring Program Personnel	14 230
Quality Assurance Specialist	1 000
<u>Total, Labor Costs</u>	<u>45 815</u>
<u>Other Costs</u>	
Equipment and Supplies	1 000
Annual Allowance for Repairs	5 000
Utilities and Services	5 000
License Fee	650
NEL-PIA Insurance	2 500
<u>Total, Other Costs</u>	<u>14 150</u>
Subtotal	59 965
<u>Contingency (25%)</u>	<u>14 991</u>
<u>Total, Annual Continuing Care Costs</u>	<u>74 956</u>

(a)The number of significant figures shown is for computational completeness and does not imply accuracy to the nearest dollar.

(b)See Table J.5-13 for initial costs.

(c)Third-party inspection costs are based on an assumed cost of \$500 per man-day.

TABLE J.5-12. Estimated Annual Costs for an Ongoing Environmental Monitoring Program During Continuing Care

<u>Labor</u>	<u>Task</u>	<u>Estimated Cost (\$)(a)</u>
Health Physicist	Collect data, archive storage of samples and data.	1 450
Health Physicist, Supervisor	Preparation of reports and data analysis.	3 940
Radiochemist	Laboratory analysis of samples.	3 940
Instrument Specialist	Maintain equipment, calibrations.	2 710
Quality Assurance Specialist	Audit functions.	900
<u>Total, Labor Costs</u>		<u>12 940</u>
<u>Contractor's 10% Overhead and Profit</u>		<u>1 290</u>
		14 230
<u>Other Costs</u>		
Utilities and Services		1 000
<u>Equipment and Supplies</u>		<u>1 000</u>
<u>Total, Other Costs</u>		<u>2 000</u>
Subtotal		16 230
<u>Contingency (25%)</u>		<u>4 058</u>
<u>Total, Annual Environmental Monitoring Program Costs</u>		<u>20 288</u>

(a)The number of significant figures shown is for computational completeness and does not imply accuracy to the nearest dollar.

TABLE J.5-13. Estimated Security Costs During Continuing Care^(a,b)

<u>Activities</u>	<u>Cost (\$ thousands)</u>
Site Security Survey	2.0 ^(c)
Reduce Circumference of of Existing Security Fence	10.0 ^(c)
Utilize Portions of Exist- ing Security Equipment; Add Others	40.0 ^(d)
Install Remote Readout at Offsite Location	25.0 ^(c,e)
Lease of Equipment, Main- tenance, Monitoring, Response (includes tem- porary maintenance when required), Patrol (unscheduled, any hour), and Supervision	8.8
Total (first year)	85.8
Total/year (after first year)	8.8

- (a) Includes onsite detection and notification systems for: security (intrusion), fire, and radiation levels.
- (b) Data taken from Table H.4-3 of Reference 2.
- (c) One-time cost only.
- (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-hour video tape recorder, multiplex systems (self trouble-shooting), battery backup power where required, and capacitance perimeter fence protection.
- (e) Includes installation charges for security agency hardware, which are primarily for the offsite installation costs of "slave" annunciator drops from existing plant security equipment.

- addition of special equipment as indicated by the security agency's site survey
- installation of remote readout facilities either onsite or offsite.

In summary, security is provided during continuing care by several methods. Locks on the gates in the fence around the decommissioned facility provide the first level of security. The fence is maintained in good condition throughout the continuing care period. The locks and secured entryways of the structures and the remote surveillance equipment and alarm system provide the final level of security.

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 J.5-1 delineating responsibilities for reportable events. It is imperative that such a notification and reporting process be kept current throughout the continuing care period.

J.6 ESTIMATED OCCUPATIONAL RADIATION DOSES FOR PASSIVE SAFE STORAGE

In this section, detailed analyses are made to estimate the external occupational radiation dose accumulated by decommissioning workers while preparing the reference BWR for passive safe storage and continuing care. The analyses are performed in the same manner as those described for immediate dismantlement in Section I.4 of Appendix I.

The total corrected external occupational radiation dose for preparations for passive safe storage is estimated to be about 375 man-rem. The dose accumulated during continuing care is estimated to be about 1.3 man-rem the

FIGURE J.5-1. Postulated Security and Administrative Notification and Reporting Process

first year, diminishing each year until after 50 years a total of 10 man-rem has been accumulated by the surveillance, maintenance, and security forces.

J.6.1 Estimated External Occupational Radiation Dose During Preparations for Passive Safe Storage

Some of the same basic activities that are performed for immediate dismantlement are also performed for the preparations for passive safe storage. These include defueling the reactor, comprehensive radiation surveys, weekly radiation surveys, general cleanup, system decontamination and draining, and radioactive liquid waste handling. Other tasks, of course, are different, but the external occupational doses for all tasks are based on a 6-hour effective working day in the radiation zone per worker, the source of radiation dose and its intensity, and the percentage of time a given worker spends working on that particular

task. Subjective judgement is used for each task analysis regarding the percentage of time spent on that task by individual workers, since each task is different.

The results of the analyses for the preparations for passive safe storage are presented in Table J.6-1. The dose estimates for each task are corrected for radioactive decay using the half-life of ^{60}Co to the time when the task is one-half completed.

J.6.2 Estimated Occupational Radiation Dose During the Continuing Care Period

From the types and quantities of manpower given in Section J.4-6, and the estimated radiation dose data for continuing care, the anticipated cumulative occupational external radiation dose for surveillance, maintenance, and security activities for intervals of 10, 30, 50, and 100 years after final reactor shutdown are computed. During continuing care, it is postulated that the inspection and maintenance personnel spend only a small percentage (from 2 to 10% of their onsite time in radiation zones. The balance of their time is postulated to be spent in areas outside of the exclusion area or offsite.

The results of these analyses are presented in Table J.6-2. The majority of the occupational dose to personnel is accumulated during the first 30 years after shutdown of the reference BWR. Less than 0.5 man-rem total occupational external radiation dose are accumulated during the next 70 years.

J.7 DEFERRED DISMANTLEMENT

It is possible that sometime after the reference BWR is put in passive safe storage the owner will want to eliminate the continuing care responsibilities and terminate the possession-only license. A possession-only license may be terminated (in accordance with present regulations) when the facility is sufficiently decontaminated to prevent risk to the public. Depending on the length of the continuing care period, deferred dismantlement is a way to achieve the necessary decontamination of the reference BWR.

TABLE J.6-1. Estimated Occupational Doses for Preparations for Passive Safe Storage

Location	Task	Supervisors (a)		Utility Operators and Laborers		Craftsmen		Health Physics Technicians		Task Totals			Corrected Dose (c)
		Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Decay Factor (b)	
Reactor Building/Primary Containment													
1.	Install HEPA Filters (d)	400	0.400	-- (e)	--	2 712	2.712	270	0.270	3 382	3.382	1.000	3.382
2.	Comprehensive Radiation Survey (d)	--	--	--	--	--	--	100	0.500	100	0.500	0.995	0.498
3.	Discharge and Ship Fuel (d)	941	8.469	6 268	56.412	--	--	2 752	24.768	9 961	89.649	0.904	81.043
4.	Drain Suppression Pool to Radwaste; Water-Jet Clean (d)	27	0.189	322	1.932	27	0.135	27	0.135	403	2.391	0.957	2.288
5.	Drain Reactor Well Pool to Radwaste; Water-Jet Clean (d)	3	0.015	95	0.275	5	0.025	5	0.025	68	0.340	0.931	0.317
6.	Chemical Decon Reactor Water Recirculation and Cleanup Systems (d)	4	0.200	572	5.143	8	0.760	36	1.836	620	7.939	0.921	7.312
7.	Clean Up, Stage and Shield Hot Spots (f)	75	0.750	500	5.000	--	--	50	0.500	625	6.250	0.913	5.706
8.	Drain Contaminated Systems to Radwaste (d)	--	--	12	0.180	4	0.060	2	0.030	18	0.270	0.903	0.244
9.	Chemical Decon Residual Heat Removal, Low- and High-Pressure Core Spray Systems (d)	12	0.360	259	4.405	20	1.100	118	3.540	409	9.403	0.893	8.397
10.	Drain Dryer and Separator Pool to Radwaste; Water-Jet Clean (d)	9	0.045	97	0.485	8	0.040	8	0.040	122	0.410	0.886	0.452
11.	Chemical Decon Drain Systems (d)	--	--	431	0.431	108	0.108	--	--	539	0.539	0.879	0.474
12.	Drain Spent Fuel Pool to Radwaste; Water-Jet Clean	9	0.045	109	0.545	9	0.045	9	0.045	136	0.680	0.802	0.545
13.	Chemical Decon Fuel Pool Cooling and Cleanup System (d)	2	0.010	52	0.312	2	0.020	18	0.108	74	0.450	0.791	0.356
14.	Cover and Seal Spent Fuel Pool and Dryer and Separator Storage Pool	57	0.285	208	1.040	172	0.860	18	0.090	455	2.275	0.787	1.790
15.	Seal Equipment and Personnel Hatches into Primary Containment (g)	79	0.395	528	2.640	264	1.320	53	0.265	924	4.620	0.785	3.627
16.	Decontaminate HVAC Electrical Miscellaneous Steel Structures and Equipment and Concrete; Apply Protective Paint	217	0.989	1 436	6.580	--	--	145	0.662	1 798	8.240	0.770	6.345
17.	Isolate and Seal Equipment, Piping, Rooms, Stack HVAC Ducts, Rail Tunnel and Steam Tunnel	256	3.200	1 140	14.250	1 068	13.370	202	2.525	2 666	33.325	0.751	25.027
18.	Seal Drywell Top Head and Unneeded Reactor Building Doors (g)	59	0.118	--	--	396	0.792	40	0.800	495	0.990	0.741	0.734
19.	Install HEPA-Filtered Vents	11	0.077	36	0.252	54	0.378	11	0.077	112	0.784	0.741	0.581
20.	Deactivate Unnecessary Utilities	94	0.752	313	2.504	313	2.504	32	0.256	752	6.016	0.737	4.434
21.	Install Intrusion, Radiation Monitoring and Fire Alarm Systems	200	0.600	80	0.240	240	0.720	80	0.240	600	1.800	0.730	1.314
22.	Final Radiation Survey (h)	--	--	--	--	--	--	100	0.400	100	0.400	0.732	0.239
Subtotals, Reactor Building (c, i)		2 455	17	12 418	102	5 410	25	4 076	37	24 359	181		155
Turbine Generator Building													
1.	Install HEPA Filters (d)	1 100	1.100	--	--	7 450	7.450	745	0.745	9 295	9.295	1.000	9.295
2.	Comprehensive Radiation Survey (d)	--	--	--	--	--	--	100	0.200	100	0.200	0.896	0.179
3.	General Cleanup (d)	100	0.100	700	0.700	--	--	70	0.070	870	0.870	0.886	0.771
4.	Drain Contaminated Systems to Radwaste; Water-Jet Clean Condensate Storage Tanks (d)	8	0.008	51	0.051	--	--	5	0.005	64	0.064	0.867	0.056
5.	Drain Condenser to Radwaste; Water-Jet Clean (j)	72	0.072	480	0.480	--	--	48	0.048	600	0.600	0.838	0.503
6.	Chemical Decon Drain Systems (d)	--	--	305	0.305	76	0.076	--	--	381	0.381	0.822	0.313
7.	Decontaminate HVAC, Electrical Miscellaneous Steel Structures and Equipment and Concrete; Apply Protective Paint	148	0.383	980	2.532	--	--	31	0.186	1 159	3.101	0.813	2.521
8.	Isolate and Seal Equipment, Piping, Rooms, Stack and HVAC Ducts	140	0.420	518	1.560	626	1.878	97	0.291	1 381	4.148	0.791	3.277
9.	Install HEPA-Filtered Vents	3	0.003	11	0.011	17	0.017	3	0.003	34	0.034	0.783	0.027
10.	Deactivate Unnecessary Utilities	60	0.060	198	0.198	198	0.198	20	0.020	476	0.476	0.780	0.371
11.	Install Intrusion, Radiation Monitoring and Fire Alarm Systems	200	0.300	80	0.120	240	0.360	80	0.120	600	0.900	0.772	0.695
12.	Final Radiation Survey (k)	--	--	--	--	--	--	100	0.200	100	0.200	0.766	0.153
Subtotals, Turbine Generator Building (c, i)		1 831	2	3 323	6	8 607	10	1 299	2	15 060	20		18
Radwaste and Control Building													
1.	Comprehensive Radiation Survey	--	--	--	--	--	--	80	0.080	80	0.080	0.862	0.069
2.	General Cleanup	15	0.075	180	2.700	--	--	15	0.075	210	2.850	0.855	2.437
3.	Decontaminate Equipment External Surfaces; Apply Protective Paint	45	0.180	360	21.600	240	4.800	45	0.900	690	27.480	0.848	23.303
4.	Decontaminate Electrical Equipment, Miscellaneous Steel Structures, and HVAC; Apply Protective Paint	122	0.122	648	0.648	576	0.576	109	0.109	1 455	1.455	0.836	1.216
5.	Decontaminate Concrete; Apply Protective Paint	33	0.132	436	6.540	345	3.450	33	0.132	847	10.254	0.827	8.480
6.	Chemical Decon Drain Systems	--	--	435	0.435	109	0.109	--	--	544	0.544	0.791	0.430
7.	Install Temporary Radwaste System	2	0.004	16	0.032	24	0.048	2	0.004	44	0.088	0.780	0.069
8.	Chemical Decon Equipment	85	0.850	828	6.624	367	12.111	99	0.990	1 379	20.575	0.780	16.048
9.	Install HEPA-Filtered Vents	309	2.472	1 236	9.888	1 854	14.832	309	2.472	3 708	29.664	0.755	22.396
10.	Deactivate Unnecessary Equipment and Utilities	132	0.132	264	0.264	396	0.396	66	0.066	858	0.858	0.766	0.657
11.	Isolate and Seal Equipment and Areas	309	2.472	1 236	9.888	1 854	14.832	309	2.472	3 708	29.664	0.755	22.396
12.	Install Intrusion, Radiation Monitoring and Fire Alarm Systems	160	0.480	40	0.040	240	0.240	70	0.070	510	0.830	0.735	0.610
13.	Final Radiation Survey	--	--	--	--	--	--	176	0.176	176	0.176	0.726	0.128
Subtotals, Radwaste and Control Building (c, i)		1 212	7	5 679	59	6 005	51	1 313	8	14 209	125		99
Site and Support Facilities													
1.	Final Radiation Survey	--	--	--	--	--	--	2 112	0.010	2 112	0.010	0.726	0.007
Subtotals, Site and Support Facilities (c, i)		0	0	0	0	0	0	2 112	0	2 112	0		0
Ancillaries													
1.	Operate Radwaste Systems (d)	--	--	286	1.430	144	21.600	36	0.495	466	23.525	0.877	20.631
2.	Routine Radiation Surveys	--	--	--	--	--	--	996	4.084	996	4.084	0.860	3.512
3.	Package Dry Solid Wastes	--	--	752	15.040	15	0.450	--	--	767	15.490	0.858	13.290
4.	Miscellaneous (k)	--	--	--	--	--	--	--	--	--	--	--	66
Subtotals, Ancillaries (c, i)		0	0	1 038	16	159	22	1 032	5	2 229	43		103
Totals		5 498	26	22 458	183	20 181	108	9 832	52	57 969	369		375

- (a) Includes shift engineers, crew leaders, craft supervisors and senior health physics technicians.
(b) The decay factor is based on the half-life of ⁶⁰Co and is calculated to the mid-point of each task.
(c) The number of significant figures shown is for computational convenience and does not imply precision of that degree.
(d) Same as shown in Table I.4-1 in Appendix I.
(e) A dash means that category of decommissioning staff is not involved in the particular task.
(f) Estimated 65% of that shown in Table I.4-1.
(g) Same as shown in Table K.4-1 in Appendix K.
(h) Estimated 80% of that shown in Table I.4-1.
(i) Dose totals are rounded to the nearest whole number.
(j) Estimated 25% of that shown in Table I.4-1.
(k) Estimated twice that shown in Table I.4-1.
(l) Consists of an allowance of up to 1 rem per year for selected management and support staff whose radiation doses are not specifically estimated, together with an allowance of 5% of the total explicitly estimated task radiation dose, to account for any omissions and uncertainties in the analyses.

TABLE J.6-2. Estimated Cumulative Occupational External Radiation Dose During Continuing Care

Estimated Dose for the First Year of Continuing Care

<u>Labor</u>	<u>Estimated Exposure (man-hours)</u>	<u>Average Dose Rate (R/hr)^(a)</u>	<u>Estimate Dose (man-rem)</u>
Surveillance and Maintenance Representative	26	0.005	0.13
Repairman	50	0.010	0.50
Third-Party Inspections	48	0.005	0.24
Quality Assurance Specialist	10	0.002	0.02
Radiation and Environmental Surveillance	50	0.005	0.25
Security	<u>90</u>	0.002	<u>0.18</u>
Totals	274		1.32

<u>Cumulative Occupational Radiation Dose For:</u>	<u>Total Cumulative Radiation Dose (man-rem)^(b)</u>
The First Year of Continuing Care (CC)	1.32
10 Years After Final Reactor Shutdown (8 years of CC)	6.53
30 Years After Final Reactor Shutdown (28 years of CC)	9.78
50 Years After Final Reactor Shutdown (48 years of CC)	10.02
100 Years After Final Reactor Shutdown (98 years of CC)	10.04

(a) Estimated for the start of the continuing care period.

(b) Calculated assuming ^{60}Co to be the dominant occupational dose contributor during the first 100 years of continuing care.

The same basic activities necessary for immediate dismantlement are also assumed for deferred dismantlement following passive safe storage. Neutron-activated materials in the reactor vessel internals and the reactor vessel still require removal and segmenting remotely under water, using the same techniques that are assumed for immediate dismantlement. The sacrificial shield concrete is still radioactive due to activated trace elements, such as ^{152}Eu and ^{154}Eu , and may have to be removed.

The radioactive contamination of the piping systems, equipment, pool liners, etc., is primarily ^{60}Co . Thus, for continuing care periods of less than 50

years (~ 10 half-lives for ^{60}Co), it is assumed the surfaces remain radioactively contaminated at levels greater than those that permit unrestricted use of the material. It is also assumed that after 50 years of decay, the radioactive contamination on the bulk of the formerly contaminated material has decayed to levels that permit release for either salvage as scrap or disposal as nonradioactive waste.

A number of dismantlement tasks are accomplished during the preparations for passive safe storage (i.e., discharging and shipping the fuel; draining and decontamination of contaminated liquid systems; and removal of radioactive wastes, such as filters, resins, and evaporator bottoms). During deferred dismantlement, the time not expended on these tasks is offset by the time spent on familiarization of the work force with the facility, removal of locks and barriers installed to secure the facility, and restoration of essential services that were unneeded during the continuing care period. Therefore, it is assumed that the basic work force and time required for deferred dismantlement are the same as for immediate dismantlement.

J.7.1 Costs for Deferred Dismantlement

The costs of accomplishing deferred dismantlement are estimated by examining the general cost categories for immediate dismantlement shown in Table I.3-1 in Appendix I and determining the impact of the differences in accomplishing dismantlement after periods of continuing care of several lengths. It is assumed that the management and support staff is the same for deferred dismantlement as it is for immediate dismantlement. However, fewer decommissioning workers are required for deferred dismantlement, since the radiation dose rates are lower.

Estimates are given in Table J.7-1 of the volumes of the various types of radioactive materials that are packaged and shipped for burial when dismantlement starts either immediately or 10 to 30, 50, and 100 years after reactor shutdown. The volume of radioactive waste from preparations for passive safe storage is also given.

The volume of activated material is assumed to remain constant over the 100-year time span. Depending on the particular radionuclides present in the

TABLE J.7-1. Burial Volumes of Radioactive Materials from Decommissioning

Decommissioning Alternative	Start of Decommissioning (years after shutdown)	Burial Volume (m ³)				Decommissioning Total
		Activated Material	Contaminated Material	Radioactive Waste	Alternative Total	
Immediate Dismantlement	0	228	17 219	1 492	18 939	18 939
Preparations for Passive Safe Storage	0	--	--	1 055	1 055	--
Deferred Dismantlement	10	228	17 219	437	17 884	18 939
	30	228	17 219	437	17 884	18 939
	50	228	150	350	728	1 783
	100	228	150	240	618	1 673

sacrificial shield, the activated concrete may decay to unrestricted use levels in 100 years, but this is not assumed in this study.

The volume of contaminated material is assumed to remain constant through 30 years, but to decrease to 150 m³ by 50 years and thereafter.

The volume of radioactive waste estimated for dismantlement after 10 to 30 years is assumed to be the difference between the volumes for immediate dismantlement and for preparations for passive safe storage. This volume is assumed to be reduced to 350 m³ at 50 years and 240 m³ at 100 years.

Estimated costs are given in Table J.7-2 for dismantlement of the reference BWR after 10 to 30, 50, and 100 years of passive safe storage. The estimated costs for immediate dismantlement, taken from Table I.3-1 in Appendix I, are included for comparison. The only cost category that does not decrease with time is the disposal of activated material.

The total cost for each of the deferred dismantlement decommissioning options is given in Table J.7-3. The values listed under "total" include the cost of continuing care during safe storage for the number of years shown. For example, the total given for deferred dismantlement starting 30 years after reactor shutdown includes the cost of preparations for passive safe storage, the cost of continuing care for the period from the end of preparations for passive safe storage until the start of dismantlement and the cost of dismantlement.

TABLE J.7-2. Estimated Dismantlement Costs

Cost Category	Immediate(a) Dismantlement	Costs (\$ millions)		
		Dismantlement	Deferred	
		10 to 30 Years	50 Years	100 Years
Disposal of Radioactive Materials				
Neutron Activated Materials	2.300	2.300	2.300	2.300
Contaminated Materials	4.909	4.909	0.043	0.043
Radioactive Wastes	1.469	0.255	0.204	0.140
Staff Labor	17.561	16.545	14.210	14.210
Energy	3.519	1.479	1.479	1.479
Special Tools and Equipment	2.016	1.728	1.728	1.728
Miscellaneous Supplies	1.859	0.590	0.590	0.590
Specialty Contractors	0.356	0.168	0.168	0.168
Nuclear Insurance	0.800	0.400	0.400	0.400
License Fees	0.051	0.020	0.020	0.020
Subtotal	34.840	28.394	21.142	21.078
Contingency (25%)	8.710	7.099	5.286	5.270
Totals	43.550	35.493	26.428	26.348

(a) From Table I.3-1 in Appendix I.

TABLE J.7-3. Total Costs for Various Decommissioning Alternatives

Years After Reactor Shutdown	Decommissioning Costs in Millions of 1978 Dollars				Total
	Immediate Dismantlement	Preparations For Passive Safe Storage	Continuing Care	Deferred Dismantlement	
0	43.6	--	--	--	43.6
10	--	21.3	0.6	35.5	57.4
30	--	21.3	2.1	35.5	58.9
50	--	21.3	3.6	26.4	51.3
100	--	21.3	7.4	26.3	55.0

J.7.2 Occupational Radiation Dose from Deferred Dismantlement

For this study it is assumed that the occupational radiation dose 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 dose of N_0 , that same task performed t years later during deferred dismantlement would cause a dose 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 are performed in approximately the same way as for immediate dismantlement, using the same techniques and equipment, the occupational radiation dose accumulated during deferred dismantlement can be estimated from the product of the dose that was accumulated during immediate dismantlement times the decay factor for ^{60}Co over the passive safe storage period. Several of the job items given in Table I.4-1 in Appendix I for the radiation dose accumulated during immediate dismantlement, such as reactor defueling and chemical decontamination, should be subtracted from the total dose before reduction, since they are not performed during deferred dismantlement. Several other items, such as unsecuring the facility and re-establishing essential services, should be added. However, for purposes of estimating the occupational dose for deferred dismantlement, a simple reduction of the immediate dismantlement dose 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 J.7-4 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 J.7-4. Estimated Occupational Radiation Dose from Various Dismantlement Alternatives

Years After Reactor Shutdown	Occupational Radiation Dose (man-rem)				Totals
	Immediate Dismantlement	Preparations for Passive Safe Storage	Continuing Care	Deferred Dismantlement	
0	1 845	--	--	--	1 845
10	--	375	1.3	495	871
30	--	375	6.5	36	418
50	--	375	10.0	3	388
100	--	375	10.0	<1	386

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3. U.S. Code of Federal Regulations, Title 10, Part 170, "Fees for Facilities and Materials Licenses and Other Regulatory Services Under the Atomic Energy Act of 1954, As Amended", Superintendent of Documents, GPO, Washington, DC 20555, 1979.

* Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and the National Technical Information Service, Springfield, Virginia 22161.

APPENDIX K

ENTOMBMENT DETAILS

Entombment is one of the alternative modes considered in this study for decommissioning the reference BWR. Entombment requires the encasement of all unreleasable radioactive materials not removed from the facility in concrete or other structural material sufficiently durable (strong and structurally long-lived) to ensure retention of the radioactivity until it decays to levels permitting unconditional release.⁽¹⁾ If the entombed radioactivity includes significant levels of long-lived neutron-activation products (e.g., ⁵⁹Ni with an 80,000-year half-life), the required retention period may be exceedingly long (i.e., tens of thousands to millions of years), depending on the acceptable release limits for residual radioactivity.

This appendix discusses the following aspects of entombing the reference BWR:

- postulated entombment scenarios
- entombment tasks and schedules
- entombment costs
- radiological occupational safety of entombment.

The information in this appendix forms the bases for the activities, costs, and occupational radiation doses for entombment presented in Chapters 9, 10, and 11 in Volume 1, respectively.

K.1 POSTULATED ENTOMBMENT SCENARIOS

Entombment of radioactive materials in the reference BWR is assumed to take place within the confines of the steel primary containment vessel and the surrounding concrete biological shield. All plant areas outside of the entombment barrier are decontaminated to allow unrestricted release if desired.

The entombment barrier consists of the primary containment vessel enclosed within a monolithic concrete envelope (the sealed biological shield) that rests

on the Reactor Building foundation mat. Sealing of all barrier penetrations (e.g., for personnel, equipment, material, and services) is necessary. Equipment and personnel access openings, as well as the stub ends of cut-off piping, are sealed by welded plate closures. All openings through the biological shield are then filled with cast-in-place, reinforced concrete. The removable concrete head plugs are grouted in place to complete the encasement of the radioactive materials within an integral, monolithic concrete envelope. The postulated entombment structure for the reference BWR is illustrated in Figure K.1-1.

Entombment structures other than the one considered in this study may possibly be used for BWR decommissioning. Some of these possible entombment structures are independent of the plant design (i.e., generic to all BWRs). An example of this type is the entombment of the turbine generator and main condenser in place. The biological shield surrounding this equipment could be extended over the top of the equipment and all penetrations sealed to provide a durable monolithic structure. This would involve the construction of a reinforced concrete roof at the top of the shield wall, with structural columns added within the enclosed area to support the roof. In addition, reinforcing steel would be installed at all penetrations in the shield wall and the penetrations would then be filled with cast-in-place, reinforced concrete. This structure would contain enough volume to entomb the turbine generator, the main condenser, the remainder of the contaminated material in the Turbine Generator Building, and additional contaminated material removed from the Reactor Building and/or the Radwaste and Control Building.

Other entombment structure possibilities may be present in individual BWRs. These are design-dependent; they occur as a result of the specific design of the plant in question. An example of such a design-dependent entombment structure in the reference BWR is the use of the tank and pump area along the east wall of the Radwaste and Control Building at elevation 133.2 m (see Section C-C, Figures C.2-11 and C.2-13 in Appendix C). Entombment in this area would involve sealing all material and services penetrations, moving the radioactive materials to be entombed into the area, and sealing the three access hatchways. Structural strengthening may also be required to increase the durability of the resulting entombment structure.

FIGURE K.1-1. Postulated Entombment Structure for the Reference BWR

More than one entombment structure could be used in the decommissioning of a single reactor facility. This would increase the costs of construction and subsequent surveillance of the entombment barriers. However, the overall cost of decommissioning might be decreased because of the reduction or elimination of the volume of radioactive material requiring offsite disposal.

The entombment scenarios considered in this study, the adaption of the primary containment vessel to entombment, and the storage volume availability within the postulated entombment structure are discussed in the following subsections.

K.1.1 Entombment Scenarios

Two entombment scenarios, both using the postulated entombment structure shown previously in Figure K.1-1, are considered in this study: 1) entombment with the reactor vessel internals removed and 2) entombment with the reactor vessel internals in place. This latter scenario is really a form of hardened safe storage since eventual dismantlement is necessary. For both entombment scenarios, dismantlement of the facility outside the entombment structure is carried out as discussed in Section I.1 of Appendix I, with the difference being that as much as possible of the contaminated equipment and material is placed in the entombment structure rather than being packaged and shipped to offsite disposal. This reduces the radioactive material disposal costs associated with initial decommissioning. The plant structures (other than the entombment structure) may also be demolished following completion of the entombment activities (see Appendix L). However, demolition of the Reactor Building is questionable for two reasons: 1) the postulated entombment structure is an integral part of the Reactor Building, with the two sharing many structural features (including the foundation), and 2) the Reactor Building, if left in place, provides a secondary barrier around the entombed materials, providing weather protection and enhanced security for the entombment structure. For these two reasons, the Reactor Building is assumed in this study to be left in place.

Further discussions of scenario 1 and scenario 2 for entombment of the reference BWR follow.

K.1.1.1 Scenario 1

Those reactor vessel internals containing long-lived activation products (^{59}Ni , ^{94}Nb) are removed from the facility and shipped offsite to a nuclear waste repository. As much as possible of the remaining radioactive material in the plant is consolidated within the entombment structure. Because of the relatively short half-lives of the entombed radioactivity, it may be possible to terminate the nuclear license after a continuing care period of about 110 years and release (for unrestricted use) the entombment structure. It should be noted, however, that current regulations and regulatory guidance require a comprehensive survey prior to such release to verify that the radioactive contamination remaining is within acceptable release limits.

K.1.1.2 Scenario 2

The reactor vessel internals (with associated long-lived radionuclides) are left in place, and as many radioactively contaminated materials as possible are moved inside the entombment structure. Under existing regulations, the licensee is responsible for surveillance and maintenance of the entombment structure as a nuclear waste repository for an indefinite time period, unless the reactor vessel internals are subsequently removed.

K.1.2 Adaptation of the Primary Containment Vessel and Biological Shield to Entombment

The primary containment vessel is a free-standing steel pressure vessel with a floor dividing the drywell (upper portion) from the wetwell or suppression chamber (lower portion). The vessel is enclosed within the concrete biological shield wall, as shown previously in Figure K.1-1, but is separated from the wall by an annulus of compressible insulation material about 50 mm thick. Further description of the vessel and shield can be found in Appendix C.

Following chemical decontamination of the reactor and related systems (e.g., recirculation loops) and removal of all liquid volumes (see Section H.5 of Appendix H), the primary containment vessel and biological shield are ready for modification to accommodate entombment.

For scenario 1, the reactor vessel internals are removed and shipped off-site for disposal, thus removing the long-lived activation products and also increasing the volume available for contaminated material storage inside the entombment structure.

For both entombment scenarios, the wetwell (suppression chamber) is the primary storage area for contaminated materials. Additional access to the wetwell is obtained by cutting holes in the drywell floor immediately inside the drywell personnel and equipment hatches. The downcomer vent pipes in the wetwell are cut off near the bottom surface of the drywell floor (the ceiling of the wetwell) and then cut into shorter segments, thus allowing consolidation of their volume and more complete use of the wetwell space. Piping that penetrates the primary containment vessel (and the biological shield) is cut off flush with the interior vessel wall and at points sufficiently far away from the wall (inside) to allow adequate space for seal-welding all wall openings with steel plate. Short sections of piping or conduit removed to allow working space are placed inside the reactor pedestal in the wetwell cavity. The original wetwell hatch and the drywell access hatches are sealed with welded steel plate after the vessel volume below them is filled with contaminated materials. After seal-welding, all penetrations through the biological shield are filled with cast-in-place reinforced concrete.

The integrated schedule and sequence of tasks for adapting the primary containment vessel and the biological shield as the entombment structure and filling the structure with contaminated materials are presented later in Section K.2.

K.1.3 Storage Volume Within the Entombment Structure

The amount of contaminated material that can be entombed inside the primary containment vessel depends on the free and easily filled volume available for use within the vessel. Table K.1-1 shows estimates of the gross volume; the volume displaced by existing piping equipment, and internal structures; and the net volume available for storage of contaminated material within the drywell, the wetwell, and the vessel as a whole. As shown in the table, about 13,330 m³ is available for storage in the entombment structure. For scenario 1, the storage volume is increased by about 570 m³ by the removal of the activated reactor vessel internals, for a total of about 13,900 m³. (For scenario 2,

TABLE K.1-1. Estimated Storage Volume Within the
Primary Containment Vessel

Gross Volume Information ^(a)				Displacement Information ^(a)		Net
Portion of Vessel	Elevation (m)	Radius(m) ^(b)	Volume(m ³)	Item	Volume (m ³)	Volume (m ³) ^(c)
<u>Drywell:</u>						
Drywell Cover:				Concrete:		
Top Ellipsoid	180.1 to 182.5	4.83	117.3	Sacrificial Shield	247	
Cylindrical Section	177.7 to 180.1	4.83	175.9	Reactor Pedestal	224	
Conical Drywell Section:				Piping and Valves: ^(d)	272	
Adapter	174.7 to 177.7	5.7 to 4.83	261.9	Equipment: ^(e)		
Adapter to Floor	152.7 to 174.7	13.1 to 5.7	6 422.4	Reactor Vessel	990 ^(f)	
				HVAC Systems	70	
				Recirculation Pumps	143	
				Miscellaneous Items:		
				Hoists, Radial Beams, Stairs and Ladders, etc.	~200	
<u>Subtotals</u>			<u>6 977</u>		<u>2 146</u>	<u>4 830</u>
<u>Wetwell:</u>						
Cylindrical Section	137.1 to 152.7	13.1	8 410.4	Concrete:		
Bottom Ellipsoid	130.5 to 137.1	13.1	2 372.2	Drywell Floor	82	
				Reactor Pedestal	720	
				Floor Liner	650	
				Piping: ^(d)		
				Downcomer Vents	344	
				Downcomers	115	
				Piping Penetrations	109	
				Structural Members:		
				Support Columns	210	
				Radial Beams and Personnel Access Ways	50	
<u>Subtotals</u>			<u>10 783</u>		<u>2 280</u>	<u>8 500</u>
<u>Totals^(c)</u>			<u>17 760</u>		<u>4 430</u>	<u>13 330</u>

(a) Number of figures shown is for computational completeness and does not imply accuracy to that many figures.

(b) Derived from information in Section C.2.1.1 of Appendix C.

(c) Totals are rounded to the nearest 10 m³.

(d) Derived from information in Table C.3-8 in Appendix C.

(e) Derived from information in Tables C.3-3 and C.4-2 in Appendix C.

(f) Includes the volume below the reactor vessel enclosed by the reactor pedestal, which is assumed unavailable for use without removal of the reactor vessel.

it is possible [with the dryer and separator removed] to pour smaller items, such as contaminated concrete rubble, into the reactor vessel to filter down between the internals, thus increasing the entombment storage volume by several hundred cubic meters. However, to make effective use of this space, segregation of materials by size is required. To simplify the analysis, this approach is not considered further.)

Because of the variety of shapes and sizes of both the volume available within the entombment structure and the contaminated materials to be stored there, as well as the difficulty in placing materials in some portions of the structure, a volume utilization efficiency of 50% is assumed. Therefore, 6950 m³ of contaminated material can be entombed in scenario 1 and 6665 m³ can be entombed in scenario 2. From the information presented in Section C.3 of Appendix C and summarized in Table K.1-2, an estimated volume of about 10,550 m³ of contaminated material located outside of the primary containment

TABLE K.1-2. Estimated Volumes of Contaminated Material Outside the Primary Containment Vessel

Material Category	Removal Characteristics	Estimated Volume (m ³) ^(a)
Piping and Valves	Piping cut into 2.1-m (max.) lengths. Large-bore valves removed whole.	3 981 ^(b)
Small Equipment	Unsectioned, removed whole.	518 ^(c)
Large Equipment ^(d)	Sectioned into transportable sizes, most pieces of ~2.4-m maximum dimension.	2 122 ^(c)
Large Tanks	Sectioned into steel-plate panels of ~2.4-m maximum dimension.	26 ^(c,e)
Concrete	Rubble spalled from contaminated surfaces.	516 ^(f)
Main Turbine	Sectioned into transportable sizes.	1 406 ^(g)
Main Condenser	Sectioned into transportable sizes, most pieces of ~2.4-m maximum dimension.	1 820 ^(g)
Pool Liners ^(h)	Sectioned into pieces of ~2.4-m maximum dimension.	60 ⁽ⁱ⁾
Spent Fuel Racks ^(j)	Unsectioned, removed whole.	308 ⁽ⁱ⁾
Total		10 757

(a) Number of figures shown is for computational completeness and does not imply accuracy to the nearest m³.

(b) Calculated from data in Table C.3-11 in Appendix C, piping assumed to be of maximum diameter in selected range, for conservatism.

(c) Calculated from data in Tables C.3-1, C 3-4, and C 3-6 in Appendix C.

(d) Does not include main turbine or main condenser.

(e) Calculated by dividing total mass by the approximate specific gravity of steel (7.75).

(f) Volume from Figures D.2-1 through D 2-7 in Appendix D.

(g) Volume from Table I.3-4 in Appendix I.

(h) Does not include reactor well pool liner, which is entombed in place.

(i) Estimated from information in Appendix C and Reference 2.

(j) Includes all the spent fuel racks, even though some are removed during the planning and preparation period.

vessel requires entombment or offsite disposal. By subtracting from this total the amount that can be entombed, it is calculated that 3600 m³ and 3885 m³ of contaminated material require packaging and shipment to offsite disposal in scenarios 1 and 2, respectively. In addition, the reactor internals require offsite disposal in scenario 1.

The contaminated materials to be entombed are introduced into the primary containment vessel at four access points. The wetwell hatch (elevation 143.6 m) is used to introduce some materials during the early stages of entombment. Two of the access points, the drywell personnel and equipment hatches (elevation 152.7 m) receive the bulk of the material. (Most of this material is transferred to the wetwell through the holes in the drywell floor.) Toward the end of the entombment procedure, a limited amount of material is also introduced into the drywell from the refueling floor (elevation 185.0 m) through holes cut in the refueling bellows that surround the reactor vessel flange.

All materials to be entombed must be segmented and moved to one of these points of access to the entombment structure. Materials within the Reactor Building are moved to the appropriate floor level in either of two ways: 1) small items (e.g., small, whole equipment pieces or boxes or bins of piping sections or concrete rubble) can be transported using the elevator (equipment key number 700, Figures C.2-2 through C.2-4, Appendix C), and 2) larger items can be lifted through the series of aligned hatches (equipment key numbers 701 and 702, Figures C.2-2 and C.2-3) using the 113.6-Mg bridge crane at the top of the building. Materials in the Turbine Generator and Radwaste and Control Buildings are moved to the Reactor Building either directly through existing air-lock doors or indirectly by moving them outside to the Reactor Building railroad tunnel. (All materials moved outside must be adequately packaged to prevent the inadvertent spread of contamination.) Additional passageways could be cut between these three buildings by demolishing sections of intervening walls at convenient locations. However, any breaches in the walls of the Reactor Building must eventually be sealed if the building is to serve as a secondary barrier around the entombment structure, as assumed in this study.

Once the contaminated materials are brought to the entombment structure access points, they are placed in their best storage locations in the structure. The wetwell, which receives the bulk of the material, is filled first using the lower three access points. Filling of the drywell begins using the two hatches at the 152.7-m elevation and is completed using the holes cut in the refueling bellows. Larger items placed in the wetwell are lowered into place using the 27-Mg recirculation-pump maintenance crane in the drywell (equipment key number 705, Figure 1.2-6 of Reference 2) through one of the new holes in the drywell floor. Use of this crane requires the removal of a small portion of the steel-deck floor at the 156.3-m elevation to allow adequate working space. Larger items placed in the drywell are moved into position and stacked. Smaller items are simply dumped into the drywell through the holes in the refueling bellows to fill the spaces between the larger items.

The volume utilization efficiency could be increased (and offsite transportation and disposal costs reduced accordingly) by using any of a number of techniques, including careful selection of material placement locations, nesting of piping and other equipment, and relatively complete filling of voids in emplaced waste with smaller items such as concrete rubble. However, this increased efficiency would likely increase manpower requirements and, consequently, manpower costs and occupational radiation exposures. It is beyond the scope of this study to optimize the storage, but this should be considered during the planning of any actual entombment project.

Some of the contaminated materials in the reference BWR are not amenable to entombment. These include certain of the large equipment items (e.g., the moisture separator reheaters), the main turbine, and the fuel storage racks. (See Table K.1-2). These materials must be packaged and shipped offsite for disposal as described in Appendix I.

K.1.4 Special Tools and Equipment

Not all of the tools and equipment needed for entombment are required during the operating phase of reactor life. The more unique devices used in entombment are described in Sections G.1, G.2, and G.4 of Appendix G. The special tools and equipment required for scenario-1 entombment are the same as those

for immediate dismantlement, which are listed in Table I.1-1 of Appendix I. For scenario 2, since the reactor vessel internals remain in place, the first four items in the list are not required, but the rest of the requirements are the same.

K.1.5 Summary of Disassembly Methods

Many of the disassembly methods used for entombment are the same as those for immediate dismantlement, which are summarized in Table I.1-2 of Appendix I. However, some methods are unique to entombment; these are summarized in Table K.1-3.

TABLE K.1-3. Entombment Disassembly Data and Methods^(a)

Task	Estimated Cutting Length (m)	Nominal Material Thickness(m)	Estimated Cutting Time (hr)	Strategy and Work Conditions(b)	Methods
8 Cut Suppression Pool Downcomers and Bracing	6.0×10^2	(Schedule 80 Pipe)	4.4×10^2	(1,2)	Cut downcomer vent pipes into three equal-length segments and place in void space at bottom of reactor pedestal
15 Cut Suppression Chamber Accesses through Drywell Floor					
Concrete	$(1.6 \times 10^1 \text{ m}^2 \text{ surface area})$	6.1×10^{-1}	8.0×10^1	(2,3)	Break out concrete from hatch areas. Reinforcing steel cut with oxyacetylene torch. Debris dropped into wetwell.
Steel Beams	1.1×10^1	1.7×10^{-2}	4.0×10^1	(1,2)	Cut both ends of steel beams crossing hatch areas and drop beams into wetwell.
17 Cut Primary Containment Piping Penetrations and Seal	3.3×10^2	(Schedule 80 Pipe)	2.3×10^2	(1,2)	Piping cut off at inside wall of primary containment vessel and sufficiently far away from the wall to allow pipe segments to be removed and steel plates to be welded over penetrations.
18 Cut Drywell Bellows Access Openings	2.8×10^1	3.0×10^{-2}	1.2×10^2	(1,2,4)	Holes cut through bulkhead surrounding drywell bellows to allow dumping of material into drywell from refueling floor.

(a) Only tasks unique to entombment are listed, all tasks performed within Reactor Building, all materials are contaminated rather than neutron-activated.

(b) 1 = oxyacetylene torch, 2 = contamination control envelope, 3 = jack hammer, 4 = remote cutting.

K.2 ENTOMBMENT SCHEDULES AND MANPOWER ESTIMATES

The schedules and manpower requirements for entombment are developed in the same manner and on the same bases as those for immediate dismantlement (see Section I.2 of Appendix I). The entombment schedules are quite similar to the immediate dismantlement schedule, largely because most of the tasks outside of primary containment are identical for both modes. Detailed schedules and manpower requirements are shown only for scenario-1 entombment; scenario 2, a modification of scenario 1, is discussed in the text.

K.2.1 Scenario-1 Entombment

The overall schedule and sequence of tasks for scenario-1 entombment of the reference BWR are shown in Figure K.2-1, together with the decommissioning worker requirements. The planning and preparation tasks (see Section H.2 of Appendix H) for the assumed 2-year period preceding reactor shutdown are also shown in the schedule. As shown in the figure, scenario-1 entombment is estimated to require 46.5 months after reactor shutdown to complete and to involve about 4000 man-months of decommissioning worker effort.

Most entombment tasks are accomplished on a two-shift-per-day, 5-days-per-week basis. However, some tasks (e.g., shipment of neutron-activated components and operation of the radwaste systems) require three shifts per day, 7 days per week.

The manpower required to complete the entombment project is classified as either dedicated or non-dedicated. Manpower requirements for dedicated personnel (those assigned to specific tasks) are given by task in Table K.2-1. Non-dedicated personnel (those not assigned to specific tasks, but rather given general responsibilities on each shift, two shifts per day) include one shift engineer, one crew leader, one craft supervisor, and one senior health physics technician, and, for each building being actively decommissioned, one packing and support crew and one health physics technician.

The total staff labor requirements for scenario-1 entombment of the reference BWR are given in Table K.2-2. The requirements are given in equivalent man-years for the 2 years before and the 4 years following final reactor shutdown, including the management and support staff as well as the decommissioning workers. A total effort of about 630 man-years is estimated for completion of scenario-1, including 250 man-years for the management and support staff and nearly 380 man-years for the decommissioning workers. This estimate is based on the same manpower assumptions used for immediate dismantlement (see Section I.2.4 of Appendix I). Because of these assumptions, the decommissioning worker requirements in Table K.2-2 for the 4 years following reactor shutdown exceed those given at the bottom of Figure K.2-1. This "excess" manpower is

FIGURE K.2-1. Overall Schedule and Sequence and Decommissioning Worker Requirements for Scenario-1 Entombment Tasks

TABLE K.2-1. Manpower Requirements for Scenario-1 Entombment Tasks

Building	Task(b)	Task Duration (months)	Dedicated Manpower Requirements (man-months)(a)				
			Crew Leaders	Utility Operators	Laborers	Craftsmen	Health Physics Technicians
Reactor Building/Primary Containment							
1.	Install HEPA Filters(c)	2	2.0	---	---	30.0	---
2.	Comprehensive Radiation Survey	1	---	---	---	---	6.0
3.	Discharge and Ship Fuel	2/15	---	24.0	---	---	---
4.	Remove Dryer and Separator	2	---	8.0	---	8.0	---
5.	Drain Suppression Pool to Radwaste; Water-Jet Clean(e)	4	16.8	68.4	---	---	16.8
6.	Remove Reactor Vessel Internals	6.5	---	39.0	---	26.0	---
7.	Ship Activated Components(f)	18	75.6	230.4	---	---	75.6
8.	Cut Suppression Pool Downcomers and Bracing(g)	1	---	---	42.0	21.0	21.0
9.	Chemical Decon Residual Heat Removal, Low- and High-Pressure Core Spray Systems(e)	0.5	4.3	12.8	---	---	4.3
10.	Drain Contaminated Systems to Radwaste(e)	1.5	6.3	25.7	---	---	6.3
11.	Drain Reactor Well Pool to Radwaste; Water-Jet Clean(e)	1	4.2	17.1	---	---	4.2
12.	Clean Up, Stage, and Shield Hot Spots in Primary Containment	1	---	---	6.0	---	---
13.	Chemical Decon Reactor Water Recirculation and Cleanup Systems(e)	1	8.6	25.7	---	---	8.6
14.	Drain Dryer and Separator Pool to Radwaste; Water-Jet Clean(e)	1	4.2	17.1	---	---	4.2
15.	Cut Suppression Chamber Accesses Through Drywell Floor(g)	0.5	---	---	24.1	---	---
16.	Chemical Decon Drain Systems(e)	0.5	4.3	12.8	---	---	4.3
17.	Cut Primary Containment Piping Penetrations and Seal(g)	2	---	8.0	30.0	16.0	---
18.	Cut Drywell Bellows Access Openings	0.5	---	---	4.0	---	---
19.	Remove Reactor Building Piping	10.5	---	147.0	126.0	84.0	---
20.	Drain Spent Fuel Pool to Radwaste; Water-Jet Clean(e)	2	8.4	34.2	---	---	8.4
21.	Chemical Decon Fuel Pool Cooling and Cleanup System(e)	0.5	2.1	4.3	---	---	2.1
22.	Remove Liners from Spent Fuel Pool and Dryer and Separator Pool	1	---	---	8.0	8.0	---
23.	Remove Reactor Building Equipment	3	---	6.0	30.0	6.0	---
24.	Seal Equipment and Personnel Hatch Openings into Primary Containment	1	---	---	4.0	4.0	---
25.	Remove Reactor Building Contaminated Concrete	2.5	---	---	20.0	---	---
26.	Seal Rail Tunnel, Steam Tunnel, and Biological Shield Penetrations	1	---	---	6.0	2.0	---
27.	Seal Drywell Top Head and Reactor Building External Doors	0.5	---	---	2.0	1.0	---
28.	Remove HVAC and Disable Crane	2.5	---	---	10.0	10.0	---
29.	Final Radiation Survey	1	---	---	---	---	2.0
30.	Install Security and Surveillance Monitoring Equipment; Disconnect Unnecessary Utilities	1	---	---	---	8.0	---
Turbine Generator Building							
1.	Install HEPA Filters(c)	5.5	5.5	---	---	82.5	---
2.	Comprehensive Radiation Survey(h)	1	---	---	---	---	2.0
3.	Clean Up and Stage	1	---	---	4.0	---	---
4.	Drain Contaminated Systems to Radwaste; Water-Jet Clean Condensate Storage Tanks(e)	3	12.6	51.3	---	---	12.6
5.	Remove Turbine	5	---	---	20.0	10.0	---
6.	Remove Piping	9	---	---	36.0	108.0	---
7.	Remove Condenser	4	---	16.0	48.0	48.0	---
8.	Drain Condenser to Radwaste; Water-Jet Clean(e)	3	12.6	51.3	---	---	12.6
9.	Chemical Decon Drain Systems(e)	0.5	4.3	12.8	---	---	2.1
10.	Remove Equipment	3	---	12.0	12.0	18.0	---
11.	Remove Contaminated Concrete	1.5	---	---	12.0	---	---
12.	Remove HVAC and Electrical Systems	1	---	---	4.0	2.0	---
13.	Final Radiation Survey(h)	0.5	---	---	---	---	1.0
Radwaste and Control Building							
1.	Comprehensive Radiation Survey and Release of Support Facilities/Comprehensive Radiation Survey of Radwaste and Control Building(h)	2/1	---	---	---	---	10.0
2.	General Cleanup	0.5	---	---	2.0	2.0	---
3.	Chemical Decon Drain Systems(e)	1.5	6.3	25.7	---	6.3	6.3
4.	Chemical Decon Equipment; Water-Jet Clean(e)	3	12.6	51.3	---	12.6	12.6
5.	Install Temporary Radwaste System	0.5	---	---	1.0	1.0	---
6.	Remove Piping(g)	8	---	80.0	64.0	128.0	80.0
7.	Remove Equipment(g)	9	---	90.0	108.0	90.0	90.0
8.	Remove Contaminated Concrete	2	---	---	16.0	---	---
9.	Remove Miscellaneous Metal Structures	1	---	---	4.0	4.0	---
10.	Remove HVAC and Electrical Systems	1	---	---	8.0	8.0	---
11.	Final Radiation Survey	1	---	---	---	---	2.0

(a) Besides these dedicated personnel, non-dedicated personnel per shift (two-shift-per-day basis) include one shift engineer, one crew leader, one craft supervisor, and one senior health physics technician, as well as one packing and support crew (one utility operator and three craftsmen) and one health physics technician for each building undergoing active decommissioning.

(b) All tasks are conducted on a two-shift-per-day basis, 5 days per week, except as noted otherwise.

(c) Performed primarily before reactor shutdown, but included here for the sake of consistency.

(d) Indicates either no dedicated personnel required or sufficient non-dedicated personnel available to complete the task.

(e) Conducted on a three-shift-per-day basis, 7 days per week; includes operation of radwaste systems.

(f) Conducted on a three-shift-per-day basis, 7 days per week.

(g) Radiation exposure limitations require multiple task assignments per shift for crew members.

(h) Requires assignment of additional senior health physics technician(s).

TABLE K.2-2. Staff Labor Requirements for Scenario-1 Entombment

Position	Time Relative to Final Reactor Shutdown (year)						Total Staff Labor Required (man-years)
	-2	-1	1	2	3	4	
	Annual	Staff	Labor	Requirement	(man-years)	(a)	
Management and Support Staff							
Decommissioning Superintendent	0.3	1.0	1.0	1.0	1.0	1.2 ^(b)	5.5
Secretary	1.0	2.0	3.0	3.0	3.0	2.4 ^(b)	14.4
Clerk	0	1.0	2.0	2.0	2.0	0.9	7.9
Decommissioning Engineer	1.0	1.0	1.0	1.0	1.0	1.2 ^(b)	6.2
Assistant Decommissioning Engineer	1.0	1.0	1.0	1.0	1.0	0.7	5.7
Radioactive Shipment Specialist	0	1.0	1.0	1.0	1.0	0.7	4.7
Procurement Specialist	0.3	1.0	1.0	1.0	1.0	0.7	5.0
Tool Crib Attendant	0	0	2.0	2.0	2.0	1.4	7.4
Control Room Operator ^(c)	0	0	5.0	5.0	5.0	3.6	18.6
Security Supervisor	0	0	1.0	1.0	1.0	0.9	3.9
Security Shift Supervisor ^(d)	0	0	5.0	5.0	5.0	4.4	19.4
Security Patrolman ^(d)	0	0	39.0	28.0	13.0	11.4	91.4
Contracts and Accounting Specialist	0.3	1.0	1.0	1.0	1.0	1.2 ^(b)	5.5
Health and Safety Supervisor	1.0	1.0	1.0	1.0	1.0	1.2 ^(b)	6.2
Health Physicist	0	0.5	1.0	1.0	1.0	0.8	4.3
Protective Equipment Attendant	0	0	2.0	2.0	2.0	1.8	7.8
Industrial Safety Specialist	0.3	1.0	1.0	1.0	1.0	0.9	5.2
Quality Assurance Supervisor	0.3	1.0	1.0	1.0	1.0	1.2 ^(b)	5.5
Quality Assurance Engineer	0.5	2.0	1.0	1.0	1.0	0.9	6.4
Quality Assurance Technician	0	0.5	4.0	4.0	4.0	3.5	16.0
Consultant (Safety Review)	0.5	0.5	0.5	0.5	0.5	0.5	3.0
Subtotals, Management and Support Staff	6.5	15.5	74.5	63.5	48.5	41.5	250.0
Decommissioning Workers^(e)							
Shift Engineer	1.0	2.0	2.0	2.0	2.0	1.8	10.8
Crew Leader ^(f)	0	0.7	12.2	9.2	2.0	1.5	25.6
Utility Operator ^(g)	0	3.0	52.0	44.9	14.5	6.3	120.7
Laborer	0	0	9.4	20.6	12.7	10.0	52.7
Craft Supervisor	0	0.6	2.0	2.0	2.0	1.8	8.4
Craftsman	0	10.0	17.1	36.0	25.1	14.9	103.1
Senior Health Physics Technician	0	1.0	2.4	2.3	2.0	2.0	9.7
Health Physics Technician ^(h)	0	3.0	13.9	13.6	12.0	6.2	48.7
Subtotals, Decommissioning Workers	1.0	20.3	111.0	130.6	72.3	44.5	379.7
Totals	7.5	35.8	185.5	194.1	120.8	86.0	629.7

(a) Rounded to the next higher 0.1 man-year.

(b) Includes 4 additional months following active decommissioning in order to complete the documentation and other unspecified license and contract termination requirements; shown as part of the fourth year, even though it extends 2.5 months into the fifth year.

(c) Based on one operator per shift in the control room, three shifts per day, 7 days per week.

(d) Based on 10 CFR Part 73 and information obtained from Washington Public Power Supply System; includes both response and access-control personnel on a three-shift, 7-day-week basis.

(e) Requirements during the 4 years following reactor shutdown are based on manpower values in Figure K.2-1, unless otherwise noted; individual man-month requirements in the figure are rounded to the next higher 1.0 man-month in calculating these requirements.

(f) Based on a constant loading of 11 people through month 21 following reactor shutdown, with additional personnel added as required to meet schedule demands during that period, and diminishing thereafter as the schedule allows.

(g) Based on a constant loading of 52 people through month 21 following reactor shutdown, diminishing thereafter as the schedule allows except during months 29 through 37 when 14 trained personnel are maintained to meet the requirements during months 38 through 40.

(h) Based on a constant loading of 12 people through month 40 following reactor shutdown, with additional personnel added as required to meet schedule demands during that period.

assumed to be assigned to the numerous, small, unspecified work items that inevitably accompany a project of this magnitude.

K.2.2 Scenario-2 Entombment

The overall schedule and sequence of tasks for scenario-2 entombment is very similar to that for scenario-1 entombment, shown previously in Figure K.2-1. For scenario 2, these scenario-1 tasks are deleted from the schedule:

- remove dryer and separator
- remove reactor vessel internals
- ship activated components.

However, this does not affect the timing of the other tasks or the overall length of the project (46.5 months) because the critical path of entombment tasks is unchanged. Dedicated manpower personnel requirements by task (see Table K.2-1) are also unchanged, except for the deletion of the three aforementioned tasks.

The management and support staff requirements for scenario 2 are the same as those for scenario 1, shown previously in Table K.2-2: a total of 250 man-years of management and support staff effort is required. Decommissioning worker requirements for scenario 2 are found by subtracting the dedicated manpower requirements associated with the deleted tasks from the total requirements; thus, decommissioning worker requirements total slightly more than 340 man-years. Therefore, a total effort of just over 590 man-years is estimated for completion of scenario 2.

K.3 COSTS FOR ENTOMBMENT

The estimated costs for entombment of the reference BWR (both scenarios) are presented in this section. These costs are all adjusted to early 1978 and, for consistency, the same cost estimating bases are used throughout this report for all decommissioning modes (see Appendix M). The estimated costs for both scenario-1 and scenario-2 entombment are summarized and totaled in Table K.3-1.

TABLE K.3-1. Summary of Estimated Costs for Entombment

Cost Category	Entombment Scenario 1		Entombment Scenario 2	
	Estimated Costs (\$ millions)(a,b)	Percent of Total(c)	Estimated Costs (\$ millions)(a,b)	Percent of Total(c)
Disposal of Radioactive Materials				
Neutron-Activated Materials	2.394		0	
Contaminated Materials	1.846		1.992	
Radioactive Wastes(d)	1.469		1.469	
Total Disposal Costs	5.709	17.6	3.461	12.4
Staff Labor	18.095	55.7	16.999	60.8
Energy	3.775	11.6	3.775	13.5
Special Tools and Equipment	2.016	6.2	0.866	3.1
Miscellaneous Supplies	1.859	5.7	1.859	6.6
Specialty Contractors(e)	0.172	0.5	0.172	0.6
Nuclear Insurance	0.800	2.5	0.800	2.9
License Fees	0.039	0.1	0.039	0.1
Subtotals	32.465	100.0	27.971	100.0
Contingencies (25%)	8.116		6.993	
Totals, Entombment Costs	40.581		34.964	
Annual Continuing Care Costs	0.040		0.040	
Other Possible Costs				
Spent Fuel Shipment	3.788 ^(f)		3.788 ^(f)	
Facility Demolition and Site Restoration(g)	8.059 ^(h)		8.059	
Deep Geological Disposal of Highly Activated Materials	0.495 ^(h)		0	
Fuel Channel Disposal	0.617 ⁽ⁱ⁾		0.617 ⁽ⁱ⁾	
Subtotals	12.959		12.464	
Contingencies (25%)	3.240		3.116	
Totals, Other Possible Costs	16.199		15.580	

(a) Costs adjusted to early 1978.

(b) The number of significant figures shown is for computational completeness and does not imply accuracy to the nearest \$1000.

(c) Individually rounded to the nearest 0.1%.

(d) Includes both wet solid wastes and dry solid wastes.

(e) Includes temporary radwaste and environmental monitoring services.

(f) If required, shipment by special train costs an estimated additional \$2.451 million, maximum.

(g) Does not include demolition of the Reactor Building or the Guardhouse.

(h) Incremental cost in addition to the cost for shallow-land burial of these materials; the maximum additional cost for shipment by special train is estimated at \$1.254 million.

(i) Alternate deep geologic disposal costs an estimated \$1.047 million; shipment by special train to the deep geologic disposal facility increases this cost by an estimated maximum of \$0.456 million.

The total cost for scenario-1 entombment is estimated at about \$40.6 million, including a 25% contingency; the same cost for scenario-2 entombment is about 14% less, or about \$35.0 million total. The major contributors to the total cost of entombment are disposal of radioactive materials, staff labor, and energy (at approximately 18, 56, and 12% of the total, respectively, for scenario 1 and about 12, 61, and 13% of the total, respectively, for scenario 2). Combined costs for special tools and equipment and for miscellaneous supplies make up an additional 12% of the total for scenario 1 and 10% for

scenario 2. The remaining costs, between 3 and 4% of the total for both scenarios, are contributed by specialty contractors, nuclear insurance, and license fees.

The annual costs for continuing care of the entombment structure are estimated at about \$40,000.

The total cost for other possible entombment requirements (i.e., spent fuel shipment, facility demolition and site restoration, deep geologic disposal of highly activated materials [scenario 1 only], and fuel channel disposal) is estimated at approximately \$16.2 million for scenario 1 and \$15.6 million for scenario 2, including 25% contingency allowances. This total cost could go as high as \$21.4 million for scenario 1 and \$19.5 million for scenario 2, depending on the possible additional requirements specified in the footnotes to Table K.3-1.

The individual cost categories shown in Table K.3-1 are developed and discussed in the following subsections.

K.3.1 Costs for Disposal of Radioactive Materials

As is the case during immediate dismantlement, three distinct types of radioactive materials in the reference BWR require offsite disposal during scenario-1 entombment: 1) neutron-activated materials, 2) contaminated materials, and 3) radioactive wastes. For scenario-2 entombment, only the latter two types of material require offsite disposal because all neutron-activated materials are assumed to be entombed onsite. The total cost for disposal of these materials is about \$5.7 million for scenario 1 (about 18% of the total entombment cost) and about \$3.5 million (about 12% of the total) for scenario 2. The disposal cost includes the container, transportation, and burial costs, but does not include the direct labor costs for removing and packaging the materials. Table K.3-2 contains a summary, for both entombment scenarios, of the number of shipments, the burial volumes, and the disposal costs for the radioactive materials.

The disposal costs of the three types of radioactive materials are discussed individually in the following subsections.

TABLE K.3-2. Summary Information Regarding Offsite Disposal of Radioactive Materials from Entombment

Radioactive Material Category	Entombment Scenario 1			Entombment Scenario 2		
	Number of Shipments	Burial Volumes (m ³) (a)	Disposal Costs (\$ millions) (b)	Number of Shipments	Burial Volumes (m ³) (a)	Disposal Costs (\$ millions) (b)
Neutron-Activated Materials	283	130	2.394 ^(c)	0	0	0
Contaminated Materials	330	6 420	1.846	356 ^(d)	6 928 ^(d)	1.992 ^(d)
Radioactive Wastes ^(e)						
Wet Solid	286	814	1.073	286	814	1.073
Dry Solid	86	678	0.396	86	678	0.396
Totals	985	8 042	5.709	728	8 420	3.461

(a) Includes disposable containers where required.

(b) Adjusted to early 1978; the number of figures shown is for computational completeness and does not imply accuracy to the nearest \$1000.

(c) An additional cost of \$0.848 million is required if deep geologic disposal of highly activated materials is necessary.

(d) Estimated by multiplying values for scenario 1 by the ratio of the volumes of contaminated material requiring offsite disposal for scenario 2 vs. scenario 1 (3885 m³/3600 m³), based on availability of storage space in the entombment structure (see Section K.1.3).

(e) Values assumed to be the same as for immediate dismantlement.

K.3.1.1 Neutron-Activated Materials

All of the neutron-activated materials in the reference BWR are contained in the reactor vessel, the vessel internals, and the sacrificial shield (see Section E.1 of Appendix E), and are located inside the primary containment vessel. However, the reactor vessel and the sacrificial shield are assumed to be entombed in place and, therefore, only offsite disposal of the vessel internals is considered for scenario-1 entombment. (As stated previously, no offsite disposal of activated materials is assumed for scenario-2 entombment.) Table K.3-3 gives a detailed breakdown of the masses, the radioactivity contents, the number of pieces (after segmenting), the number of packaging containers, and the disposal costs for the offsite disposal of neutron-activated components (except the fuel channels) in the reference BWR.

The total radioactivity expected to be present in the neutron-activated materials disposed of offsite is approximately 6.6 million curies. The packaged materials are estimated to require 283 overweight truck shipments to a shallow-land burial facility and to occupy 130 m³ of space at the facility. The total estimated cost for shallow-land disposal of the neutron-activated materials is about \$2.4 million.

There is a possibility that the more highly activated materials may require deep geologic disposal. This alternative is discussed later in Section K.3.10.3.

TABLE K.3-3. Costs for Disposal of Neutron-Activated Materials (Scenario-1 Entombment)

Component ^(a)	Estimated Mass (kg) ^(b)	Estimated Radioactivity Content (Ci)	No. of Pieces	No. of Containers ^(c)	Container Costs (\$) ^(d)	Cask Rental Costs (\$) ^(e)	No. of Shipments	Trans- portation Costs (\$) ^(f)	Handling Sur- charges (\$) ^(g)	Burial Volumes (m ³) ^(h)	Burial Costs (\$) ⁽ⁱ⁾	Liner Sur- charges (\$) ^(j)	Curie Sur- charges (\$) ^(k)	Total Costs (\$) ^(m)
Steam Separator Assembly	9 500 ⁽ⁿ⁾	9 600	951	28p	12 600	14 000	14	26 880	5 600	10	940	36 400 ^(o)	7 320	103 740
Orificed Fuel Supports	5 420	700	740	14p	6 300	7 000	7	13 440	2 800	5	470	7 700	0	37 710
Control Rods and In-Core Instruments	19 810 ⁽ⁿ⁾	189 000	610	8p ^(p)	48 000	27 600	8	15 360	12 310 ^(o)	15	1 400	42 810 ^(o)	11 780	159 260
Control Rod Guide Tubes	3 500 ⁽ⁿ⁾	100	370	12p	5 400	6 000	6	11 520	2 400	4	380	3 900	0	29 600
Jet Pump Assemblies	6 000 ⁽ⁿ⁾	20 000	110	40p	38 400 ^(q)	20 000	20	38 400	8 000	14	1 310	85 000 ^(o)	6 700	197 810
Top Fuel Guide	2 300	30 100	408	72p	111 600 ^(q)	36 000	72	138 240	14 400	24	2 250	153 000 ^(o)	22 030	477 520
Core Support Plate	18 500	650	258	31p	13 950	15 500	16	30 720	6 200	11	1 030	10 080	0	77 480
Core Shroud	32 000	6 300 000	558	13v + 127c	354 470 ^(q)	70 000	140	268 800	28 000	47	4 400	228 850	356 510	1 311 030
Totals ^(m)	97 030	6 550 150	4 005	345	590 720	196 100	283	543 360	79 710	130	12 180	567 740	404 340	2 394 150

- (a) Listed in order of removal.
- (b) Estimated from values presented in Table E.1-6 of Appendix E.
- (c) Estimated on the following bases: p = packaging arrangement; v = volume at 60% packing fraction; c = curie limit of 50,000 Ci per container.
- (d) Based on information in Section M.2 of Appendix M.
- (e) Based on Table M.3-1; assumes 5 days per shipment.
- (f) Based on Table M.4-4.
- (g) Based on cask handling fee in Table M.5-1, except as otherwise noted.
- (h) Includes the disposable container; rounded to the nearest whole m³.
- (i) Based on Table M.5-1; based on volume only; rounded to nearest \$10.
- (j) Based on Table M.5-1 except as otherwise noted; based on dose rate at the container surface.
- (k) Based on Table M.5-1 and a per-shipment curie inventory.
- (l) The number of figures shown is for computational completeness only and does not imply accuracy to that many significant figures.
- (m) Includes only the neutron-activated portion of the component, rest disposed of as contaminated material.
- (n) Based on cost estimate information supplied by Nuclear Engineering Co., Louisville, KY (minus 30% for escalation since September 1977).
- (p) Specially constructed steel boxes.
- (q) Includes costs for lead shielding.

TABLE K.3-3. Costs for Disposal of Neutron-Activated Materials (Scenario-1 Entombment)

The fuel channels are assumed to be shipped together with the spent reactor fuel to an offsite repository. However, in the event that this is not possible, a presentation of the disposal costs for these components is contained in Section K.3.10.4.

K.3.1.2 Contaminated Materials

Contaminated materials in the reference BWR are assumed to include much of the piping and equipment located in the Reactor Building/Primary Containment, the Turbine Generator Building, and the Radwaste and Control Building. In addition, many concrete surfaces in these three buildings are assumed to be contaminated and require surface removal to a depth of 0.051 m. However, much of this contaminated material can be placed in the entombment structure and entombed onsite, thus significantly reducing the amount of this material requiring offsite disposal.

The cost calculations for offsite disposal of contaminated materials during entombment are based on the analysis for immediate dismantlement presented in Section I.3.1.2 of Appendix I. However, only some of the materials shown in Table I.3-4 in Appendix I require offsite disposal during entombment. The amount of burial volume required for all contaminated materials outside of Primary Containment (from Table I.3-4 in Appendix I) is 13,285 m³. The actual volume of these contaminated materials is estimated in Table K.1-2 as 10,757 m³. Using these figures, the ratio of burial volume versus actual volume is estimated as 1.24. The actual volume requiring offsite disposal for scenario 1 is calculated at 3600 m³ in Section K.1.3. Multiplying this actual volume by the aforementioned volume ratio, the minimum burial volume of contaminated materials requiring offsite disposal for scenario-1 entombment is estimated to be 4446 m³.

Table K.3-4 presents breakdowns of disposal costs for contaminated materials during scenario-1 entombment. Only components assumed to be shipped offsite are shown in the table; these components are selected on the basis of size, difficulty of placing in the entombment structure, and entombment schedule. In an actual entombment project, such a material selection would probably be based on a more detailed breakdown of material categories. However, the overall cost of contaminated material disposal should not be significantly different than the cost calculated here.

TABLE K.3-4. Costs for Disposal of Contaminated Materials Assumed to be Shipped Offsite (Scenario-1 Entombment)^(a)

Location	Component ^(b)	Estimated Mass (kg) ^(c)	Number of Disposable Containers ^(d)	Container Costs (\$) ^(e)	Number of Shipments ^(f)	Transport Costs (\$) ^(g)	Handling Costs (\$) ^(h)	Burial Volumes (m ³) ⁽ⁱ⁾	Burial Costs (\$) ^(j)	Total Disposal Costs (\$) ^(k)
Primary Containment										
Steam Dryer		38 560	11	4 400	3	5 760	0	40	3 750	13 910
Steam Separator Assembly ^(m)		42 000	26	10 400	3	5 760	0	95	8 890	25 050
Miscellaneous Reactor Vessel Internals ^(m)		69 600	26	10 400	4	7 680	0	95	8 890	26 970
Control Rod Drive Equipment ^(m)		100 960	44	17 600	8	15 360	0	161	15 070	48 030
Subtotals, Primary Containment ^(k)		251 120	107	42 800	18	34 560	0	391	36 600	113 960
Reactor Building										
Equipment										
MSIV Leakage Control		410	-- ⁽ⁿ⁾	--	--	--	--	--	--	--
Residual Heat Removal		82 250	0 + 3 ^(o) + 4 ^(p)	3 450	5	9 600	1 490	62	5 810	20 350
Subtotals, Reactor Building ^(k)		82 660	0 + 7 ^(o,p)	3 450	5	9 600	1 490	62	5 810	20 350
Turbine Generator Building										
Turbine		1 662 780	220 + 58 ^(o)	156 550	82	157 440	14 480	1 406	131 560	460 030
Condenser		1 570 000	500	200 000	125	240 000	0	1 820	170 300	610 300
Equipment										
Low-Pressure Feedwater Heaters		738 000	2 + 42 ^(p)	7 100	42	80 640	15 270	737	68 970	171 980
Condensate Off Gas Treatment		2 080	0 + 4 ^(p)	300	0 ^(q)	290 ^(q)	0	8	750	1 340
Steam Evaporators/Turbine Bypass Valve Assembly		32 220	0 + 3 ^(p)	600	2	3 840	560	71	6 650	11 650
Moisture Separator Reheaters		417 310	0 + 26 ^(p)	32 000	26	62 920 ^(r)	9 360	715	66 900	171 180
Reactor Feedwater Pumps		109 770	10 + 10 ^(o)	16 520	6	11 520	1 200	194	18 250	47 490
High-Pressure Feedwater Heaters		146 950	0 + 8 ^(p)	3 600	8	15 360	3 000	121	11 320	33 280
Subtotals, Turbine Generator Building ^(k)		4 679 110	732 + 151 ^(o,p)	416 670	291	572 010	43 870	5 072	474 700	1 507 250
Radwaste and Control Building										
Miscellaneous Metal Structures ^(s)		19 440	11	4 400	1	1 920	0	40	3 750	10 070
HVAC Equipment ^(t)		69 510	208	83 200	12	14 160 ^(u)	0	757	70 840	168 200
Electrical Equipment ^(v)		63 210	27	10 800	3	5 760	0	98	9 170	25 730
Subtotals, Radwaste and Control Building ^(k)		152 160	246	98 400	16	21 840	0	895	83 760	204 000
Totals ^(k)		5 165 050	1 085 + 158 ^(o,p)	561 320	330	638 010	45 360	6 420	600 870	1 845 560

(a) Adapted from Table I.3-4 in Appendix I; only materials requiring offsite disposal are listed.

(b) Listed in order of removal (i.e., main headings only) from each location.

(c) Obtained from Appendix C, or estimated from information in Appendix C or from data supplied by Washington Public Power Supply System.

(d) Assumed to be 1.2-m by 1.2-m by 2.4-m plywood boxes, unless otherwise indicated.

(e) Based on information in Section M.2 of Appendix M.

(f) Assumed to be overweight shipments, unless otherwise indicated.

(g) Based on Table M.4-4.

(h) Based on Table M.5-1.

(i) Includes the disposal container, where required; rounded to the next whole m³.

(j) Based on Table M.5-1; based on an assumed container surface dose rate of <0.20 R/hr; rounded to the next highest \$10.

(k) The number of figures shown is for computational completeness and does not imply accuracy to that many significant figures.

(m) Includes only the contaminated portion of the component, rest shipped as neutron-activated material (see Table K.3-3).

(n) Indicates equipment packaged and shipped with other equipment.

(o) Specially fabricated plywood boxes.

(p) Equipment (or segments of equipment) on which external openings or surfaces are capped or covered and seal-welded to form self-contained packages.

(q) Combined with other shipments.

(r) Overweight-over-size shipments.

(s) Includes steel stairs, gratings, and ladders assumed to be contaminated.

(t) Includes all exhaust equipment.

(u) Legal-weight shipments.

(v) Assumed to be 5% of the total cable trays and hangers.

TABLE K.3-4. Costs for Disposal of Contaminated Materials Assumed to be Shipped Offsite (Scenario-1 Entombment)^(a)

The materials shipped offsite for disposal are estimated to require 330 truck shipments to and 6,420 m³ of burial space at a shallow-land burial site. (The burial volume is larger than the minimum burial volume calculated above, mainly because of the many large components not amenable to placement in the entombment structure.) The total disposal cost for contaminated materials during scenario-1 entombment of the reference BWR is about \$1.8 million.

For scenario-2 entombment, because the reactor vessel internals are not removed and shipped offsite, less space is available in the entombment structure. In Section K.1.3, it is calculated that 3885 m³ (actual volume) of contaminated material require offsite disposal for scenario 2 as opposed to the 3600 m³ for scenario 1. Multiplying by the ratio of these volumes, the contaminated material disposal requirements for scenario 1 are adjusted to estimate the requirements for scenario 2. The material disposed of offsite during scenario 2 entombment is thus estimated to require 356 truck shipments and to occupy 6,928 m³ of burial volume; the total estimated disposal cost is about \$2.0 million.

K.3.1.3 Radioactive Wastes

Radioactive wastes (radwastes) result directly from entombment of the reference BWR. The disposal requirements for these wastes during entombment (either scenario) are assumed to be the same as during immediate dismantlement. (See Section I.3.1.3 of Appendix I for details of the analysis.)

Wet solid wastes are estimated to require 286 truck shipments, 816 m³ of burial space, and about \$1.1 million for disposal. Disposal of dry solid wastes is estimated to cost about \$400,000 and to require 86 overweight truck shipments and 678 m³ of shallow-land burial space. The total disposal cost for radioactive wastes, the sum of the costs for the wet solid wastes and the dry solid wastes, is about \$1.5 million.

K.3.2 Costs for Staff Labor

The costs for staff labor during scenario-1 entombment are shown in detail in Table K.3-5. Approximately 56% of the total cost for scenario 1 is associated with the staff requirements (presented in Table K.2-2). A total staff labor cost of about \$18.1 million is estimated for scenario-1 entombment of the reference BWR. Specialty contractor labor is not included in this total.

TABLE K.3-5. Costs for Staff Labor During Entombment (Scenario 1)

Position	Time Relative to Final Reactor Shutdown (year)						Total Staff Labor Costs(a) (\$ thousands)
	-2	-1	1	2	3	4	
	Annual Staff Labor Costs (\$ thousands)(a,b)						
Management and Support Staff							
Decommissioning Superintendent	22.4	74.5	74.5	74.5	74.5	89.4 ^(c)	409.8
Secretary	20.3	40.6	60.9	60.9	60.9	48.8 ^(c)	292.4
Clerk	0	20.3	40.6	40.6	40.6	18.3	160.4
Decommissioning Engineer	63.5	63.5	63.5	63.5	63.5	76.2 ^(c)	393.7
Assistant Decommissioning Engineer	43.8	43.8	43.8	43.8	43.8	30.7	249.7
Radioactive Shipment Specialist	0	32.9	32.9	32.9	32.9	23.1	154.7
Procurement Specialist	9.3	30.7	30.7	30.7	30.7	21.5	153.6
Tool Crib Attendant	0	0	46.4	46.4	46.4	32.5	171.7
Control Room Operator	0	0	145.0	145.0	145.0	104.4	539.4
Security Supervisor	0	0	32.9	32.9	32.9	29.7	128.4
Security Shift Supervisor	0	0	153.5	153.5	153.5	135.1	595.6
Security Patrolman	0	0	830.7	596.4	276.9	242.9 ^(c)	1 946.9
Contracts and Accounting Specialist	11.6	38.4	38.4	38.4	38.4	46.1 ^(c)	211.3
Health and Safety Supervisor	50.4	50.4	50.4	50.4	50.4	60.5 ^(c)	312.5
Health Physicist	0	19.7	39.4	39.4	39.4	31.6	169.5
Protective Equipment Attendant	0	0	46.4	46.4	46.4	41.8	181.0
Industrial Safety Specialist	13.2	43.8	43.8	43.8	43.8	39.5	227.9
Quality Assurance Supervisor	13.2	43.8	43.8	43.8	43.8	52.6 ^(c)	241.0
Quality Assurance Engineer	19.7	78.8	39.4	39.4	39.4	35.5	252.2
Quality Assurance Technician	0	11.6	92.8	92.8	92.8	81.2	371.2
Consultant (Safety Review)	33.5	33.5	33.5	33.5	33.5	33.5	201.0
Subtotals, Management and Support Staff	300.9	626.3	1 983.3	1 749.0	1 429.5	1 274.9	7 363.9
Decommissioning Workers							
Shift Engineer	43.8	87.6	87.6	87.6	87.6	78.9	473.1
Crew Leader	0	26.2	455.1	343.2	74.6	56.0	955.1
Utility Operator	0	81.3	1 409.2	1 216.8	393.0	170.8	3 271.1
Laborer	0	0	243.5	533.6	329.0	259.0	1 365.1
Craft Supervisor	0	23.7	78.8	78.8	78.8	71.0	331.1
Craftsman	0	271.0	463.5	975.6	680.3	403.8	2 794.2
Senior Health Physics Technician	0	32.9	79.0	75.7	65.8	65.8	319.2
Health Physics Technician	0	75.3	348.9	341.4	301.2	155.7	1 222.5
Subtotals, Decommissioning Workers	43.8	598.0	3 165.6	3 652.7	2 010.3	1 261.0	10 731.4
Totals	344.7	1 224.3	5 148.9	5 401.7	3 439.8	2 535.9	18 095.3

(a)The number of significant figures shown is for computational completeness and does not imply accuracy to the nearest \$100.

(b)Calculated as the product of the data given in Table K.2-2 and the corresponding data given in Table M.1-1; rounded to the next higher \$100.

(c)Includes extra time, extending into the fifth year, allotted for completion of documentation and other unspecified license and contract termination requirements.

The costs for staff labor during scenario-2 entombment are arrived at by subtracting out the costs for dedicated manpower applied to the tasks deleted from scenario 1. The deleted tasks are: 4. Remove Dryer and Separator, 6. Remove Reactor Vessel Internals, and 7. Ship Activated Components. The manpower requirements given previously in Table K.2-1 for these tasks are multiplied by the corresponding data in Table M.1-1 in Appendix M, the products are summed, and the resulting total is subtracted from the total in Table K.3-5. Thus, a total staff labor cost of about \$17.0 million is estimated for scenario 2; this represents about 61% of the total cost for scenario 2.

K.3.3 Costs for Energy

The costs for energy during entombment (either scenario) are presented in Table K.3-6, together with the estimated usage of both electricity and fuel oil. The usage of both energy forms is estimated by detailed analysis of the entombment tasks and schedule and the requirements for the essential systems presented in Table H.4-1 in Appendix H.

A total of 123,470 MWh of electricity, costing \$1,852,050, and 14,570 m³ of fuel oil, costing \$1,923,290, are estimated to be used during entombment. The total cost for energy, about \$3.8 million, is about 12 and 14% of the total entombment cost for the two scenarios, respectively.

K.3.4 Costs for Special Tools and Equipment

For scenario-1 entombment, the costs for special tools and equipment are assumed to be the same as those presented in Table I.3-9 in Appendix I for immediate dismantlement. Thus, for scenario 1, the estimated total cost for special tools and equipment is about \$2.0 million, which represents about 6% of the total cost. For scenario 2, the first four items listed in Table I.3-9 are unnecessary. Therefore, the reduced cost for scenario 2 is about \$870,000, which is about 3% of the total cost.

K.3.5 Costs for Miscellaneous Supplies

A variety of supplies are used during entombment. The requirements and estimated costs for these items are assumed to be the same as those given in

TABLE K.3-6. Costs for Energy During Entombment

Energy Form System(s) or Equipment	Estimated Energy Usage ^(a)	Costs (\$) ^(b)
Electricity	(MWh)	
HVAC		
Main Buildings ^(c)	63 820	957 300
Control Tower	1 240	18 600
Other Buildings	7 850	117 750
Lighting	11 120	166 800
Communications	410	6 150
Security	2 720	40 800
Fire Protection	140	2 100
Compressed Air	2 550	38 250
Cranes and Hoists	1 150	17 250
Domestic Water	16 810	252 150
Contaminated Water ^(d)	9 780	146 700
Decommissioning Equipment	5 880 ^(e)	88 200
Subtotals, Electricity	123 470	1 852 050
Fuel Oil	(m³)	
Building Heating	14 375 ^(f)	1 897 500
Decontamination Solution Concentrator	195 ^(g)	25 790 ^(g)
Subtotals, Fuel Oil	14 570	1 923 290
Total Energy Cost		3 775 340

(a)Based on information presented in Table I.3-8 in Appendix I, with values adjusted as necessary to account for differences between entombment and immediate dismantlement; rounded to the nearest 10 MWh for electricity and to the nearest whole m³ for fuel oil.

(b)Based on Section M.6 of Appendix M.

(c)Includes the Reactor Building, the Turbine Generator Building, and the radwaste area of the Radwaste and Control Building.

(d)Includes all such systems listed in Appendix C for all anticipated operations; includes appropriate information from Table H.5-9 in Appendix H.

(e)Estimated at approximately 5% of the total.

(f)Assumes four heating periods during entombment.

(g)From Table H.5-9.

Table I.3-10 in Appendix I for immediate dismantlement. The total estimated cost for miscellaneous supplies during entombment of the reference BWR (either scenario 1 or scenario 2) is about \$2.2 million, which is about 6 and 7% of the total entombment cost for the two scenarios, respectively.

K.3.6 Costs for Specialty Contractors

The estimated costs for specialty contractors are given in Table K.3-7. As discussed in Section H.3 of Appendix H, these specialty contractors perform temporary radwaste handling and environmental monitoring during entombment. The costs for a hauling contractor for disposal of radioactive materials are not shown in this table, but are included in Section K.3.1 as "transportation costs."

TABLE K.3-7. Costs for Specialty Contractors During Entombment

<u>Specialty Contractor^(a)</u>	<u>Cost Estimating Basis (\$/unit time)</u>	<u>Estimated Costs (\$)^(b)</u>
Temporary Radwaste Handling	10 000/mo ^(c)	22 500
<u>Environmental Monitoring</u>	38 500/yr ^(d)	<u>149 200</u>
Total Cost		171 700

(a) Does not include the hauling contractor, costs for which are shown as "transportation costs" in Section K.3.1, at a total of about \$1.9 million.

(b) Rounded to the next highest \$100.

(c) Based on information shown on page 10-4 of Reference 3 and corroborated with Mr. Ed Brooks of Chem-Nuclear Systems, Inc., Bellevue, Washington.

(d) Based on information presented on page G-37 of Reference 3.

The total cost for specialty contractors during entombment, excluding the hauling contractor, is about \$172,000, which is less than 1% of the total cost for either entombment scenario.

K.3.7 Costs for Nuclear Insurance

The costs for nuclear liability insurance during entombment are assumed to be the same as those estimated for immediate dismantlement, which are presented in Section I.3.7 of Appendix I. The total estimated cost for nuclear insurance is \$800,000 and represents less than 3% of the total cost for either entombment scenario.

K.3.8 Costs of Licensing Fees

The fees charged for licensing services performed by the NRC are delineated in 10 CFR Part 170.⁽⁴⁾ The costs of licensing fees during entombment of the reference BWR are shown in Table K.3-8. The total cost for licensing services is \$38,500, which is approximately 0.1% of the total cost for either entombment scenario.

TABLE K.3-8. Costs of Licensing Fees During Entombment^(a)

Licensing Service Category	Fee (\$)	Total Costs (\$)
Facility License Amendment (Class IV)	12 300	12 300 ^(b)
Health, Safety, and Environmental Inspection	650/year	2 600 ^(c)
Safeguards Inspection	11 800/year	23 600 ^(d)
Total		38 500

(a) From Reference 4.

(b) Based on one license amendment to obtain a possession-only license prior to entombment.

(c) Based on the time the license is in force (i.e., approximately 4 years for entombment).

(d) Based on having spent fuel on site for 19 months (i.e., two yearly fees charged).

K.3.9 Costs for Continuing Care and Possible Deferred Dismantlement

As discussed in Section K.1, existing regulations require the licensee to continue surveillance and maintenance of the entombment structure as a nuclear waste repository until such time as the structure is released for unrestricted use. Furthermore, a comprehensive radiation survey is required prior to such release to verify that the radioactive contamination either meets acceptable release limits or is removed from the site.

The costs for continuing care (i.e., maintenance and surveillance) of the entombment structure are assumed to be less than those for continuing care of the reference BWR in passive safe storage (see Section J.5.2. of Appendix J). Based on information presented in Reference 5, the annual continuing care costs are estimated to be about \$40,000.

While selection of entombment entails a commitment of the site for as long as required for the entombed radioactivity to decay to unrestricted release levels, there may be instances where earlier release of the site becomes imperative and dismantlement of the entombed structure is necessary. Because of the many variables involved, no firm estimate of the costs for possible deferred dismantlement of the entombment structure is made. However, these

costs are anticipated to be at least of the same order of magnitude as those for deferred dismantlement of the reference BWR after a period of passive safe storage (see Section J.7 of Appendix J). Although deferred dismantlement of the entombment structure involves only a portion of the original plant, the dismantlement of that portion is made more difficult and, consequently, more expensive by the possibility of having to penetrate the strong, durable entombment structure and remove the (more or less) randomly mixed materials inside for a radiation survey. In addition, the materials with radioactive contamination exceeding release limits will require packaging and shipping to an offsite disposal facility.

K.3.10 Other Possible Costs

Four additional categories of costs could figure into the total entombment cost, depending on how they are classified. In this study, they are presented separately, since they cannot be clearly identified as belonging to entombment. The tasks that result in these costs are:

- shipment of the spent reactor fuel to an offsite repository
- demolition of the structures other than the Reactor Building and restoration of the site
- alternative disposal of the highly activated materials in a deep geologic disposal facility
- disposal of the fuel channels.

Discussions of the costs associated with these tasks are presented in the following subsections.

K.3.10.1 Spent Fuel Shipment

Since the ultimate disposition of the spent reactor fuel is not known, it is assumed in this study that the 764 fuel assemblies from the final reactor core load are shipped by rail, together with their fuel channels, to a repository located 2400 km from the reference BWR. The resulting costs for entombment are the same as those calculated for immediate dismantlement in Section I.3.9.1 of Appendix I.

The total estimated cost for shipping the spent fuel and channels to the assumed repository is about \$3.8 million. If special trains are required, the cost is increased by up to about \$2.5 million.

K.3.10.2 Facility Demolition and Site Restoration

The costs for demolishing the decontaminated and uncontaminated structures of the reference BWR are summarized in Table L.3-1 of Appendix L and are not repeated here. The total cost for demolishing all of the site structures is estimated at \$13,244,000 (without contingency), including labor, supplies, overheads, and profit, but not including extraordinary insurance premiums, bonding, or state sales tax. However, for entombment, the entire Reactor Building is assumed to be left standing as a secondary barrier around the entombed materials (see Section K.1.1). In addition, the Guardhouse is assumed to remain for various possible uses during continuing care. Costs for reducing the size of the exclusion area (relocating fence lines) are assumed to be offset by reduced site restoration costs due to the reduced area requiring restoration. By subtracting the costs for demolishing the Reactor Building and the Guardhouse from the total, the cost for demolition and site restoration after entombment is estimated to be \$8,059,000. Details of the cost estimates for this task, which is not required by current regulations, are given in Section L.3 of Appendix L.

K.3.10.3 Deep Geologic Disposal of Highly Activated Materials

The information presented in Table K.3-3 is based on the assumption that all of the neutron-activated components are placed in a shallow-land disposal site. Because of the large initial amount of radioactivity, the long half-lives of ^{59}Ni and ^{94}Nb , and the potential biological hazard of ^{14}C , it is possible that the highly activated materials may require disposal in a deep geologic disposal facility, similar to the proposed disposal of high-level radioactive wastes and fuel cladding hulls that result from spent fuel reprocessing. An analysis is therefore made to estimate the additional cost of placing the highly activated materials in deep geologic storage rather than in shallow-land storage.

The materials possibly requiring deep geologic disposal for scenario-1 entombment are the same as those for immediate dismantlement. From the analysis presented in Section I.3.9.3 of Appendix I, the total cost of placing these materials into deep geologic disposal is \$2,889,390. Subtracting the total cost for shallow-land disposal of these materials (\$2,394,150, from Table K.3-3), the incremental cost for disposing of these materials in deep geologic disposal is \$495,240 more than the shallow-land burial.

If a special train is required, an additional total estimated cost of \$1,254,000 is necessary for 22 single-cask shipments (see Section M.4.2). The use of more than one cask per shipment would reduce this cost inversely to the number of casks per shipment, but would not affect the disposal costs presented previously.

K.3.10.4 Fuel Channel Disposal

The assumption that the fuel channels are shipped and stored with the spent fuel is based on practicality rather than present practice. Presently, this is not a viable option and, therefore, an analysis is performed (in Section I.3.9.4 of Appendix I) to determine the alternative costs for disposing of the fuel channels as activated waste in either a shallow-land disposal facility or a deep geologic disposal facility.

The total cost for shallow-land burial of the fuel channels is estimated to be \$617,140. Deep geologic disposal is estimated to cost \$1,047,000, or \$429,860 more than shallow-land disposal. Shipment of the fuel channels by special train would cost an additional \$456,000, maximum.

K.4 EXTERNAL OCCUPATIONAL RADIATION DOSE FOR ENTOMBMENT

Detailed analysis is made of the external occupational radiation dose accumulated by the decommissioning workers during scenario-1 entombment of the reference BWR. The analysis is carried out in the same manner as that described for immediate dismantlement (see Section I.4 of Appendix I). The results of this analysis are presented in Table K.4-1.

TABLE K.4-1. Estimated Occupational Radiation Doses for Scenario-1 Entombment

Location	Task	Supervisors (a)		Utility Operators and Laborers		Craftsmen		Health Physics Technicians		Task Totals			Corrected Dose (c)
		Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Decay Factor (b)	
Reactor Building/Primary Containment													
1	Install HEPA Filters	400	0 400	--- (d)	---	2 712	2 712	270	0 270	3 382	3 382	1 000	3 382
2	Comprehensive Radiation Survey	---	---	---	---	---	---	100	0 500	100	0 500	0 995	0 498
3	Discharge and Ship Fuel	941	8 469	6 268	56 412	---	---	2 752	24 768	9 961	89 649	0 904	81 043
*4	Remove Dryer and Separator	130	0 650	865	4 325	545	2 725	87	0 435	1 627	8 135	0 989	8 046
5	Drain Suppression Pool to Radwaste, Water-Jet Clean(e)	27	0 189	322	1 932	27	0 135	27	0 135	403	2 391	0 978	2 338
*6	Remove Reactor Vessel Internals	735	3 675	3 293	16 465	1 605	8 025	490	2 450	6 123	30 615	0 944	28 903
*7	Ship Activated Reactor Vessel Internals Segments	1 865	9 325	12 432	62 160	---	---	1 250	6 250	15 547	77 735	0 896	69 651
8	Cut Suppression Pool Downcomers and Bracing	79	3 160	1 056	42 240	528	21 120	106	4 240	1 769	70 760	0 947	67 010
9	Chemical Decon Residual Heat Removal, Low- and High-Pressure Core Spray Systems(e)	12	0 360	259	4 403	20	1 100	118	3 540	409	9 403	0 954	8 970
10	Drain Contaminated Systems to Radwaste(e)	---	---	12	0 180	4	0 060	2	0 030	18	0 270	0 949	0 256
11	Drain Reactor Well Pool to Radwaste, Water-Jet Clean(e)	3	0 015	55	0 275	5	0 025	5	0 025	68	0 340	0 906	0 308
12	Clean Up, Stage, and Shield Hot Spots in Primary Containment	108	1 080	720	7 200	---	---	72	0 720	900	9 000	0 906	8 154
13	Chemical Decon Reactor Water Recirculation and Cleanup Systems(e)	4	0 200	572	5 148	8	0 760	36	1 872	620	7 980	0 896	7 150
14	Drain Dryer and Separator Pool to Radwaste, Water-Jet Clean(e)	9	0 045	97	0 485	8	0 040	8	0 040	122	0 610	0 896	0 547
15	Cut Suppression Chamber Accesses Through Drywell Floor	40	1 600	264	26 400	132	13 200	26	1 300	462	42 500	0 879	37 358
16	Chemical Decon Drain Systems(e)	---	---	431	0 431	108	0 108	---	---	539	0 539	0 879	0 474
17	Cut Primary Containment Piping Penetrations and Seal	119	4 760	792	47 520	792	47 520	79	3 950	1 782	103 750	0 877	90 989
18	Cut Drywell Bellows Access Openings	79	0 395	---	---	528	2 640	53	0 265	660	3 300	0 865	2 855
19	Remove Reactor Building Piping	1 370	27 400	9 150	183 000	3 786	75 720	915	18 300	15 221	304 420	0 819	249 320
20	Drain Spent Fuel Pool to Radwaste, Water-Jet Clean(e)	9	0 045	109	0 545	9	0 045	9	0 045	136	0 680	0 803	0 546
21	Chemical Decon Fuel Pool Cooling and Cleanup System(e)	2	0 010	52	0 312	2	0 020	16	0 108	74	0 450	0 792	0 356
22	Remove Liners from Spent Fuel Pool and Dryer and Separator Pool	160	0 800	1 057	5 285	925	4 625	106	0 530	2 248	11 240	0 769	8 644
23	Remove Reactor Building Equipment	2 175	4 350	14 494	28 988	1 106	2 212	1 450	2 900	19 225	38 450	0 752	28 914
24	Seal Equipment and Personnel Hatch Openings into Primary Containment	79	0 395	528	2 640	264	1 320	53	0 265	924	4 620	0 704	3 252
25	Remove Reactor Building Contaminated Concrete	1 000	3 000	6 764	20 292	-	---	680	2 040	8 444	25 332	0 658	16 668
26	Seal Rail Tunnel, Steam Tunnel, and Biological Shield Penetrations	119	0 595	792	3 960	264	1 320	79	0 395	1 254	6 270	0 645	4 044
27	Seal Drywell Top Head and Reactor Building External Doors	59	0 118	---	---	396	0 792	40	0 080	495	0 990	0 633	0 627
28	Remove HVAC and Disable Crane	396	1 188	2 640	7 920	660	1 980	264	0 792	3 960	11 880	0 622	7 389
29	Final Radiation Survey	---	---	---	---	---	---	100	0 100	100	0 100	0 611	0 061
30	Install Security and Surveillance Monitoring Equipment, Disconnect Unnecessary Utilities	158	0 158	---	---	1 056	1 056	---	---	1 214	1 214	0 604	0 733
Subtotals, Reactor Building(c,f)		10 078	72	63 024	529	15 490	189	9 195	76	97 787	867		738
Turbine Generator Building													
1	Install HEPA Filters	1 100	1 100	---	---	7 450	7 450	745	0 745	9 295	9 295	1 000	9 295
2	Comprehensive Radiation Survey	---	---	---	---	---	---	100	0 200	100	0 200	0 957	0 191
3	Clean Up and Stage	100	0 100	700	0 700	---	---	70	0 070	870	0 870	0 947	0 824
4	Drain Contaminated Systems to Radwaste, Water-Jet Clean Condensate Storage Tanks(e)	8	0 008	51	0 051	---	---	5	0 005	64	0 064	0 926	0 059
5	Remove Turbine	407	0 407	2 711	5 422	1 242	2 484	272	0 544	4 632	8 857	0 891	7 892
6	Remove Piping	2 530	15 180	2 484	14 904	14 411	86 466	1 440	8 640	20 865	125 190	0 848	106 161
7	Remove Condenser	1 651	1 792	11 011	21 566	7 755	14 538	1 100	2 154	21 517	40 050	0 848	33 962
8	Drain Condenser to Radwaste, Water-Jet Clean(e)	1	0 001	10	0 010	---	---	1	0 001	12	0 012	0 853	0 010
9	Chemical Decon Drain Systems(e)	---	---	305	0 305	76	0 076	---	---	381	0 381	0 837	0 319
10	Remove Equipment	990	4 950	4 026	20 130	2 591	12 955	660	3 300	8 267	41 335	0 816	33 729
11	Remove Contaminated Concrete	300	0 300	2 039	2 039	---	---	204	0 204	2 543	2 543	0 797	2 027
12	Remove HVAC and Electrical Systems	53	0 053	315	0 315	167	0 167	36	0 036	571	0 571	0 786	0 449
13	Final Radiation Survey	---	---	---	---	---	---	50	0 050	50	0 050	0 784	0 039
Subtotals, Turbine Generator Building(c,f)		7 140	24	23 652	65	33 692	124	4 683	16	69 167	229		195
Radwaste and Control Building													
1	Comprehensive Radiation Survey	---	---	---	---	---	---	80	0 800	80	0 800	0 989	0 791
2	General Cleanup	15	0 075	180	2 700	---	---	15	0 075	210	2 850	0 828	2 360
3	Chemical Decon Drain Systems(e)	---	---	435	0 435	109	0 109	---	---	544	0 544	0 819	0 446
4	Chemical Decon Equipment, Water-Jet Clean(e)	45	0 195	360	21 505	---	---	45	0 915	450	22 615	0 794	17 956
5	Install Temporary Radwaste System	2	0 004	16	0 032	24	0 048	2	0 004	44	0 088	0 814	0 072
6	Remove Piping	2 707	13 001	9 845	115 686	19 688	163 848	7 382	38 239	39 622	330 774	0 740	244 773
7	Remove Equipment	3 048	7 621	6 508	171 001	4 976	124 638	4 035	52 138	18 567	355 398	0 674	239 538
8	Remove Contaminated Concrete	144	0 402	2 640	8 640	690	1 950	546	1 134	4 020	12 126	0 642	7 785
9	Remove Miscellaneous Metal Structures	16	0 032	312	0 624	312	0 624	39	0 078	679	1 358	0 638	0 866
10	Remove HVAC and Electrical Systems	361	0 900	888	4 200	1 012	4 460	428	1 069	2 689	10 629	0 631	6 707
11	Final Radiation Survey	---	---	---	---	---	---	80	0 080	80	0 080	0 624	0 050
Subtotals, Radwaste and Control Building(c,f)		6 338	22	21 184	325	26 811	296	12 652	95	66 985	737		521
Ancillaries													
1	Operate Radwaste Systems(e)	---	---	286	1 430	144	21 600	36	0 495	466	23 525	0 901	21 196
2	Routine Radiation Surveys	---	---	---	---	---	---	2 300	9 377	2 300	9 377	0 779	7 305
3	Package Dry Solid Wastes(g)	---	---	2 134	46 962	66	1 600	---	---	2 200	48 562	0 784	38 073
4	Miscellaneous(h)	---	---	---	---	---	---	---	---	---	---	---	163
Subtotals, Ancillaries(c,e)		0	0	2 420	48	210	23	2 336	10	4 966	81		230
Totals(c)		23 556	118	110 280	967	76 203	632	28 866	197	238 905	1 914		1 684

(a) Includes shift engineers, crew leaders, craft supervisors, and senior health physics technicians
(b) The decay factor is based on the half-life of ⁶⁰Co and is calculated at the midpoint of the corresponding task timeline shown on Figure K 2-1
(c) The number of significant figures shown is for computational convenience and does not imply accuracy to the nearest millirem
(d) A dash indicates that, for the specified decommissioning task, that particular category of staff is not used
(e) Based on data presented in Table H 5-11 in Appendix H
(f) Dose totals are rounded to the nearest man-rem
(g) Assumed to be the same as for immediate dismantlement
(h) Consists of an allowance of up to 1 rem per year for selected management and support staff whose radiation doses are not specifically estimated, together with an allowance of 5% of the total explicitly estimated task radiation dose, to account for any omissions and uncertainties in the analyses
* These tasks deleted for entombment scenario 2

The total corrected external occupational radiation dose for scenario-1 entombment is about 1684 man-rem. For scenario-2 entombment, subtraction of the doses for the deleted tasks results in a total dose of about 1573 man-rem.

Two other tasks not shown in Table K.4-1 provide the potential for external radiation dose and occur as a direct result of entombment: 1) transporting the radioactive materials to the disposal site, and 2) handling these materials at the disposal site. Totals of 985 and 728 shipments of radioactive materials must be transported to, and handled at, the disposal site for scenarios 1 and 2, respectively. The external radiation dose from transportation is discussed in Section N.5 of Appendix N. The external radiation dose from handling these materials at the disposal site is beyond the scope of this study, since it is considered under another nuclear license.

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6. K. J. Schneider and C. E. Jenkins, Technology, Safety and Costs of Decommissioning a Reference Fuel Reprocessing Plant, NUREG-0278, p. 7-129, U.S. Nuclear Regulatory Commission Report by Pacific Northwest Laboratory, September 1977.**

* Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and the National Technical Information Service, Springfield, Virginia 22161.

** Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.

APPENDIX L

DEMOLITION AND SITE RESTORATION DETAILS

Once all radioactive materials in a boiling water reactor (BWR) are removed or decontaminated, the Nuclear Regulatory Commission (NRC) is requested to terminate the possession-only license and release the site for unrestricted use. Following license termination, the utility decides whether the remaining onsite structures are to be demolished or left standing. (Section 4.2 of Volume 1 contains a discussion of some of the considerations in making this decision.) Although NRC has no jurisdiction over removal of noncontaminated structures and restoration of the site, development of demolition and site restoration costs is presented in this appendix for completeness. To ensure realism, a demolition contractor was retained to develop the costs of demolition, just as though he were bidding on the job.⁽¹⁾

L.1 ASSUMPTIONS FOR DEVELOPMENT OF COST ESTIMATES

The analyses of the effort and costs involved in demolishing the reference BWR structures and restoring the site are based on the following assumptions:

1. All above-ground structures on the plant site are demolished and removed.
2. Building structures are demolished down to 1 m below grade; holes are drilled in the sub-basement floors for drainage; the empty below-grade volumes are filled to within 1 m of the grade level with concrete rubble; and the last meter is backfilled with native topsoil.
3. The demolition contractor has salvage rights, with a resulting fair profit going to the contractor. The total estimated cost reflects this profit where applicable.
4. Excess rubble and other debris is disposed of at a landfill 16 km from the site.

L.2 FACTORS CONSIDERED IN ESTIMATION OF DEMOLITION COSTS

Cost estimates for demolition of the reference BWR plant are influenced by several factors, including demolition methods, problem areas in cost estimation, salvage considerations, disposal of debris, labor rates, and equipment requirements. These factors are discussed in the following subsections.

L.2.1 Demolition Methods

Removal of site structures is accomplished using conventional demolition methods when possible. However, those buildings designed as Seismic Category I structures are more difficult to demolish than conventional industrial buildings.

In general, building demolition proceeds from the top down after machinery and equipment are removed. Where there is a bridge crane, it is used for moving such things as machinery, equipment, and piping.

The top structure of a steel-capped building is removed by reversing the construction procedure. A hydraulic crane and man-lift cranes, positioned on the floor below the roof, are used to remove the roofing, steel siding, and structural framework.

Heavy equipment needed for demolition of reinforced concrete buildings is hoisted to the top floor of the structure. Typical equipment includes a cable crawler crane, a crawler backhoe with a hydraulic hammer and a hydraulic shear, a track loader, and track drills.

Floor openings are cut, one above another, for dropping rubble from the upper levels to the ground level. Before being pulled down, reinforced concrete columns are weakened at the floor level either by using the hydraulic hammer or by blasting. Exterior walls are pulled in after the wall columns are "softened" and the walls are broken loose from the corners. The walls are then broken up using a wrecking ball on the cable crane. After the concrete is broken away, reinforcing steel (rebar) is cut into pieces about 1.2 m long using the hydraulic shear. (Cutting the rebar into 1.2-m lengths upgrades it to prepared scrap, thus increasing its salvage value.) Floor slabs are broken

up using the wrecking ball. Heavily reinforced, Seismic Category I concrete structures must be drilled and blasted, since a wrecking ball will probably be ineffective. Prior to blasting, a crane is used to place blasting mats and shields.

After most of a floor is demolished, heavy equipment is moved on ramps to the next lower floor. A crane is used to lower smaller equipment from floor to floor. The remaining portions of the floor are then removed using the crane and wrecking ball.

L.2.2 Problem Areas in Estimation of Demolition Costs

No precedent exists for estimating the costs of demolishing the heavily reinforced, massive concrete structures of the reference BWR. The estimates presented in this appendix result from comparisons of the reference BWR structures with industrial-type structures that have been demolished. In addition, judgment factors are applied, based on experience, for the massiveness, grade of concrete, extra-heavy reinforcing steel, and the height of the structures.

An area of concern in estimating demolition costs is the value used for balling and separating the concrete from the rebar, both with and without weakening by explosives. Concrete in the reference BWR structures is high quality, extra thick, well aged, and well bonded to extra-heavy reinforcing steel. Most of the structures also have confining and self-reinforcing cross walls that restrict access and make use of a wrecking ball difficult. Singly, these factors tend to increase demolition costs markedly, and their combination compounds the effect.

L.2.3 Salvage Considerations

Salvage is ordinarily used in the wrecking industry as a contingency/profit item to compensate for such inherent risks as unanticipated demolition difficulty, human error, and variation in employee productivity. Where salvage is easily separable from demolition (i.e., substantial mechanical or electrical equipment salvage), it is given separate consideration in estimating costs. The value of structural salvage is generally accounted for by reducing the unit demolition cost rather than by a separate itemized account. The

costs of handling, loading, and shipping salvaged scrap are deducted from the published market price for scrap steel, with the balance credited to the facility owner. This ensures that the wrecking contractor recovers his costs, including the added expense of preserving the highest salvage value, and that the owner receives a fair allowance for the scrap.

L.2.4 Transport of Demolition Debris

Loading and hauling demolition debris generally costs more than loading and hauling dirt or gravel. Demolition trucks are not loaded or dumped as fast as dirt and gravel trucks because of their longer, higher-sided trailers and the uneven sizes of the materials they haul. Hauling costs in the wrecking industry reflect these conditions.

L.2.5 Labor Rates

The cost estimates for the demolition of the reference BWR are based on the labor rates in Seattle, Washington, in January 1978. The labor rates used in this analysis are given in Table L.2-1. The payroll costs include base pay, FICA, unemployment insurance, union contributions, state industrial insurance, property damage insurance, and medical insurance. They do not include overhead or profit. Bodily injury public liability insurance, which was formerly a nominal charge based on payroll, is not included in the payroll costs but is estimated at 9.96% of gross receipts.

TABLE L.2-1. Labor Rates for Demolition

<u>Occupation</u>	<u>Base Pay (\$/hr)</u>	<u>Payroll Cost (\$/hr)</u>
Teamsters (38-m ³ trucks)	11.69	16.78
Operating Engineers - Operator	11.98	17.29
- Oiler	11.19	16.34
Laborers	9.10	14.15

L.2.6 Equipment Requirements for Demolition

Because of the variety of operations involved in the demolition of the reference BWR, a number of equipment types and sizes are required.

Table L.2-2 lists the major equipment items, together with the use and required number of each item.

TABLE L.2-2. Equipment Requirements For Demolition of the Reference BWR

<u>Item</u>	<u>Use</u>	<u>Number Required</u>
200-Ton Crane	High lifts of heavy loads	1
65-Ton Crane	Lifts and breaking ball for concrete demolition	1
45-Ton Crane	Lifts and breaking ball for concrete demolition	2
25-Ton Crane	Lifts of light loads	2
50-Ton Hydraulic Crane	Dismantle roof strucutres	2
30-Ton Hydraulic Crane	Dismantle roof structures	2
1-m ³ Backhoe	Clean up and load debris	3
977 Track Loader	Clean up and load debris	2
Demolition Trailers w/Tractors	Haul debris	10
Oxyacetylene Cutting Outfits	Cut steel and other metal	40
Clamshells	Pick up debris	2
Track Drills	Drill holes for explosives	4
Air Hammers	Demolish concrete	2
Hydraulic Hammers	Demolish concrete	4
Hydraulic Shears	Cut rebar to salvage length	3
Compressors	Power pneumatic tools	As needed
Service Truck	Service equipment	1
Breaking Balls	Demolish concrete	As needed
Battering Ram	Demolish concrete walls	1
Rigging	Move equipment and salvage	As needed

L.3 DEMOLITION AND SITE RESTORATION COSTS

The estimated costs for demolition and site restoration of the reference BWR after decommissioning by immediate dismantlement are summarized in Table L.3-1. These costs include labor, supplies, overhead, and profit. They do not include extraordinary insurance premiums, bonding, or state sales tax. Details of the cost estimates are given in the following subsections.

TABLE L.3-1. Summary of Estimated Costs for Demolition and Site Restoration of the Reference BWR^(a)

<u>Activity</u>	<u>Estimated Costs (\$ thousands)</u>
Structure Demolition	
Reactor Building and Primary Containment Vessel	5 169
Turbine Generator Building	3 364
Radwaste and Control Building	1 923
Diesel Generator Building	445
Service Building	130
Cooling Towers	1 252
Circulating Water Pumphouse	173
Spray Pond Complex	282
Makeup Water Pumphouse	43
Office Building	9
Warehouse	26
Guard House	16
Gas Bottle Storage Building	2
Site Restoration	410
Subtotal	13 244
Contingency (25%)	3 311
Total	16 555

^(a)Estimates given in January 1978 dollars.

L.3.1 Reactor Building and Primary Containment Vessel

Before the bridge crane is removed, it is used to remove any equipment and material in the way of the subsequent operations.

Since the biological shield outside the Primary Containment Vessel is integral with the Reactor Building, the two are demolished together. The top ring rebar of the shield is removed first to permit track drills to drill vertically into the shield wall. A triangular drilling pattern is used to optimize blasting efficiency and concrete breakage. After each blast, the rubble is cleaned up, and the rebar is cut up and, where possible, salvaged. A track loader is used to remove the broken concrete.

After they are broken loose from the corners, outside walls are pulled inward onto the working floor. Wall columns are "softened" at floor level with a hydraulic hammer or with small explosive charges, with the walls tied to prevent them from falling outward. Secondary breakage of the concrete is accomplished with a crane and wrecking ball.

The cost breakdown for the demolition of the Reactor Building and Primary Containment Vessel is given in Table L.3-2.

TABLE L.3-2. Estimated Demolition Costs for the Reactor Building and the Primary Containment Vessel

<u>Material/Activity</u>	<u>Amount</u>	<u>Unit</u>	<u>Estimated Unit Cost (\$/unit)</u>	<u>Estimated Costs (\$ thousands)</u>
Structural Steel/Removal	1 699	Mg	151.38	257
Reinforced Concrete/				
Drilling and Blasting	31 368	m ³	81.60	2 560
Secondary Breaking and Rubble Cleanup	31 368	m ³	48.00	1 506
Loading, Hauling, and Disposal	31 368	m ³	18.00	565
<u>Primary Containment Vessel/Removal</u>	1 854	Mg	151.38	<u>281</u>
Total				5 169

L.3.2 Turbine Generator Building

Although the Turbine Generator Building is not as tall or massive as the Reactor Building, it is considerably heavier and stronger than most industrial structures. Demolition methods used for this building are similar to those for the Reactor Building, described previously.

The cost breakdown for the demolition of the Turbine Generator Building is given in Table L.3-3.

TABLE L.3-3. Estimated Demolition Costs for the Turbine Generator Building

<u>Material/Activity</u>	<u>Amount</u>	<u>Unit</u>	<u>Estimated Unit Cost (\$/unit)</u>	<u>Estimated Costs (\$ thousands)</u>
Structural Steel/Removal	12 580	m ²	29.63	373
Reinforced Concrete Above Grade/ Drilling and Blasting	31 132	m ³	36.00	1 121
Secondary Breaking and Rubble Cleanup	31 132	m ³	36.00	1 121
Loading, Hauling, and Disposal	31 132	m ³	18.00	560
Reinforced Concrete Below Grade/ Drilling and Blasting	3 490	m ³	24.00	84
Secondary Breaking, Rubble Cleanup, Loading, and Hauling	3 490	m ³	30.00	<u>105</u>
Total				3 364

L.3.3 Radwaste and Control Building

Procedures for demolishing the Radwaste and Control Building are similar to those used for the Reactor Building. Since the closely spaced, self-reinforcing cross walls are particularly difficult to demolish with a crane and wrecking ball, blasting (using track drills to drill down into the walls) is the most effective way of removing them.

The demolition cost breakdown for this building is given in Table L.3-4.

TABLE L.3-4. Estimated Demolition Costs for the Radwaste and Control Building

<u>Material/Activity</u>	<u>Amount</u>	<u>Unit</u>	<u>Estimated Unit Cost (\$/unit)</u>	<u>Estimated Costs (\$ thousands)</u>
Structural Steel/Removal	6 170	m ²	29.63	183
Reinforced Concrete/ Drilling and Blasting	19 340	m ³	36.00	696
Secondary Breaking and Rubble Cleanup	19 340	m ³	36.00	696
<u>Loading, Hauling, and Disposal</u>	19 340	m ³	18.00	<u>348</u>
Total				1 923

L.3.4 Diesel Generator Building

The Diesel Generator Building is a Seismic Category I structure of reinforced concrete and, thus, is difficult and expensive to demolish. Its demolition cost breakdown is given in Table L.3-5.

TABLE L.3-5. Estimated Demolition Costs for the Diesel Generator Building

<u>Activity</u>	<u>Amount</u>	<u>Unit</u>	<u>Estimated Unit Cost (\$/unit)</u>	<u>Estimated Costs (\$ thousands)</u>
Drilling and Blasting	3 016	m ³	81.60	246
Secondary Breaking and Rubble Cleanup	3 016	m ³	48.00	145
<u>Loading, Hauling, and Disposal</u>	3 016	m ³	18.00	<u>54</u>
Total				945

L.3.5 Service Building

The exterior above-grade walls of the Service Building are made of pre-cast concrete panels. Below-grade construction is of reinforced concrete. Structural steel supports the second floor, and interior walls are largely of steel-studded drywall construction. Costs for Service Building demolition are given in Table L.3-6.

TABLE L.3-6. Estimated Demolition Costs for the Service Building

<u>Material</u>	<u>Amount</u>	<u>Unit</u>	<u>Estimated Unit Cost (\$/unit)</u>	<u>Estimated Costs (\$ thousands)</u>
Steel	395	Mg	50.46	20
<u>Concrete</u>	1 838	m ³	60.00	<u>110</u>
Total				130

L.3.6 Cooling Towers

The six cooling towers are concrete structures set on cast-in-place reinforced concrete basins. The estimated costs for demolishing the towers and basins are given in Table L.3-7.

TABLE L.3-7. Estimated Demolition Costs for the Cooling Towers

<u>Material</u>	<u>Amount</u>	<u>Unit</u>	<u>Estimated Unit Cost (\$/unit)</u>	<u>Estimated Costs (\$ thousands)</u>
Concrete and Cement Asbestos Board in Towers	16 818	m ³	60.00	1 009
<u>Reinforced Concrete in Basins</u>	4 050	m ³	60.00	<u>243</u>
Total				1 252

L.3.7 Circulating Water Pumphouse

This reinforced concrete building with steel superstructure is demolished by conventional means. The cost breakdown for its demolition and removal is shown in Table L.3-8.

TABLE L.3-8. Estimated Demolition Costs for the Circulating Water Pumphouse

<u>Material</u>	<u>Amount</u>	<u>Unit</u>	<u>Estimated Unit Cost (\$/unit)</u>	<u>Estimated Costs (\$ thousands)</u>
Steel Superstructure	112	Mg	151.38	17
<u>Reinforced Concrete</u>	2 593	m ³	60.00	<u>156</u>
Total				173

L.3.8 Spray Pond Complex

The two spray ponds and two Standby Service Water Pumphouses are Seismic Category I structures. A crane with wrecking ball is used to demolish these structures at an estimated unit cost of \$60.00/m³. Thus, for 4,700 m³ of material, the total cost is \$282,000.

L.3.9 Makeup Water Pumphouse

This reinforced concrete structure is demolished with a crane and wrecking ball. The estimated unit cost for demolition and removal is \$60.00/m³. A total of 718 m³ of concrete is involved, resulting in a total cost of \$43,000.

L.3.10 Office Building

The 600-m² steel-framed and -sided Office Building is demolished by conventional means, and the floor slab is broken up using a crane with wrecking ball. The estimated unit demolition cost is \$14.68/m² of floor area, for a total cost of \$9,000.

L.3.11 Warehouse

This 1,800-m² building is similar in construction to the Office Building. Using an estimated unit demolition cost of \$14.68/m², the total demolition cost is \$26,000.

L.3.12 Guard House

The reinforced-concrete Guard House is demolished with a crane and wrecking ball, at an estimated unit cost of \$60.00/m³. The building contains 265 m³ of concrete, resulting in a total demolition cost of \$16,000.

L.3.13 Gas Bottle Storage Building

The estimated unit cost to demolish and remove the 88-m² precast concrete Gas Bottle Storage Building is \$24.77/m² of floor area. Thus, the total removal cost is \$2,000.

L.3.14 Site Restoration

All of the subgrade portions of the reference BWR buildings are back-filled. Concrete rubble from the demolition of the structures is used as

necessary to backfill up to a level 1 m below grade. To restore the site, the top meter is backfilled with native topsoil and seeded with native ground cover; costs are given in Table L.3-9.

TABLE L.3-9. Estimated Site Restoration Costs

<u>Activity</u>	<u>Material Required (m³)</u>	<u>Unit Cost (\$/m³)</u>	<u>Cost (\$ thousands)</u>
Loading and Hauling	43 320	7.52	326
<u>Placing, Grading, and Seeding</u>	43 320	1.94	<u>84</u>
Total			410

L.4 AREAS FOR POTENTIAL COST REDUCTION

Adoption of innovative demolition procedures can lead to significant cost reductions. Two options for potential cost reduction are discussed below.

L.4.1 Crushing of Concrete

Substantial savings on hauling costs could be made by the use of a portable rock-crushing plant to crush the concrete. Such a crusher could handle pieces up to 0.9 m square, provided that the rebar is less than 1.2 m long. The cost of crushing the concrete (as of December 1978) is quoted at \$5.20 to \$6.50 per m³, stockpile measure. This cost could be partially offset by selling or using the crushed material. An appropriate use is backfilling the below-grade portions of the buildings. If crushed concrete is used on the site, a net savings of about \$6/m³ of the normal disposal costs would be realized.

L.4.2 Degradation of Concrete by Heating

An innovative method for removing the concrete in the Primary Containment Vessel is heating the concrete to the temperature (650 to 800°C) at which it degrades to a loose, crumbly material. The Primary Containment Vessel lends itself to this treatment, since it resembles a large furnace. However, this is an untried method and its feasibility would have to be verified on a small scale before being used on a large structure.

The exhaust from large portable oil or gas burners could be directed into the Primary Containment Vessel through its lower penetrations. The degraded concrete could be shaken off the reinforcing skeleton with little effort. The difference in thermal expansion of steel and concrete should result in the reinforcing steel being virtually free of concrete, thus allowing the salvage of nearly 100% of the reinforcing steel.

The cost of removing the concrete by this method is estimated at about \$12 to \$24 per m^3 , compared to the \$140-per- m^3 cost of drilling, blasting, and secondary breaking used in the cost estimate for the Reactor Building.

REFERENCES

1. J. M. McFarland, Report on Representative Cost Estimates for Demolition of Structures at a Boiling Water Reactor Site, PNL-3053, Prepared for Pacific Northwest Laboratory by McFarland Wrecking Corporation, Seattle, WA 98108, February 1979.

APPENDIX M

COST ESTIMATING BASES

The cost information developed in this study is based on unit cost data presented in this appendix. Categories for which basic cost data are given include: salaries, waste packaging, cask rental, transport, waste disposal, special equipment, and services and supplies. The data presented are all late 1977 or early 1978 costs.

The unit cost data presented can be used to develop cost estimates for other decommissioning projects. However, to ensure the applicability of the estimate to any specific situation, the data should be carefully examined and adjusted as necessary.

M.1 MANPOWER COSTS

Salary data for typical decommissioning staff positions are given in Table M.1-1. The original 1975 data base has been adjusted by a factor of 1.29, based on the Handy-Whitman Index, to account for escalation between 1975 and 1978.⁽¹⁾

M.2 RADIOACTIVE WASTE PACKAGING COSTS

The costs of packaging radioactive waste materials for disposal at a shallow-land burial site include the shipping container cost, the cost of any additional shielding, and, in the case of concentrator bottoms or ion-exchange resins, the cost of a solidifying agent. These costs are discussed in the following subsections.

M.2.1 Shipping Container Costs

The shipping containers assumed to be used for packaging radioactive waste materials for disposal are listed in Table M.2-1. A brief description, together with the displaced burial volume, the particular application, and the unit cost, is included for each type of container.

TABLE M.1-1. Decommissioning Staff Salary Data^(a)

Position	Base Pay (\$/yr)	Assumed Overhead Rate (%)	Cost (\$/yr)	Reference
<u>Management and Support Staff</u>				
Decommissioning Superintendent	43 800	70	74 500	2
Secretary	13 500	50	20 300	3
Clerk	13 500	50	20 300	3
Decommissioning Engineer	37 400	70	63 500	2
Assistant Decommissioning Engineer	25 800	70	43 800	2
Radioactive Shipment Specialist	19 300	70	32 900	3
Procurement Specialist	18 000	70	30 700	3
Tool Crib Attendant	15 500	50	23 200	4
Control Room Operator	19 300	50	29 000	(b)
Security Supervisor	19 300	70	32 900	3
Security Shift Supervisor	18 000	70	30 700	(b)
Security Patrolman	14 200	50	21 300	5
Contracts and Accounting Specialist	22 600	70	38 400	2
Health and Safety Supervisor	29 600	70	50 400	3
Health Physicist	23 200	70	39 400	3
Protective Equipment Attendant	15 500	50	23 200	3
Industrial Safety Specialist	25 800	70	43 800	3
Quality Assurance Supervisor	25 800	70	43 800	3
Quality Assurance Engineer	23 200	70	39 400	3
Quality Assurance Technician	15 500	50	23 200	3
Consultant (Safety Review)	67 000	0	67 000	3
<u>Decommissioning Workers</u>				
Shift Engineer	25 800	70	43 800	(b)
Crew Leader	21 900	70	37 300	4
Utility Operator	18 000	50	27 100	4
Laborer	17 300	50	25 900	5
Craft Supervisor	23 200	70	39 400	3
Craftsman	18 000	50	27 100	4
Sr. Health Physics Technician	19 300	70	32 900	3
Health Physics Technician	16 800	50	25 100	4

(a)Adjusted to January 1978.

(b)Study estimate.

TABLE M.2-1. Shipping Containers for Packaging Radioactive Materials

Description	Burial Volume (m ³)	Application	Estimated Unit Cost (\$) (a)
Steel Cask Liner 0.63 m O.D. x 1.02 m high, 150 kg empty	0.33	Shallow-Land Burial of Activated Reactor Components	450
Steel Box Specially Fabricated	Variable	Shallow-Land Burial of Large or Odd-Shaped Activated Reactor Components	6 000
Fiberglassed Plywood Box 1.2 m x 1.2 m x 2.4 m, 175 kg empty	3.64	Low Specific Activity Materials Piping, Concrete, Contaminated Components	400
Fiberglassed Plywood Box Specially Fabricated	Variable	Low Specific Activity Material, Over-Sized or Extra Heavy	~28/m ² of surface
Steel Cask Liner 1.38 m O.D. x 1.9 m high, 680 kg empty	2.84	Solidified Wet Solid Wastes	550
Standard Steel Drum 0.21 m ³ , 23 kg empty	0.21	Compacted Dry Solid Wastes	20
Stainless Steel Canister 0.76 m O.D. x 3.05 m high	1.38	Alternate Deep Geologic Disposal of Highly Activated Reactor Components	5 000

(a) Adjusted to early 1978.

M.2.2 Additional Shielding Costs

In some cases, additional lead shielding must be added to shipping containers to reduce surface radiation dose rates. The addition of this shielding is estimated to cost an average of \$1.23/kg, including labor and energy.

M.2.3 Solidifying Agent Costs

In this study, it is assumed that all concentrator bottoms, neutralized drain-decontamination solutions, and ion-exchange resins are mixed with a solidifying agent and poured into steel cask liners. The assumed solidifying agent is cement blended with varying amounts of sodium silicate and proprietary leach inhibitors. The agent is added at the rate of 978 kg/m³ of concentrator bottoms or neutralized drain-decontamination solutions and 498 kg/m³ of resin. The cost of this solidifying agent per container (~2.3-m³ capacity) is estimated at \$214 for concentrator bottoms or neutralized drain solutions and \$129 for resin.

M.3 CASK CHARGES

Some of the waste material shipped to a burial site is sufficiently radioactive to require transport in reusable shielded casks. In general, it is more economical to rent such casks than to purchase them, especially the larger ones. The casks assumed in this study for use in shipping highly radioactive materials are listed in Table M.3-1, together with physical characteristics and estimated rental charges.

TABLE M.3-1. Shielded Casks for Shipment of Radioactive Materials

<u>Description</u>	<u>Tare (kg)</u>	<u>Application</u>	<u>Daily Rental (\$)^(a)</u>
1.24 m O.D. x 1.56 m high 150-mm Pb thickness (B3 Cask)	9 300	Activated Components	100
NAC-1 Spent Fuel Cask (Nuclear Assurance Corp.)	22 000	Activated Control Rods	690
1.63 m O.D. x 2.34 m high 100-mm Pb thickness	16 330	Solidified Wet Solid Wastes	100
1.95 m O.D. x 1.04 m high 50-mm Pb thickness (7D-3L Cask)	7 000	Compacted Dry Solid Wastes (maximum seven 0.21-m ³ drums/cask)	100
IF-300 Spent Fuel Cask (General Electric Co.)	120 000	Spent Fuel to Repository; Highly Activated Components to Geologic Disposal (Alternate)	3 500 (30-day minimum)

(a)Adjusted to early 1978.

M.4 TRANSPORTATION COSTS

Most radioactive materials resulting from decommissioning are assumed to be shipped to a burial site in exclusive-use trucks. The exceptions, both assuming rail transport, are spent fuel shipments and the alternative case where highly activated reactor components are postulated to be shipped to deep geologic disposal. The shipping costs for both truck and rail transport are discussed in the following subsections.

M.4.1 Shipment by Exclusive-Use Trucks

Transport of radioactive waste materials from the BWR to a shallow-land burial site is assumed to be by truck. A rate schedule for legal-size and -weight shipments is shown in Table M.4-1, forming the basis for transportation costs used in this study. The table is reproduced from the published rates of a carrier licensed to transport radioactive materials.⁽⁶⁾

The gross vehicle weight (GVW) for normal shipments by truck is assumed to be less than 20.41 Mg. A 1600-km (1000-mi) round-trip distance is assumed. Thus, from the table, shipping charges are \$0.64/km (\$1.03/mi), for a total round-trip cost of \$1030.

The maximum allowed GVW without special equipment and special permission varies from state to state, ranging from 30.21 to 45.36 Mg, with the majority of states having limits in the vicinity of 33.11 Mg. Overweight charges by states also vary widely.⁽⁶⁾ For this study, a maximum allowed GVW of 33.11 Mg

TABLE M.4-1. Transportation Rates for Legal-Size and -Weight Truck Shipments (effective April 1, 1978)

Item No. 3000-D
Cancels 3000-C [12]

SECTION II - MILEAGE COMMODITY RATES

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.

[12] Reissued from Supplement 12 and effective April 26, 1978.

For Explanation of Abbreviations and Reference Marks, See Item 820.

for normal shipments and the overweight charges for the state of Washington are assumed to apply (the rates for Washington state are the highest). These overweight charges are shown in Table M.4-2. An additional surcharge of \$0.126 per km is imposed by the carrier for shipments with GVW greater than 20.41 Mg. Shipments in excess of 33.11 Mg GVW require special equipment and special permission. Carrier charges for these cases must be determined on a case-by-case basis.

For this study, the GVW of an unloaded exclusive-use van or tractor-trailer is assumed to be 12.70 Mg. Therefore, the payload per shipment is 7.71 Mg legal weight and 20.41 Mg overweight.

To ensure rapid turnaround on cask shipments, a second driver is required, costing an additional \$0.093/km (\$0.15/mi).⁽⁶⁾ This adds \$150 (round-trip) to the cost of each of these shipments.

TABLE M.4-2. Additional Charges when Gross Vehicle Weight Exceeds 20.41 Mg^(a,b)

<u>GVW (Mg)</u>	<u>Charge (\$/km)</u>
20.41 to 23.13	0.047
23.14 to 25.85	0.093
25.86 to 28.57	0.140
28.58 to 31.29	0.233
31.30 to 34.01	0.326
34.02 to 36.73	0.420
36.74 to 47.85 ^(c)	0.513

(a) In addition to the charges shown in the table, a flat charge of \$5.00 is levied.

(b) From Reference 6, for the state of Washington.

(c) Maximum GVW under any circumstances, from Reference 7, for the state of Washington.

In this study, the moisture separator reheaters are assumed to be segmented, their ends capped, and the segments shipped as oversize (as well as overweight) shipments. Regulations and conditions governing oversize shipments in the state of Washington are assumed.⁽⁷⁾ Table M.4-3 summarizes the applicable requirements for oversize shipments on two-lane highways. The oversize shipments assumed in this study are estimated to cost \$500/shipment more than legal-size shipments of the same weight. This additional cost covers the expense of special permits and escort cars.

The base shipping costs assumed in this study for truck transport are summarized in Table M.4-4.

TABLE M.4-3. Requirements for Oversize Truck Shipments^(a)

<u>Characteristic Dimension of Vehicle/ Load Combination</u>	<u>Special Permit Required in Excess of:</u>	<u>Escort Car Required in Excess of:</u>	<u>Maximum Allowed Dimensions</u>
Width	2.44 m (8 ft)	3.05 m (10 ft)	4.27 m (14 ft)
Height	4.11 m (13.5 ft)	--(b,c)	--(b,c)
Length	19.81 m (65 ft)	30.48 m (100 ft)	--(b)

(a)Based on regulations in the state of Washington for two-lane highways. See Reference 7.

(b)No specific requirement, but escort car may be required at discretion of Highway Department.

(c)Heights exceeding 4.42 m (14.5 ft) are generally considered unacceptable because of the special routing and preparations required.

TABLE M.4-4. Shipping Cost per Truck Shipment

<u>Status</u>	<u>Payload (Mg)</u>	<u>GVW (Mg)</u>	<u>Cost (\$)^(a)</u>
Legal	7.71	20.41	1 180
Oversize	7.71	20.41	1 680
Overweight	20.41	33.11	1 920
Oversize and Overweight	20.41	33.11	2 420

(a)1600-km round-trip distance, two drivers.

M.4.2 Shipment by Rail

Shipment by rail is assumed for spent fuel and for the alternate case of deep geologic disposal of the highly activated reactor components. Assuming a round-trip distance of 4828 km (3000 mi), the shipping cost is estimated to be \$110/Mg (\$5/100 lb).⁽⁸⁾ This amounts to about \$25,100 for a rail car carrying a GE IF-300 cask. Should a special train be required, a surcharge of \$11.81/km (\$19/mi), or about \$57,000 per trip, is added.

M.5 WASTE DISPOSAL COSTS

A basic assumption in this study is that all of the radioactive material removed from the reference BWR during decommissioning is placed in a shallow-land burial facility. The alternative placing of the highly activated reactor components in a deep geologic repository is also considered. The unit costs of waste disposal are given in this section.

M.5.1 Costs for Shallow-Land Burial

The shallow-land burial costs used in this study are based on a late 1977 price list from Nuclear Engineering Co., Inc., which operates burial sites at Richland, Washington, and Beatty, Nevada. The costs used in this study are for the Beatty site and are given in Table M.5-1.⁽⁹⁾

M.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 here are somewhat speculative. Based on an analysis and discussion presented in Reference 10, a cost of \$7100/m³ is assumed for such disposal of highly activated materials. Thus, for the canisters presently considered for deep geologic disposal (1.39-m³ burial volume), the disposal charge is \$9869/canister.

M.6 COSTS OF SERVICES AND SUPPLIES

Various types of services and supplies are required for decommissioning. The estimated unit costs of the major items are discussed here.

TABLE M.5-1. Shallow-Land Burial Costs^(a)

I DISPOSAL CHARGES, NON-TRU WASTE

A. Steel Drums, Wood or Fiber Boxes

Container Surface Dose Rate (R/hr)(b)	Price per Unit Volume	
	(\$/m ³)	(\$/ft ³)
0.00 - 0.20	93.57	2.65
0.21 - 1.00	105.93	3.00
1.01 - 2.00	123.59	3.50
2.01 - 5.00	160.66	4.55
5.01 - 10.00	197.74	5.60
10.01 - 20.00	268.36	7.60
20.01 - 40.00	354.87	10.05
40.01 - 60.00	547.31	15.50
60.01 - 80.00	674.42	19.10
80.01 - 100.00	741.51	21.00
>100.00	by request	by request

B. Disposable Liners

Container Surface Dose Rate (R/hr)(b)	Surcharge Per Liner (\$)	Price per Unit Volume	
		(\$/m ³)	(\$/ft ³)
0.00 - 0.20	NONE	93.57	2.65
0.21 - 1.00	50.00	93.57	2.65
1.01 - 2.00	125.00	93.57	2.65
2.01 - 5.00	175.00	93.57	2.65
5.01 - 10.00	250.00	93.57	2.65
10.01 - 20.00	325.00	93.57	2.65
20.01 - 40.00	400.00	93.57	2.65
40.01 - 60.00	475.00	93.57	2.65
60.01 - 80.00	550.00	93.57	2.65
80.01 - 100.00	625.00	93.57	2.65
>100.00	by request	by request	by request

C. Minimum Disposal Charge \$100.00/shipment

II SURCHARGE(S)

A. Curie Surcharge (per shipment)

- 0 - 100 curies no charge
- 101 - 300 curies \$300.00
- 301 - License Limits \$300.00 plus
(i.e., 50,000 Ci maximum) \$ 0.05/Ci above 300 Ci

B. Handling Surcharge

- 0 - 4.54 Mg
(0 - 10,000 lb) no charge
- 4.54 - 22.68 Mg^(c)
(10,001 - 50,000 lb) \$50.00 + 2.2¢/kg over 4.54 Mg
(\$50.00 + 1¢/lb over 10,000 lb)
- Special equipment by special quotation

C. Cask Handling Fee \$200.00

(a) Prices shown as of September 1977 See Reference 9

(b) Maximum reading at container surface, irrespective of physical size or configuration

(c) Limit of site equipment at Beatty

Electricity

A principal services cost item is electric power. In this study, a unit cost of \$0.015/kWh, or \$15/MWh, is assumed for electricity.

Fuel Oil

At the reference BWR, fuel oil is used to provide energy for building heat and radwaste concentrator operation. A unit cost of \$132/m³ is assumed in this study for fuel oil.

Decontamination Chemicals

The unit costs of chemicals used for recirculatory (loop-type) decontamination are estimated to be:

- EDTA - \$9.92/kg
- Oxalic Acid - \$5.36/kg
- Citric Acid - \$3.75/kg.

For a mixture of these three chemicals, one-third each by weight, the resulting cost is \$6.34/kg.

The chemicals used for single-pass decontamination of drains are estimated to have the following unit costs:

- Phosphoric Acid - \$3.87/kg
- Sodium Hydroxide (50-wt% solution) - \$178.29/m³ (\$37.44/drum).

Filter/Demineralizer Resins

The unit costs of filter/demineralizer resins used for contaminated water cleanup are estimated to be \$8.36/kg of bead resin and \$5.87/kg of mixed Powdex resins, at a mixture by weight of two parts cation Powdex resin to one part anion Powdex resin.

Welding and Cutting Supplies

The capping of segments of large heat exchangers requires carbon steel plate. Costs for this plate are estimated to be:

- 6.35-mm (1/4-in.) plate - \$20.34/m²
- 15.88-mm (5/8-in.) plate - \$49.39/m².

Welding and cutting requires the use of various gases. Unit costs for these are estimated to be:

- Oxygen - \$2.70/6.34-m³ cylinder, or \$0.43/m³
- Acetylene - \$12.65/6.51-m³ cylinder, or \$1.94/m³.

For gas-arc welding of carbon steel plate, the total unit costs for materials (including gas and welding rod) are as follows:

- 6.35-mm (1/4-in.) fillet weld - \$1.25/m
- 12.70-mm (1/2-in.) fillet weld - \$5.02/m.

Paint

The unit cost of the paint used for sealing contaminated surfaces is estimated to be \$5.28/l.

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* Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and the National Technical Information Service, Springfield, Virginia 22161.

** Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.

APPENDIX N

PUBLIC RADIOLOGICAL SAFETY ASSESSMENT DETAILS

The purpose of this appendix is to quantify the parameters and define the methodology for estimating the public radiological safety impacts of decommissioning the reference BWR power station. For each of three separate decommissioning alternatives analyzed, atmospheric releases of radioactivity are calculated for the decommissioning tasks assumed to be performed in the reference BWR buildings. Details for estimating the public radiation exposure resulting from spent-fuel or waste shipments are also included in this appendix.

The impacts of decommissioning on the safety of the public are principally related to the hazards associated with the atmospheric release of radioactive materials during decommissioning, both from planned tasks and from accidents. The scenarios analyzed are designed to use realistically maximized parameters. These parameters are carefully chosen to cover a broad spectrum of conditions. Atmospheric release mechanisms are quantified using either measured release factors or information about known physical or chemical behaviors under the postulated conditions. The estimated routine radionuclide releases to the atmosphere are tabulated in the chronological order of the decommissioning tasks for each building for each decommissioning mode. A detailed probabilistic analysis of the postulated accident scenarios is not within the scope of this study. However, selected generic accident categories are considered that include operational accidents, natural phenomena, aircraft crashes, and man-caused events.

The following sections contain detailed discussions of the technical approach and public safety details for immediate dismantlement, safe storage, entombment, and waste transportation. A summary of this information is given in Sections 11.2 and 11.4 of Chapter 11 (Volume 1).

N.1 TECHNICAL APPROACH

To estimate the radiological public safety impacts of the release of radioactivity to the atmosphere from decommissioning tasks, the following basic assumptions are made:

1. The facility has fulfilled existing criteria for the containment of radionuclides for accidents and natural phenomena during its operational lifetime.
2. The spent fuel from the final core loading is assumed to be shipped from the BWR during a 15-month period for immediate dismantlement. Fuel handling accidents, except for transportation accidents, are assumed to be covered in the FSAR for the reference BWR and are not considered further in this study.⁽¹⁾
3. To ensure proper air flow for the decommissioning workers, and to protect against uncontrolled atmospheric releases of radioactivity, HEPA filters are assumed to be installed in the Reactor Building, Turbine Generator Building, and Radwaste and Control Building exhaust ventilation systems. These filters are tested in-place on a regular basis. The measured particle collection efficiency of these HEPA filters is 99.95%.⁽²⁾ Airborne releases of radioactivity are assumed to take place at ground level for atmospheric dispersion calculations, and are assumed to pass through a single HEPA filter system with an assumed transmission factor of 5×10^{-4} .⁽²⁾
4. The containment integrity of the buildings at the reference BWR is assumed to be breached only after the radioactive contamination levels are at the accepted levels for unrestricted use.
5. In areas with high levels of radioactive contamination, a temporarily installed "greenhouse," or contamination control envelope, is assumed to 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 BWR buildings from selected decommissioning tasks.

6. The leakage rate from a contamination control envelope is assumed to be 10% for all tasks involving its use. This assumed leakage is used as a maximized value to account for routine ruptures or failures of the temporary structures.
7. The airflow rates assumed in this study are: $3000 \text{ m}^3/\text{hr}$ for all contamination control envelopes, $1.3 \times 10^4 \text{ m}^3/\text{hr}$ for the Reactor Building, $2.4 \times 10^4 \text{ m}^3/\text{hr}$ for the Turbine Generator Building, and $4.8 \times 10^4 \text{ m}^3/\text{hr}$ for the Radwaste and Control Building.
8. Unneeded hazardous chemicals and equipment are removed after the reference BWR is shut down. Decontamination agents such as phosphoric acid, ethylenediaminetetraacetic acid (EDTA), oxalic acid, and citric acid are available in the plant. Unneeded ion exchange resins and resin beds are removed.
9. The reference radionuclide inventories referenced in this appendix refer to the inventories defined in Appendix E. These inventories characterize activated structural materials, as well as surface contamination resulting from contact with the reactor water, steam, or condensate.
10. All chemical or manual decontamination tasks are assumed to remove 90% of the surface contamination.
11. The airborne concentrations of dust or liquid droplets are assumed to be $1 \times 10^{-2} \text{ g/m}^3$, equal to the concentrations observed at the Elk River reactor decommissioning.^(3,4) For tasks involving blasting or explosions, the airborne concentrations are assumed to be a factor of 10 higher, or $1 \times 10^{-1} \text{ g/m}^3$.⁽⁵⁾
12. All radioactive wastes shipped offsite are assumed to be shipped in accordance with Department of Transportation (DOT) regulations. Spent reactor fuel is assumed to be shipped 2400 km by rail to a repository, and radioactive wastes are assumed to be shipped 800 km by truck to a shallow land burial ground.
13. Radiation doses to the maximum-exposed individual and to the population residing within 80 km of the reference site are calculated using the

environmental data and assumptions discussed in Appendix F of this study. These methods are consistent with the methods outlined in Regulatory Guide 1.109.⁽⁶⁾

Other assumptions relating to specific decommissioning tasks are made and discussed where they apply to the analysis of public radiation exposure.

N.2 IMMEDIATE DISMANTLEMENT

The first option considered in the safety evaluation of decommissioning the reference BWR is immediate dismantlement of the facility. This alternative results in removal of the radioactively contaminated portions of the buildings, the equipment, and the site so that unrestricted use by the public can be permitted.

The assessment of public safety during immediate dismantlement includes a consideration of both routine tasks and postulated accidents. These tasks and accidents can generate considerable amounts of airborne radioactivity, primarily in the form of solid particles and/or suspended liquid droplets. Airborne contamination control measures are assumed to be used where they are considered necessary. The use of other appropriate controls is also considered. The inventories of radioactive contamination in the reference BWR are discussed in detail in Appendix E, and are summarized in Table N.2-1.

In the following sections, the atmospheric releases from immediate dismantlement are described for both routine tasks and postulated accidents. The atmospheric releases are estimated by determining the realistic maximum atmospheric release for each task or accident, and using this value whenever the task or accident is encountered, even for areas containing lower contamination levels.

N.2.1 Routine Immediate Dismantlement Tasks

A complete discussion of the tasks for immediate dismantlement of the reference BWR is contained in Appendix I. The chronological sequence of immediate dismantlement tasks is listed in Figure I.2-2. These tasks and their time requirements are listed for the Reactor Building, Turbine Generator

TABLE N.2-1. Summary of Radioactive Contamination
at the Reference BWR

Inventory/Location	Reference Radionuclide Inventory Number(a)	Radioactivity Level(b)
Activated Stainless Steel		
Reactor Vessel Internals	1	$7.6 \times 10^5 \text{ Ci/m}^3$
Separator	1	$1.0 \times 10^4 \text{ Ci/m}^3$
Activated Carbon Steel		
Reactor Vessel	2	$1.1 \times 10^2 \text{ Ci/m}^3$
Sacrificial Shield	2	$5.0 \times 10^1 \text{ Ci/m}^3$
Shroud	2	$1.7 \times 10^6 \text{ Ci/m}^3$
Activated Concrete		
Sacrificial Shield	3	$4.7 \times 10^{-2} \text{ Ci/m}^3$
Contaminated Piping		
Reactor Water	4	$1.1 \times 10^0 \text{ Ci/m}^2$
Steam	4	$5.0 \times 10^{-3} \text{ Ci/m}^2$
Condensate	4	$5.0 \times 10^{-2} \text{ Ci/m}^2$
Contaminated Equipment		
Reactor Water	4	$3.6 \times 10^{-1} \text{ Ci/m}^2$
Steam--General	4	$5.0 \times 10^{-3} \text{ Ci/m}^2$
Steam--Turbine	4	$5.0 \times 10^{-4} \text{ Ci/m}^2$
Condensate--General	4	$5.0 \times 10^{-2} \text{ Ci/m}^2$
Condensate--Main Condenser	4	$5.0 \times 10^{-3} \text{ Ci/m}^2$
Condensate--Feedwater Heaters	4	$5.0 \times 10^{-3} \text{ Ci/m}^2$
Condensate--Waste Tanks	4	$5.0 \times 10^0 \text{ Ci/m}^2$
Surface Contamination		
General Surfaces	5	$2.5 \times 10^{-3} \text{ Ci/m}^2$
Isolated Surfaces	5	$2.5 \times 10^{-2} \text{ Ci/m}^2$

(a) These numbers refer to the reference radionuclide inventories discussed in Appendix E.

(b) Estimated levels before decontamination are shown.

Building, and the Radwaste and Control Building. The following subsections contain discussions of the methods used for calculating atmospheric releases of radioactivity and the public radiation doses resulting from immediate

dismantlement of the reference BWR. Information about the routine atmospheric releases is presented in the chronological order of the immediate dismantlement tasks.

N.2.1.1 Methods for Calculating Atmospheric Releases of Radioactivity

Reference immediate dismantlement tasks are defined and analyzed to determine generic airborne release mechanisms. A summary of the reference immediate dismantlement tasks considered in this study is shown in Table N.2-2. A complete discussion of each task considered is contained below:

Radiation Survey. Radiation surveys are performed at the start of decommissioning, prior to termination of the nuclear license, and at other times during decommissioning to determine the residual levels of radioactive contamination. Radiation surveys involve crews of health physics technicians who walk through the plant and who are assumed to disturb settled radioactivity. The general housekeeping practices followed prior to and during dismantlement determine the amount of loose surface contamination, which varies with location within the reference BWR.

The relationship between the amount of material in the air above a surface and the contamination level of that surface has been studied experimentally.⁽⁷⁾ This relationship is shown by Equation N.1.

$$S_f = \frac{C_a}{C_s} \quad (N.1)$$

where:

S_f • the resuspension factor, m^{-1}

C_a • the concentration of airborne radioactivity at a given location inside a building, Ci/m^3

C_s • the concentration of radioactivity on building surfaces, Ci/m^2 .

Measured values of S_f vary between 1×10^{-5} and $2 \times 10^{-3} m^{-1}$, depending upon experimental conditions and measurement methods.^(7,8,9)

TABLE N.2-2. Immediate Dismantlement Safety Analysis Summary

Task	Operation Description	Routine Airborne Release	Routine Atmospheric Release Calculation Summary	Postulated Hazardous Situations
Radiation Survey	Crews of Health Physics Technicians Walk Through the BWR to Locate and Characterize Residual Radioactive Contamination	Airborne Dust is Removed from Floor Surfaces	Resuspension Analysis: $(\text{Resuspension Factor}) \times (\text{Surface Radioactive Concentration}) \times (\text{Air Volume Covering Floor}) \times (\text{Building HEPA Transmission Factor}) = (\text{Atmospheric Release})$	<ul style="list-style-type: none">• None Analyzed
Surface Cleaning				
1. Sweeping	Small Quantities of Loose Contamination on Floors are Removed by Manual Sweeping with Sweeping Compounds	Airborne Dust is Removed from Floor Surfaces During Sweeping	Resuspension Analysis: $(\text{Resuspension Factor}) \times (\text{Surface Radioactivity Concentration}) \times (\text{Air Volume Covering Floor}) \times (\text{Building HEPA Transmission Factor}) = (\text{Atmospheric Release})$	<ul style="list-style-type: none">• Fire in Contaminated Sweeping Compound
2. Vacuuming	Small Quantities of Loose Contamination on Surfaces Are Removed by Manual Vacuuming Procedures	Airborne Dust is Removed from Floor Surfaces During Vacuuming	Resuspension Analysis: $(\text{Resuspension Factor}) \times (\text{Surface Radioactivity Concentration}) \times (\text{Air Volume Through Vacuum Cleaner}) \times (\text{Vacuum HEPA Transmission Factor}) \times (\text{Building HEPA Transmission Factor}) = (\text{Atmospheric Release})$	<ul style="list-style-type: none">• Vacuum Bag Rupture
3. Water-Jet	Hand-Held High-Pressure Water Spray is Used to Remove Loose and Soluble Surface Contamination	Airborne Particles are Caused by High Pressure Water Sprays	Entrainment of Contaminated Water: $(\text{Surface Contamination Removal Fraction}) \times (\text{Surface Radioactive Concentration}) \times (\text{Water-Jet Cleaning Rate}) \div (\text{Liquid Flow Rate}) \div (\text{Solution Density}) \times (\text{Airborne Concentration}) \times (\text{Building Air Flow Rate}) \times (\text{Water-Jet Use Time}) \times (\text{Building HEPA Transmission Factor}) = (\text{Atmospheric Release})$	<ul style="list-style-type: none">• Spilling Contaminated Liquids• Loss of Services
4. Hand Washing	Limited Areas Are Cleaned with Mops or Sponges	Liquids are Made Airborne as Sprays	Not Calculated	<ul style="list-style-type: none">• Spilling Contaminated Liquids
Chemical Decontamination				
1. Liquid Leaks	Loop Chemical Decontamination is Used in Piping and Equipment	Liquids are Made Airborne from Liquid Leaks	Entrainment of Contaminated Water: $(\text{Surface Contamination Removal Fraction}) \times (\text{Internal Surface Radioactivity Concentration}) \times (\text{Internal Surface Area}) \times (\text{Leak-Weighted Air-Volume}) \times (\text{Airborne Concentration}) \div (\text{Volume of Liquid}) \div (\text{Density of Liquid}) \times (\text{Building HEPA Transmission Factor}) = (\text{Atmospheric Release})$	<ul style="list-style-type: none">• Loss of Services• Spills• Seal Failures
2. Spray Leaks	Loop Chemical Decontamination is Used in Piping and Equipment	Liquids are Made Airborne from Spray Leaks	Entrainment of Spray: $(\text{Surface Contamination Removal Fraction}) \times (\text{Internal Surface Radioactivity Concentration}) \times (\text{Internal Surface Area}) \times (\text{Volume of Leak}) \times (\text{Total Liquid Volume}) \times (\text{Fraction of Leak to Reach HEPA System}) \times (\text{Building HEPA Filter Transmission Factor}) = (\text{Atmospheric Release})$	<ul style="list-style-type: none">• Loss of Services• Material Failures• Seal Failures
Piping or Equipment Removal & Segmentation				
1. Cutting in Air	Piping and Equipment is Segmented for Offsite Disposal	Vaporized Metal is Made Airborne (Contamination Control Envelope)	Contaminated Piping or Equipment: $(\text{Length of Cut}) \times (\text{Width of Cut}) \times (\text{Surface Radioactivity Concentration}) \times (\text{Building HEPA Transmission Factor}) = (\text{Atmospheric Release})$ Activated Piping or Equipment: $\text{Same as Above} \times (\text{Material Thickness}) \div (\text{Material Density})$	<ul style="list-style-type: none">• Loss of Services• Oxidetylene Explosion• Local Loss of Contamination Control
2. Cutting Under Water	Piping and Equipment is Segmented for Offsite Disposal	Liquid Sprays are Made Airborne (Contamination Control Envelope)	Entrainment of Contaminated Water: $(\text{Water Radioactivity Concentration}) \times (\text{Airborne Radioactivity Concentration}) \times (\text{Control Envelope Airflow Rate}) \times (\text{Cutting Time}) \times ((\text{Control Envelope Leak Rate}) \times (\text{Building HEPA Filter Transmission Factor}) \div (\text{Water Density})) = (\text{Atmospheric Release})$	<ul style="list-style-type: none">• Loss of Services• Oxidetylene Explosion• Local Loss of Contamination Control
Concrete Removal				
1. Drilling	Holes are Drilled for Insertion of Concrete Spallers or Explosive Charges	Concrete Dust is Made Airborne (Contamination Control Envelope)	Concrete Dust Generation: $(\text{Number of Holes}) \times (\text{Surface Area per Hole}) \times (\text{Surface Radioactivity Concentration}) \times (\text{Airborne Concentration}) \times (\text{Control Envelope Air Flow Rate}) \times (\text{Drilling Time}) \times (\text{Control Envelope Leak Rate}) \times (\text{Building HEPA Filter Transmission Factor}) \div (\text{Volume of Holes Drilled}) \div (\text{Density of Concrete}) = (\text{Atmospheric Release})$	<ul style="list-style-type: none">• Loss of Services• Equipment Failure• Local Loss of Contamination Control
2. Spalling	Hydraulically Controlled Concrete Spallers	Concrete Dust is Made Airborne (Contamination Control Envelope)	Not Calculated	<ul style="list-style-type: none">• Loss of Services• Local Loss of Contamination Control
3. Blasting	Explosive Charges for Concrete Removal	Concrete Dust is Made Airborne (Contamination Control Envelope)	Concrete Dust Generation: $(\text{Control Envelope Volume}) \times (\text{Number of Blasts}) \times (\text{Airborne Concentration}) \times (\text{Radioactivity Concentration}) \div (\text{Density of Concrete}) = (\text{Atmospheric Release})$	<ul style="list-style-type: none">• Improper Detonation• Loss of Contamination Control
4. Packaging	LNG-Powered Front-End Loader	Concrete Dust is Made Airborne (Contamination Control Envelope)	Not Calculated	<ul style="list-style-type: none">• LNG Explosion

The concentration of airborne radioactivity at a given location inside a building is calculated by rearranging Equation N.1 as shown in Equation N.2.

$$C_a = S_f C_s \quad (N.2)$$

where the terms S_f , C_s , and C_a are the same as defined for Equation N.1.

The total radioactivity made airborne in a building is calculated by using an assumed average resuspension factor associated with the radiation survey of $1 \times 10^{-4} \text{ m}^{-1}$, the surface radionuclide concentrations in the building, and the total volume of air contaminated during the radiation survey as shown in Equation N.3.

$$Q_s = (1 \times 10^{-4} \text{ m}^{-1}) C_s V_s \quad (N.3)$$

where:

- Q_s • the total airborne radioactivity inside a building, Ci
- C_s • the surface radioactivity concentration on building surfaces, Ci/m²
- V_s • the volume of air covering the surface involved in the radiation survey (assumed to be 1 m deep), m³.

The airborne radioactivity generation rate during the radiation survey is found by dividing the airborne radioactivity, Q_s , by the time required to perform the survey. The total radioactivity released to the atmosphere is found by multiplying the airborne radioactivity generated in each building by the HEPA filter transmission factor.

Surface Cleaning. The surface cleaning operations considered are sweeping, vacuuming and water-jet sprays.

1. Sweeping. Loose floor contamination is assumed to be removed by manual sweeping operations with the use of sweeping compounds. The amount of airborne radioactivity generated during sweeping is determined by using a resuspension factor analysis similar to the one discussed for the

radiation survey. The amount of radioactivity made airborne is highly variable and is characterized by the properties of the loose contamination and the sweeping method. Sweeping compounds are designed to lower the resuspension by binding small particles of surface contamination to larger particles of sweeping compound.

Data from experimental studies of resuspension during sweeping are limited. One study suggests a resuspension factor of $1.9 \times 10^{-4} \text{ m}^{-1}$.⁽⁸⁾ For this study, a conservatively large resuspension factor of $5 \times 10^{-4} \text{ m}^{-1}$ is used. This factor is 5 times larger than the resuspension factor assumed for the radiation survey. The resuspended radioactivity is assumed to be confined in the lower meter of air in the room in which the sweeping occurs. Thus, the airborne radioactivity in a building is calculated using the volume of air disturbed by sweeping (assumed to be the volume 1 m deep above the surface area swept), the surface radioactivity concentration, and the assumed resuspension factor. The total atmospheric radioactivity released from sweeping is found by multiplying the airborne radioactivity generated in each building by the HEPA filter transmission factor.

2. Vacuuming. Vacuuming can be used as an alternative to manual sweeping for removal of loose surface contamination. Thus, certain areas of the buildings considered in this study are designated for vacuuming. The vacuum exhaust is assumed to be fitted with a 99.95%-efficient HEPA filter system. The airborne radioactivity in a building resulting from vacuuming is calculated by finding the product of the surface contamination level, the total area vacuumed, the assumed surface contamination removal fraction for vacuuming (0.5), and the vacuum system HEPA filter transmission factor. The total radioactivity released to the atmosphere is found by multiplying the airborne radioactivity in each building from vacuuming by the building HEPA transmission factor.
3. Water-Jet Sprays. The water-jet is a hand-held high-pressure water spray designed to remove loose surface contamination. The operating water pressure is about $3.1 \times 10^4 \text{ kPa}$. The principal mechanism for the generation of airborne radioactivity from the water-jet is the suspension of liquid

droplets containing contamination removed from surfaces. The water spray produces fine droplets smaller than 300 μm in diameter (about the size of the droplets found in fogs or mists). Larger droplets are broken into smaller droplets as they impact on the contaminated surfaces. Thus, there is the potential for a significant formation of small droplets with a considerable variation in the total quantity of radioactivity contained in the droplets.

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 $1 \times 10^{-2} \text{ g/m}^3$ with vigorous mixing in air. This is the maximum mass concentration that can be found in air velocities less than 0.046 m/s.⁽⁴⁾ Measurements have shown that there is fairly constant distribution of 10- to 20- μm diameter particles in a water-jet spray at a concentration of about $1 \times 10^{-2} \text{ g/m}^3$, including the gross entrainment of larger particles.⁽¹⁰⁾ Since particles smaller than 10 μm are in the respirable-size range, they are potentially hazardous. The quantity of radioactivity found in these airborne droplets is influenced by factors such as the quantity of radioactivity on the surface, the solubility of the surface radioactivity in the liquid, and the contact between the liquid and the surface.

Operating parameters of the water-jet vary with the requirements of the surface decontamination effort. For this study, a water flow rate of $2.3 \times 10^{-2} \text{ m}^3/\text{min}$ and a cleaning rate of $0.77 \text{ m}^2/\text{min}$ are assumed. The water-jet is assumed to remove 90% of the surface contamination.

The concentration of radioactivity in the liquid resulting from use of the water-jet is defined by Equation N.4.

$$C_v = 0.9 C_s \left(\frac{0.77 \text{ m}^2/\text{min}}{2.3 \times 10^{-2} \text{ m}^3/\text{min}} \right) \quad (\text{N.4})$$

where:

C_v • the concentration of radioactivity in the liquid resulting from use of the water-jet, Ci/m³

0.9 • the fraction of the surface contamination removed by the water spray

C_s • the radioactivity concentration on the building surfaces, Ci/m²

0.77 • the water-jet cleaning rate, m²/min

2.3×10^{-2} • the water flow rate for the water-jet, m³/min.

The airborne radioactivity generated in a building resulting from use of the water-jet is calculated from the liquid radioactivity concentration using Equation N.5.

$$Q_h = \frac{C_v}{\rho} (1 \times 10^{-2} \text{ g/m}^3) F_v t \quad (\text{N.5})$$

where:

Q_h • the airborne radioactivity in the building resulting from water-jet operations, Ci

C_v • liquid radioactivity concentration, Ci/m³

ρ • the density of the water-jet solution, g/m³

1×10^{-2} • the airborne concentration, g/m³

F_v • the building ventilation flow rate, m³/hr

t • the time spent using the water-jet, hr.

The atmospheric release of radioactivity from water-jet sprays is calculated by multiplying the airborne radioactivity in the building by the building HEPA filter transmission factor.

Chemical Decontamination. Liquid and spray leaks are assumed to occur during loop chemical decontamination of piping and equipment.

1. Liquid Leaks. A liquid leak results in a small airborne release in a building, since the liquid will retain most of the dissolved radioactive contamination until it reaches a floor drain. Therefore, only a small volume of air is assumed to contain $1 \times 10^{-2} \text{ g/m}^3$ of airborne radioactivity.

The concentration of radioactivity in the decontamination solution is a function of the radioactivity removal fraction, the surface contamination level, and the volume of decontamination solution, as shown in Equation N.6.

$$C_v = \frac{0.9 C_s}{V_c} A_s \quad (\text{N.6})$$

where:

- C_v • the liquid radioactivity concentration, Ci/m^3
- 0.9 • the surface contamination removal fraction
- C_s • surface concentration of radioactivity on internal pipe and equipment surfaces, Ci/m^2
- V_c • the volume of decontamination solution in the loop, m^3
- A_s • the internal surface area decontaminated, m^2 .

It is assumed that only a small volume of air above a liquid leak becomes contaminated. This volume is assumed to be the amount of air 2 m deep above the floor surface area contaminated by the liquid leak. The floor surface covered by a leak is assumed to be 1 m^2 for each liter of leak. Thus, the total airborne radioactivity is calculated using Equation N.7.

$$Q_1 = \frac{V_r}{\rho} (1 \times 10^{-2} \text{ g/m}^3) C_v \quad (\text{N.7})$$

where:

- Q_1 • the airborne radioactivity in the building resulting from the liquid leak, Ci

- V_r • the volume of room air containing the airborne release (the volume of air 2 m-deep over the surface area contaminated, which is assumed to be 1 m^2 per $1 \times 10^{-3} \text{ m}^3$ of leak), m^3
- 1×10^{-2} • the airborne concentration, g/m^3
- ρ • the density of the decontamination liquid, g/m^3
- C_v • the liquid radioactivity concentration, Ci/m^3 .

The total radioactivity released to the atmosphere from a liquid leak is calculated by multiplying the airborne radioactivity in the building by the building HEPA filter transmission factor.

2. Spray Leaks. Spray leaks of chemical decontamination liquids can lead to a major airborne release since the radioactive liquids are directly entrained in the building air. For this study, 0.3% of the airborne contamination in the building is assumed to be of the correct particle size to reach the building HEPA filter.⁽¹¹⁾ Thus, the total release to the environment from spray leaks is 0.3% of the leaked radioactivity multiplied by the building HEPA filter transmission factor.

Piping or Equipment Removal. Contaminated or activated piping and equipment, are assumed to be removed and segmented for disposal during immediate dismantlement. For this study, four cutting methods are assumed to be used: arc saw, plasma-arc torch, oxyacetylene torch and power hacksaw. A summary of the cutting parameters for these methods is listed in Table N.2-3.

TABLE N.2-3. Cutting Parameters for Piping and Equipment Removal

Cutting Method	Cut (kerf) Width, (m)	Airborne Release
Arc Saw	9.5×10^{-3}	Metal Fragments, Vapors
Plasma Torch	3.1×10^{-3} to 6.4×10^{-3}	Condensed Metals, Gases, Smoke Particles
Oxyacetylene Torch	6.4×10^{-3}	Condensed Metals, Vapors
Power Hacksaw	6.4×10^{-3}	Metal Fragments, Vapors

In this study, cutting tasks are assumed both in air and under a water cover.

1. In Air. For piping or equipment containing surface contamination, the airborne release in the building from cutting tasks in air is calculated using Equation N.8.

$$Q_C = L K C_S \quad (N.8)$$

where:

- Q_C • the airborne radioactivity from cutting contaminated pipe or equipment, Ci
- L • the length of cut, m
- K • the width of the cut (kerf), m
- C_S • the surface radioactivity concentration, Ci/m².

Equation N.8 is based on the conservative assumption that all of the surface radioactive contamination associated with the cut width is vaporized and made airborne during cutting operations.

The airborne release associated with cutting neutron-activated piping or equipment is calculated using Equation N.9.

$$Q'_C = L K T C_m \quad (N.9)$$

where:

- Q'_C • the airborne radioactivity from cutting neutron-activated material in air, Ci
- L, K • are defined previously for Equation N.8
- T • the thickness of the material being cut, m
- C_m • the concentration of radioactivity in the material being cut, Ci/m³.

Again all material in the kerf width is conservatively assumed to be vaporized and is assumed to be airborne in the building. Equations N.8 and N.9 are used for all cutting methods in air, assuming a maximum kerf width of 6.4×10^{-3} m for the plasma-arc torch, oxyacetylene torch and hacksaw, and 9.5×10^{-3} m for the arc saw.

The amount of radioactivity released to the atmosphere is calculated by multiplying either Q_C or Q'_C by the building HEPA filter transmission factor.

2. Underwater. Highly activated piping or equipment is assumed to be segmented by a remote, underwater cutting method, using either the plasma-arc torch or the arc saw. The heat and agitation associated with either of these cutting methods is assumed to drive some of the contaminated cover water into the air. In some cases, a contamination control envelope is assumed to be used to mitigate the airborne release. The total contamination suspended in the water is assumed to be the maximum value calculated for all underwater cutting tasks, thus the airborne releases are maximized.

The air in the vicinity of the cutting tasks is assumed to become filled with radioactive water vapor to a concentration of 1×10^{-2} g/m³ during underwater cutting tasks. Cutting parameters are assumed to be the same as those described for in-air piping or equipment removal. Equation N.10 is used to calculate the quantity of airborne radioactivity released from underwater cutting tasks.

$$Q'_C = \frac{C_V}{\rho} (1 \times 10^{-2} \text{ g/m}^3) F'_V + F_C \quad (\text{N.10})$$

where:

- Q'_C • the quantity of airborne radioactivity transmitted in the building airflow, Ci
- C_V • the maximum concentration of radioactive material in the cover water, Ci/m³

- ρ • the density of the cover water and radioactivity mixture, g/m^3
- 1×10^{-2} • the airborne concentration in the contamination control envelope, g/m^3
- F_v • the ventilation flow rate, m^3/hr
- t • the duration of the cutting operation, hr
- F_c • the leak fraction of the contamination control envelope: if used, the fraction is 0.1; if not used, the fraction is 1.0.

The total release to the environment from underwater cutting tasks is calculated by multiplying the quantity of radioactive contamination entering the building airflow by the transmission factor of the building HEPA filter system.

Contaminated Concrete Removal. Two methods of concrete removal are considered in this study. They are: concrete spalling and blasting.

1. Concrete Spalling. Structural concrete that cannot be decontaminated to the unrestricted-use levels by surface cleaning methods must be mechanically removed. For this study, the first methods considered is the use of concrete spallers to remove these contaminated concrete areas. Airborne contamination control is assumed to be provided by the use of a vacuum exhaust system during this dusty task. Holes are drilled into the concrete surfaces for insertion of the concrete spaller. To calculate the airborne radioactivity in a building associated with the dust generated during drilling, the volume of concrete for each hole, the number of holes drilled, the radioactive contamination level and the effectiveness of the vacuum system must be determined. Each hole is assumed to be 5.1×10^{-2} m deep and remove a contaminated surface area of $5.0 \times 10^{-2} \text{ m}^2$.

For structural concrete with surface contamination only, the total radioactivity associated with drilling is found by multiplying the number of holes drilled by the surface area per hole by the surface contamination level. The surface contamination is assumed to be uniformly mixed in volume of dust associated with drilling the holes. A vacuum exhaust

system is used to collect the dust generated during drilling. This system is assumed to have an airflow of 1000 m³/hr and reduce the air concentration of dust by a factor of 0.3. The quantity of airborne radioactivity generated by drilling is given by Equation N.11.

$$Q_d = \frac{N_h (5 \times 10^{-2} \text{ m}^2/\text{hole}) C_s}{V_h \rho} F_v' t (0.3) 1 \times 10^{-1} \text{ g/m}^3 \quad (\text{N.11})$$

where:

- Q_d • the quantity of radioactivity made airborne during drilling, Ci
- N_h • the number of holes drilled
- 5×10^{-2} • the surface area removed per hole, m²/hole
- C_s • the concentration of surface radioactivity, Ci/m²
- V_h • the total volume of all holes drilled, m³
- ρ • the density of concrete, 2.5×10^3 Kg/m³
- F_v' • the vacuum system airflow rate, 1.0×10^3 m³/hr
- t • the duration of the drilling operation, hr
- 0.3 • airborne cleanup factor
- 1×10^{-1} • the airborne concentration in the vacuum system, g/m³.

The radioactivity released to the atmosphere is found by multiplying the quantity of radioactivity generated by the leakage factor from the control envelope (if one is used) and by the building HEPA filter transmission factor.

After the holes are drilled, the concrete spaller is inserted into the holes and the concrete surface is scarfed. The concrete fractures and spalls in large pieces with a relatively small amount of dust generated. Water sprays are assumed to be used to control the airborne dust during the concrete spalling. Thus, the airborne release from concrete spalling

system is used to collect the dust generated during drilling. This system is assumed to have an airflow of 1000 m³/hr and reduce the air concentration of dust by a factor of 0.3. The quantity of airborne radioactivity generated by drilling is given by Equation N.11.

$$Q_d = \frac{N_h (5 \times 10^{-2} \text{ m}^2/\text{hole}) C_s}{V_h \rho} F_v t (0.3) 1 \times 10^{-1} \text{ g/m}^3 \quad (\text{N.11})$$

where:

- Q_d • the quantity of radioactivity made airborne during drilling, Ci
- N_h • the number of holes drilled
- 5×10^{-2} • the surface area removed per hole, m²/hole
- C_s • the concentration of surface radioactivity, Ci/m²
- V_h • the total volume of all holes drilled, m³
- ρ • the density of concrete, 2.5×10^3 Kg/m³
- F_v • the vacuum system airflow rate, 1.0×10^3 m³/hr
- t • the duration of the drilling operation, hr
- 0.3 • airborne cleanup factor
- 1×10^{-1} • the airborne concentration in the vacuum system, g/m³.

The radioactivity released to the atmosphere is found by multiplying the quantity of radioactivity generated by the leakage factor from the control envelope (if one is used) and by the building HEPA filter transmission factor.

After the holes are drilled, the concrete spaller is inserted into the holes and the concrete surface is scarfed. The concrete fractures and spalls in large pieces with a relatively small amount of dust generated. Water sprays are assumed to be used to control the airborne dust during the concrete spalling. Thus, the airborne release from concrete spalling

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is assumed to be insignificant when compared to the drilling operations and is not considered further in this study.

2. Concrete Blasting. Activated concrete in the sacrificial shield is assumed to be removed by blasting tasks. Holes are drilled in the concrete for inserting blasting charges. The airborne release from drilling activated concrete is calculated by using Equation N.12.

$$Q'_d = \frac{V_h C_v}{\rho} F'_v t(0.3)(1 \times 10^{-1} \text{ g/m}^3) \quad (\text{N.12})$$

where:

Q'_d • the quantity of radioactivity made airborne during drilling activated concrete, Ci

C_v • the radioactivity concentration in the activated concrete, Ci/m³

V_h , ρ , F'_v , t , 0.3, and 1×10^{-1} are the same as defined for Equation N.11.

Blasting is assumed to create an airborne dust concentration of $1 \times 10^{-1} \text{ g/m}^3$ in the volume of the contamination control envelope. The airborne radioactivity generated during blasting is found using Equation N.13.

$$Q_b = \frac{N_b (1 \times 10^{-1} \text{ g/m}^3)}{\rho} V_e C_v \quad (\text{N.13})$$

where:

Q_b • the airborne radioactivity from blasting, Ci

N_b • the number of multiple-charge blasts

1×10^{-1} • the air concentration during blasting, g/m³

V_e • the volume of the contamination control envelope, $1 \times 10^3 \text{ m}^3$

ρ • the density of concrete, $2.5 \times 10^3 \text{ kg/m}^3$

C_v • the radioactivity concentration in the activated concrete, Ci/m³.

The amount of radioactivity released to the atmosphere from drilling or blasting activated concrete is found by multiplying Q_d' or Q_b by the leakage fraction from the control envelope (if one is used) and by the building HEPA filter transmission factor.

N.2.1.2 Atmospheric Releases from Immediate Dismantlement Tasks

Radioactivity releases to the atmosphere resulting from immediate dismantlement tasks at the reference BWR are calculated using the general methods and equations described in Section N.2.1.1. The operational data used in the equations is listed in Tables N.2-4, N.2-5, and N.2-6 for the Reactor Building, Turbine Generator Building, and the Radwaste and Control Building, respectively. These tables contain chronological listings of the immediate dismantlement tasks, necessary safety analysis data, operational times, the associated reference radionuclide inventory numbers, and the contamination levels present for each task.

The calculated atmospheric releases of radioactivity for immediate dismantlement tasks are listed in Tables N.2-7, N.2-8, and N.2-9 for the three buildings considered. These tables contain the calculated airborne radioactivity generation rates for each task in each building, as well as the total airborne radioactivity generated inside the buildings. The total airborne radioactivity inside a building during each task is next multiplied by the transmission factors for the control envelope and the building HEPA filter system to obtain the calculated atmospheric release. These atmospheric releases are used in radiation dose calculations to estimate the impact of routine immediate dismantlement tasks on the public. The calculated public radiation doses are presented in the following section. Again, the reference radionuclide inventory numbers listed in Tables N.2-7, N.2-8, and N.2-9 refer to the radionuclide inventories discussed in Appendix E.

N.2.1.3 Public Radiation Doses from Routine Immediate Dismantlement Tasks

Radiation doses to the public from atmospheric releases of radioactivity during routine immediate dismantlement tasks are calculated using the radiation dose models discussed in Appendix F. The first-year doses and the

TABLE N.2-4. Immediate Dismantlement Public Safety Analysis Data Summary for the Reactor Building and Primary Containment

Task	General Method	Operational Data				Comments	Operational Time (hr)	Reference Radionuclide Inventory Number(a)	Contamination Levels, Cs or Cy (C1/m ² or C1/m ³)		
1 Install HEPA Filters	N/A ^(b)	N/A				No Airborne Release Anticipated	N/A	N/A			
2 Comprehensive Radiation Survey	N/A	Surface Area (m ²)	Air Volume, V _{s1} (m ³)			Health Physics Technicians Tour Building, 3 Complete Surveys	6 0 x 10 ¹	5	<div><div>Cs</div><div>% of Surface Area</div><div>@ 2 5 x 10⁻³ (C1/m²) @ 2 5 x 10⁻² (C1/m²)</div></div>		
3 Discharge and Ship Fuel	FSAR Procedures	N/A				Store Fuel for a Minimum of 120 days and Ship Offsite-- Described in FSAR and not Analyzed here	N/A	N/A	N/A		
4 Remove Dryer and Separator	Plasma Torch and Arc Saw	Cut Length, L, (m)	Kerf Width, K, (m)	Cutting Rate (m/hr)	Volume of Liquid, V _L , (m ³)	Two Contaminated Surfaces per Cut, Cut Remotely Underwater in a Contamination Control Envelope Activated Material is 1 2 x 10 ⁻² m Thick	2 6 x 10 ¹	4	<div><div>Cs</div><div>C_s-1 C1/m²</div><div>3 6 x 10⁻¹ C1/m²</div></div>		
• Contaminated Dryer		9 0 x 10 ²	9 5 x 10 ⁻³	2 8 x 10 ¹	5 5 x 10 ²	3 2 x 10 ¹	1	<div><div>Cy</div><div>C_y-4 C1/m³</div><div>1 0 x 10⁻⁴ C1/m³</div></div>			
5 Drain Suppression Pool to Radwaste, Water-Jet Clean	Draining	N/A				No Airborne Release Anticipated	N/A	N/A	N/A		
	Water-Jet	Cleaning Rate (m ² /hr)	Surface Area (m ²)	Liquid Volume (m ³)			4 2 x 10 ¹	4	<div><div>Cs</div><div>C_s-2 C1/m²</div><div>5 0 x 10⁻² C1/m²</div></div>		
6 Remove Reactor Vessel Internals	Plasma Torch	Cut Length, L, (m)	Kerf Width, K, (m)	Cutting Rate (m/hr)	Volume of Liquid, V _L , (m ³)	Underwater Cutting in a Contamination Control Envelope--Activated Internals are 5 0 x 10 ⁻² m Thick	5 5 x 10 ²	4	<div><div>Cs</div><div>C_s-1 C1/m²</div><div>3 6 x 10⁻¹ C1/m²</div></div>		
• Activated		2 5 x 10 ³	6 4 x 10 ⁻³	4 8 x 10 ⁰	5 5 x 10 ²	5 2 x 10 ²	1	<div><div>Cy</div><div>C_y-6 C1/m³</div><div>1 7 x 10⁻⁶ C1/m³</div></div>			
7 Ship Activated Reactor Vessel Internals	Truck Transport	N/A				Public Exposures Discussed in Section N 5 1	N/A	1	N/A		
8 Drain Reactor Well Pool to Radwaste, Water-Jet Clean	Draining	N/A				No Airborne Release Anticipated	N/A	N/A	N/A		
	Water Jet	Cleaning Rate (m ² /hr)	Surface Area (m ²)	Liquid Volume (m ³)			4 0 x 10 ⁰	4	<div><div>Cs</div><div>C_s-2 C1/m²</div><div>5 0 x 10⁻² C1/m²</div></div>		
9 Chem Decon Reactor Water Recirc and Cleanup Systems	Loop Chem Decon	Release Fraction	Surface Area (m ²)	Liquid Volume V _L , (m ³)	Volume Released (m ³)	Assumed Release Fractions of 1% Liquid and 0 1% Spray	6 7 x 10 ²	4	<div><div>Cs</div><div>C_s-1 C1/m²</div><div>3 6 x 10⁻¹ C1/m²</div></div>		
• Liquid Leak		1 x 10 ⁻²	2 6 x 10 ³	6 1 x 10 ²	6 1 x 10 ⁰		6 7 x 10 ²	4	<div><div>Cy</div><div>C_y-1 C1/m³</div><div>3 6 x 10⁻¹ C1/m³</div></div>		
• Spray Leak		1 x 10 ⁻³	2 6 x 10 ³	6 1 x 10 ²	6 1 x 10 ⁻¹						
10 Cleanup, Stage, and Shield Hot Spots in Primary Containment	Sweeping	Surface Area (m ²)							<div><div>Cs</div><div>% of Surface Area</div><div>@ 2 5 x 10⁻³ (C1/m²) @ 2 5 x 10⁻² (C1/m²)</div></div>		
		2 5 x 10 ³					1 5 x 10 ¹	5	<div><div>Cy</div><div>50</div><div>50</div></div>		
	Vacuuming	2 5 x 10 ³					8 0 x 10 ⁰	5	<div><div>Cy</div><div>50</div><div>50</div></div>		
11 Enlarge Suppression Chamber Access	N/A	N/A				No Airborne Release Anticipated	N/A	N/A	N/A		
12 Remove Reactor Vessel	Oxyacetylene Torch/Arc Saw	Cut Length, L, (m)	Kerf Width, K, (m)	Cutting Rate (m/hr)	Volume of Liquid, V _L , (m ³)	Removal is Done in Air and Segmentation is Done Under Water Both are Done in Contamination Control Envelopes The Average Thickness for Activated Material is 0 18 m	3 9 x 10 ¹	2	<div><div>Cy</div><div>C_y-1 C1/m³</div><div>1 1 x 10⁻² C1/m³</div></div>		
• Remove Activated		1 7 x 10 ²	6 4 x 10 ⁻³	4 4 x 10 ⁰	1n Air	9 4 x 10 ¹	4	<div><div>Cs</div><div>C_s-1 C1/m²</div><div>3 6 x 10⁻¹ C1/m²</div></div>			
• Remove Contaminated		5 x 10 ²	6 4 x 10 ⁻³	5 4 x 10 ⁰	1n Air	9 2 x 10 ¹	2	<div><div>Cy</div><div>C_y-2 C1/m³</div><div>1 1 x 10⁻² C1/m³</div></div>			
• Segment Activated		4 0 x 10 ²	9 5 x 10 ⁻³	4 4 x 10 ⁰	5 5 x 10 ²	4 7 x 10 ¹	4	<div><div>Cs</div><div>C_s-1 C1/m²</div><div>3 6 x 10⁻¹ C1/m²</div></div>			
• Segment Contaminated		2 6 x 10 ²	6 4 x 10 ⁻³	5 4 x 10 ⁰	5 5 x 10 ²						
13 Ship Activated Reactor Vessel Segments	Truck Transport	N/A				Public Exposures Discussed in Section II 5 1	N/A	2	N/A		
14 Remove Primary Containment Piping and Equipment(c)	Oxyacetylene Torch and Power Hacksaw	Average O.D. (mm)	Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate, (m/hr)	Cut into 2 1-m Lengths, Large Diameter Valves are Separated, Most Operations are Remote	3 4 x 10 ²	4	<div><div>Cs</div><div>% of Cuts</div><div>Reactor Water @ 1 1 (C1/m²) Steam @ 5 x 10⁻³ (C1/m²) Condensate @ 5 x 10⁻² (C1/m²)</div></div>		
• Piping		60	4 7 x 10 ²	6 4 x 10 ⁻³	1 2 x 10 ¹		3 2 x 10 ²	4	<div><div>Cy</div><div>2</div><div>14</div></div>		
		152	4 5 x 10 ²	6 4 x 10 ⁻³	1 2 x 10 ¹		1 6 x 10 ²	4	<div><div>Cy</div><div>19</div><div>0</div></div>		
		356	2 3 x 10 ²	6 4 x 10 ⁻³	1 0 x 10 ¹		9 7 x 10 ²	4	<div><div>Cy</div><div>10</div><div>6</div></div>		
		533	1 4 x 10 ³	6 4 x 10 ⁻³	1 0 x 10 ⁰		7 5 x 10 ²	4	<div><div>Cy</div><div>3</div><div>66</div></div>		
		660	1 0 x 10 ³	6 4 x 10 ⁻³	8 0 x 10 ⁰						
• Whip Restraints and Hangers			1 4 x 10 ²	6 4 x 10 ⁻³	1 2 x 10 ¹	Cut During Piping Removal	9 5 x 10 ¹	5	<div><div>Cs</div><div>C_s-2 C1/m²</div><div>2 5 x 10⁻² C1/m²</div></div>		
15 Remove Sacrificial Shield and Radial Beams	Oxyacetylene Torch	Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)					<div><div>Cy</div><div>C_y-1 C1/m³</div><div>5 0 x 10⁻¹ C1/m³</div></div>		
• Activated Steel		2 0 x 10 ²	6 4 x 10 ⁻³	1 5 x 10 ¹			1 3 x 10 ¹	2	<div><div>Cs</div><div>C_s-2 C1/m²</div><div>2 5 x 10⁻² C1/m²</div></div>		
• Contaminated Steel		7 1 x 10 ²	6 4 x 10 ⁻³	1 5 x 10 ⁰			4 7 x 10 ²	5	<div><div>Cy</div><div>2 5 x 10⁻² C1/m³</div><div>2 5 x 10⁻² C1/m³</div></div>		
• Radial Beams		7 3 x 10 ²	6 4 x 10 ⁻³	2 9 x 10 ⁰			2 5 x 10 ²	5			
	Drilling	Number of Holes, N _h	Hole Surface Area A _s , (m ²)	Hole Depth (m)	Volume of Holes (m ³)	Contamination Control Envelope	1 5 x 10 ¹	3	<div><div>Cy</div><div>C_y-2 C1/m³</div><div>4 7 x 10⁻² C1/m³</div></div>		
• Concrete		84	6 6 x 10 ⁻³	6 1 x 10 ⁻¹	4 0 x 10 ⁻³						
	Blasting	Number of Charges	Number of Blasts			Air Concentration of 100 mg/m ³ in a Volume of 1000 m ³ with a Density of 2 5 x 10 ³ kg/m ³	N/A	3	<div><div>Cy</div><div>4 7 x 10⁻² C1/m³</div><div>4 7 x 10⁻² C1/m³</div></div>		

(a)These numbers refer to the Reference Radionuclide Inventories discussed in Appendix E
(b)N/A Not Applicable
(c)Primary containment equipment is included with Reactor Building Equipment for this analysis

TABLE N.2-4. (cont'd)

Task	General Method	Operational Data				Comments	Operational Time (hr)	Reference Radionuclide Inventory Number(a)	Contamination Levels, Cs or Cv (Ci/m ² or Ci/m ³)		
16 Remove Contaminated Concrete from Primary Containment	Drilling	Number of Holes, N _H	Hole Surface Area, A _s (m ²)	Hole Depth (m)	Volume of Holes (m ³)	4.0 x 10 ⁻⁶ m ³ of Waste per Hole 2.2 x 10 ³ m ² of Concrete Surface	3.8 x 10 ³	5	2.5 x 10 ⁻² Ci/m ²		
					Volume of Waste (m ³)						
17 Remove HVAC and Electrical Systems from Primary Containment	Spalling	N/A(b)			1.8 x 10 ²	No Airborne Release Anticipated	N/A	N/A	N/A		
		Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)							
• HVAC • Electrical Equipment	Oxyacetylene Torch	2.0 x 10 ³	6.4 x 10 ⁻³	5.4 x 10 ⁰		All HVAC and Electrical Equipment in Primary Containment is Assumed to be Removed	1.1 x 10 ²	5	2.5 x 10 ⁻² Ci/m ²		
		7.6 x 10 ²	6.4 x 10 ⁻³	3.1 x 10 ¹			1.2 x 10 ²	5	2.5 x 10 ⁻² Ci/m ²		
18 Drain Contaminated Systems to Pad/aste	Draining	N/A				No Airborne Release Anticipated	N/A	N/A	N/A		
19 Chem Decon Residual Heat Removal, Low- and High-Pressure Core Spray Systems	Loop Chem Decon	Release Fraction	Surface Area, A _s (m ²)	Liquid Volume, V _L (m ³)	Volume Released (m ³)	Assumed Release Fractions of 1% Liquid and 0.1% Spray	8.9 x 10 ²	4	3.6 x 10 ⁻¹ Ci/m ² 3.6 x 10 ⁻¹ Ci/m ²		
		1 x 10 ⁻² 1 x 10 ⁻³	1.9 x 10 ³ 1.9 x 10 ³	1.9 x 10 ² 1.9 x 10 ²	1.9 x 10 ⁰ 1.9 x 10 ⁻¹						
20 Remove Reactor Building Piping	Oxyacetylene Torch and Power Hacksaw	Average OD (mm)	Cut Length, L (m)	Kerf Width, K (m)	Cut Rate (m/hr)	Large Diameter Valves are Separated, Most Operations are Remote	6.8 x 10 ²	4	Reactor Water @ 1.1 (Ci/m ²) Steam @ 5 x 10 ⁻³ (Ci/m ²) Condensate @ 5 x 10 (Ci/m ²)		
		60 152 356 533 660	9.6 x 10 ² 1.4 x 10 ² 4.4 x 10 ² 1.2 x 10 ³ 9.8 x 10 ¹	6.4 x 10 ⁻³ 6.4 x 10 ⁻³ 6.4 x 10 ⁻³ 6.4 x 10 ⁻³ 6.4 x 10 ⁻³	1.2 x 10 ¹ 1.2 x 10 ¹ 1.0 x 10 ¹ 1.0 x 10 ¹ 8.0 x 10 ⁰						
21 Drain Dryer and Separator Pool to Radwaste, Water-Jet Clean	Draining	N/A				No Airborne Release Anticipated	N/A	N/A	N/A		
		Cleaning Rate (m ² /hr)	Surface Area (m ²)	Liquid Volume (m ³)							
22 Chemical Decon Drain Systems	Water Jet	4.6 x 10 ¹	5.3 x 10 ²	1.6 x 10 ¹		Cleaning Dryer and Separator Pool Surfaces	1.2 x 10 ¹	4	5.0 x 10 ⁻² Ci/m ²		
• Liquid Leak • Spray Leak	Single-Pass Chem Decon	1 x 10 ⁻² 1 x 10 ⁻³	1.6 x 10 ³ 1.6 x 10 ³	2.8 x 10 ¹ 2.8 x 10 ¹	2.8 x 10 ⁻¹ 2.8 x 10 ⁻²	Assumed Release Fractions of 1% Liquid and 0.1% Spray of Acid Wash Solution	5.2 x 10 ¹ 5.2 x 10 ¹	4 4	5.0 x 10 ⁻² Ci/m ² 5.0 x 10 ⁻² Ci/m ²		
23 Drain Spent Fuel Pool to Radwaste, Water-Jet Clean	Draining	N/A				No Airborne Release Anticipated	N/A	N/A	N/A		
		Cleaning Rate (m ² /hr)	Surface Area (m ²)	Liquid Volume (m ³)							
24 Chemical Decon Fuel Pool Cooling and Cleanup System	Water Jet	4.6 x 10 ¹	6.1 x 10 ²	1.3 x 10 ¹		Cleaning Spent Fuel Pool Surfaces	1.3 x 10 ¹	4	5.0 x 10 ⁻² Ci/m ²		
• Liquid Leak • Spray Leak	Loop Chemical Decon	1 x 10 ⁻² 1 x 10 ⁻³	7.4 x 10 ² 7.4 x 10 ²	1.2 x 10 ¹ 1.2 x 10 ¹	1.2 x 10 ⁻¹ 1.2 x 10 ⁻²	Assumed Release Fractions of 1% Liquid and 0.1% Spray	1.2 x 10 ² 1.2 x 10 ²	4 4	5.0 x 10 ⁻² Ci/m ² 5.0 x 10 ⁻² Ci/m ²		
25 Remove Reactor Building Equipment(c)	Oxyacetylene Torch	Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)		Segmented to Fit Ship-ment Boxes	3.1 x 10 ²	4	5.0 x 10 ⁻² Ci/m ²		
		2.0 x 10 ³	6.4 x 10 ⁻³	4.5 x 10 ⁰							
26 Remove Liners from Spent Fuel Pool, Reactor Well, and Dryer and Separator Pool	Plasma-arc Torch	Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)		Cut into 2.3 x 1.1 m Segments for Packaging	1.5 x 10 ²	4	5.0 x 10 ⁻² Ci/m ²		
		2.2 x 10 ³	6.4 x 10 ⁻³	1.5 x 10 ¹							
27 Remove Reactor Building Contaminated Concrete	Drilling	Number of Holes, N _H	Hole Surface Area, A _s (m ²)	Volume of Holes (m ³)		4.0 x 10 ⁻⁶ m ³ of Waste per Hole, 2.9 x 10 ³ m ² of Concrete Surface	1.9 x 10 ³	5	2.5 x 10 ⁻³ Ci/m ²		
		3.5 x 10 ⁴	2.8 x 10 ⁰	1.4 x 10 ⁻¹							
23 Remove HVAC and Electrical Systems from Reactor Building	Spalling	N/A			1.5 x 10 ²	No Airborne Release Anticipated	N/A	N/A	N/A		
		Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)							
• HVAC • Electrical Equipment	Oxyacetylene Torch	7.0 x 10 ³	6.4 x 10 ⁻³	5.4 x 10 ⁰		All HVAC Exhaust Equipment and Only 5% of the Electrical Cable Trays and Bracing are Assumed to be Removed	3.7 x 10 ²	5	2.5 x 10 ⁻³ Ci/m ²		
		1.6 x 10 ²	6.4 x 10 ⁻³	3.1 x 10 ¹			2.4 x 10 ¹	5	2.5 x 10 ⁻³ Ci/m ²		
29 Final Radiation Survey	N/A	N/A				Assumed to be at Acceptable Release Levels--No Calculation	N/A	N/A	N/A		

(a)These numbers refer to the Reference Radionuclide Inventories discussed in Appendix E
(b)N/A = Not Applicable
(c)Primary containment equipment is included with Reactor Building Equipment for this analysis

TABLE N.2-5. Immediate Dismantlement Public Safety Analysis Data Summary for the Turbine Generator Building

Task	General Method	Operational Data			Comments	Operational Time (hr)	Reference Radionuclide Inventory Number ^(a)	Contamination Levels, C _s (Ci/m ²)		
1. Install HEPA Filters	N/A ^(b)	N/A			No Airborne Release Anticipated	N/A	N/A	N/A		
2. Comprehensive Radiation Survey		Surface Area (m ²)	Air Volume, V _s (m ³)							
	N/A	5.1 x 10 ³	5.1 x 10 ³		Health Physics Technicians Tour Building, 3 Complete Surveys	6.0 x 10 ¹	5	2.5 x 10 ⁻³		
3. Cleanup and Stage		Surface Area (m ²)	Air Volume, V _s (m ³)							
	Sweeping	1.9 x 10 ³	1.9 x 10 ³		Four-Man Crews	1.1 x 10 ¹	5	2.5 x 10 ⁻³		
	Vacuuming	1.9 x 10 ³				5.0 x 10 ⁰	5	2.5 x 10 ⁻³		
4. Remove Turbine		Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)						
	Oxyacetylene Torch	2.4 x 10 ³	6.4 x 10 ⁻³	1.0 x 10 ¹	Covers, Rotors, and Bases are Cut in a Contamination Control Envelope	2.3 x 10 ²	4	5.0 x 10 ⁻⁴		
5. Drain Contaminated Systems to Radwaste; Water-Jet Clean Condensate Storage Tanks	Draining	N/A			No Airborne Release Anticipated	N/A	N/A	N/A		
		Cleaning Rate (m ² /hr)	Surface Area (m ²)	Liquid Volume (m ³)						
	Water Jet	4.6 x 10 ¹	1.7 x 10 ³	5.1 x 10 ¹	Internal Cleaning of Condensate Storage Tanks	3.8 x 10 ¹	4	5.0 x 10 ⁻²		
6. Remove Condenser		Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)						
	Oxyacetylene Torch	2.6 x 10 ⁴	6.4 x 10 ⁻³	6.5 x 10 ⁰	Multiple Cuts of Tubes, Support Plates, and Outer Skin Inside a Contamination Control Envelope	4.0 x 10 ³	4	<div><div>% of Cuts @ 5 x 10⁻³</div><div>50</div></div> <div><div>% of Cuts @ 5 x 10⁻²</div><div>50</div></div>		
7. Drain Condenser to Radwaste; Water-Jet Clean	Draining	N/A			No Airborne Release Anticipated	N/A	N/A	N/A		
		Cleaning Rate (m ² /hr)	Surface Area (m ²)	Liquid Volume (m ³)						
	Water Jet	4.6 x 10 ¹	4.4 x 10 ⁴	1.3 x 10 ³	Surfaces are Washed, ½ Surface Cleaned Using a Contamination Control Envelope	9.4 x 10 ²	4	5.0 x 10 ⁻²		
8. Chemical Decon. Drain Systems		Release Fraction	Surface Area, A _s (m ²)	Liquid Volume, V _L (m ³)	Volume Released (m ³)					
• Liquid Leak	Single-Pass Chem. Decon.	1 x 10 ⁻²	2.2 x 10 ³	4.0 x 10 ¹	4.0 x 10 ⁻¹	Assumed Release Fractions of 1% Liquid and 0.1% Spray	7.3 x 10 ¹	4	5.0 x 10 ⁻²	
• Spray Leak		1 x 10 ⁻³	2.2 x 10 ³	4.0 x 10 ¹	4.0 x 10 ⁻²		7.3 x 10 ¹	4	5.0 x 10 ⁻²	
9. Remove Piping		Average O.D. (mm)	Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)					
	Oxyacetylene	60	5.0 x 10 ²	6.4 x 10 ⁻³	1.2 x 10 ¹	Pipe is Cut to Fit 2.3 x 1.1 m Boxes, Large Valves are Removed Separately	4.2 x 10 ²	4	<div><div>% of Cuts Steam @ 5 x 10⁻³</div><div>34</div></div> <div><div>% of Cuts Condensate @ 5 x 10⁻³</div><div>66</div></div>	
	Torches and Power	152	1.5 x 10 ³	6.4 x 10 ⁻³	1.2 x 10 ¹		1.1 x 10 ³	4	51	
	Hacksaws	356	1.6 x 10 ³	6.4 x 10 ⁻³	1.0 x 10 ¹		9.0 x 10 ²	4	35	
		533	1.7 x 10 ³	6.4 x 10 ⁻³	1.0 x 10 ¹		1.4 x 10 ³	4	59	
		660	7.4 x 10 ³	6.4 x 10 ⁻³	8.0 x 10 ⁰		4.1 x 10 ²	4	83	
		1 200	1.5 x 10 ³	6.4 x 10 ⁻³	8.0 x 10 ⁰		6.0 x 10 ²	4	79	
10. Remove Equipment		Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)						
• Misc. Equipment	Oxyacetylene Torch	3.2 x 10 ²	6.4 x 10 ⁻³	1.2 x 10 ¹	Sectioned and Sealed for Shipping	2.7 x 10 ¹	4	5.0 x 10 ⁻²		
• LP Feedwater Heaters		6.7 x 10 ³	6.4 x 10 ⁻³	2.3 x 10 ¹		2.8 x 10 ²	4	5.0 x 10 ⁻³		
• HP Feedwater Heaters		2.5 x 10 ²	6.4 x 10 ⁻³	7.4 x 10 ⁰		3.4 x 10 ¹	4	5.0 x 10 ⁻³		
• Moisture Separator Reheater		3.9 x 10 ³	6.4 x 10 ⁻³	2.4 x 10 ¹		1.7 x 10 ²	4	5.0 x 10 ⁻³		
• Condensate Storage Tanks		2.5 x 10 ³	6.4 x 10 ⁻³	2.0 x 10 ¹		1.2 x 10 ²	4	5.0 x 10 ⁻³		
11. Remove Contaminated Concrete		Number of Holes, N _H	Hole Surface Area (m ²)	Hole Volume V _H (m ³)						
	Drilling	2.3 x 10 ⁴	1.8 x 10 ⁰	9.1 x 10 ⁻²	4.0 x 10 ⁻⁶ m ³ of Waste per Hole, 200 Holes/day, Floor Area = 1.9 x 10 ³ m ²	1.3 x 10 ³	5	2.5 x 10 ⁻³		
				Volume of Waste (m ³)						
	Spalling	N/A	N/A	1.2 x 10 ²	No Airborne Release Anticipated	N/A	N/A	N/A		
12. Remove HVAC and Electrical Systems		Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)						
	Oxyacetylene Torch	9.2 x 10 ³	6.4 x 10 ⁻³	6.2 x 10 ¹	Partial Cutting in Place	1.5 x 10 ²	5	2.5 x 10 ⁻³		
13. Final Radiation Survey	N/A	N/A			Assumed to be at Acceptable Release Levels--No Calculation	N/A	N/A	N/A		

(a)These numbers refer to the Reference Radionuclide inventories discussed in Appendix E.
(b)N/A = Not Applicable.

TABLE N.2-6. Public Safety Analysis Data for Immediate Dismantlement of the Radwaste and Control Building

Task	General Method	Operational Data				Comments	Operational Time (hr)	Reference Radionuclide Inventory Number (4)	Contamination Levels, C _s (Ci/m ²)	
1 Comprehensive Radiation Survey	Health Physics Technicians	Surface Area (m ²)	Air Volume, V _s (m ³)			Radiation Monitors Tour Building 3 Complete Surveys	6.0 x 10 ¹	5	% of Surface Area @ 2.5 x 10 ⁻³ @ 2.5 x 10 ⁻²	
		5.1 x 10 ³	5.1 x 10 ³	30 70						
2 General Cleanup	Sweeping	Surface Area (m ²)				1/3 of Building Area Swept	3.0 x 10 ¹	5		
		5.1 x 10 ³	30 70							
3 Chemical Decon Drain Systems	Single-Pass Chem Decon	Release Fraction	Surface Area, A _s (m ²)	Liquid Volume, V _s (m ³)	Volume Released (m ³)	Assumed Release Fractions of 1% Liquid and 0.1% Spray	2.3 x 10 ¹ 2.3 x 10 ¹	4 4	5.0 x 10 ⁻² 5.0 x 10 ⁻²	
• Liquid Leak • Spray Leak		1 x 10 ⁻² 1 x 10 ⁻³	6.8 x 10 ² 6.8 x 10 ²	1.2 x 10 ¹ 1.2 x 10 ¹	1.2 x 10 ⁻¹ 1.2 x 10 ⁻²					
4 Chemical Decon Equipment, Water-Jet Clean	Water-Jet	Cleaning Rate (m ² /hr)	Surface Area, A _s (m ²)	Liquid Volume, V _L (m ³)	Volume Released (m ³)	Exterior Surfaces Cleaned	4.9 x 10 ¹	4	% of Surface Area 2.5 x 10 ⁻² @ 5 x 10 ⁰	
		4.6 x 10 ¹	2.3 x 10 ³	6.8 x 10 ¹					55 45	
• Liquid Leak • Spray Leak	Loop Chem Decon	Release Fraction	Surface Area, A _s (m ²)	Liquid Volume, V _L (m ³)	Volume Released (m ³)	Assumed Release Fractions of 1% Liquid and 0.1% Spray	4.0 x 10 ¹ 4.0 x 10 ¹	4 4	55 45 55 45	
		1 x 10 ⁻² 1 x 10 ⁻³	1.4 x 10 ³ 1.4 x 10 ³	1.5 x 10 ² 1.5 x 10 ²	1.5 x 10 ⁰ 1.5 x 10 ⁻¹					
5 Remove Piping	Oxyacetylene and Plasma Torches, Power Hacksaws	Average O.D. (mm)	Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)	Cut into 2.1 m Lengths	4.0 x 10 ² 1.0 x 10 ² 1.0 x 10 ² 1.6 x 10 ² 7.0 x 10 ¹ 3.8 x 10 ¹	4 4 4 4 4 4	Reactor Water 1.1 Steam @ 5 x 10 ⁻³ Cond 5 x 10 ⁻²	
		60	4.6 x 10 ²	6.4 x 10 ⁻³	1.2 x 10 ¹				6 10 84	
		152	1.3 x 10 ³	6.4 x 10 ⁻³	1.2 x 10 ¹				7 15 78	
		356	2.6 x 10 ³	6.4 x 10 ⁻³	1.0 x 10 ¹				0 1 99	
		533	1.9 x 10 ³	6.4 x 10 ⁻³	1.0 x 10 ¹				0 0 100	
		914	1.6 x 10 ²	6.4 x 10 ⁻³	8.0 x 10 ⁰				0 0 100	
6 Install Temporary Radwaste System	N/A	N/A				No Airborne Release Anticipated	N/A	N/A	N/A	
7 Remove Equipment	Oxyacetylene Torch	Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)			3.3 x 10 ²	4	% of Surface Area @ 5 x 10 ⁻³ @ 5 x 10 ⁻¹	
		6.2 x 10 ³	6.4 x 10 ⁻³	1.9 x 10 ¹	Averaged Wall Thickness and Cutting Rate				55 45	
8 Remove Contaminated Concrete	Drilling	Number of Holes, N _H	Hole Surface Area (m ²)	Hole Volume, V _H (m ³)			6.6 x 10 ²	5	% of Surface Area @ 2.5 x 10 ⁻³ @ 2.5 x 10 ⁻²	
		2.4 x 10 ⁴	1.9 x 10 ⁰	9.6 x 10 ⁻²	In a Contamination Control Envelope, 4.0 x 10 ⁻⁶ m ³ of Waste per Hole are Drilled at 200 Holes/day (Floor Area = 0 x 10 ³ m ²)				30 70	
	Spalling	N/A	N/A	1.0 x 10 ⁰	No Airborne Release Anticipated for Spalling		3.8 x 10 ²	5	30 70	
9 Remove Misc Steel Structures	Oxyacetylene Torch and Power Hacksaws	Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)			1.3 x 10 ¹	5	2.5 x 10 ⁻³	
		2.5 x 10 ²	6.4 x 10 ⁻³	1.9 x 10 ¹	Cut to Fit 2.3 x 1.1-m Boxes					
10 Remove HVAC and Electrical Systems	Oxyacetylene Torch Power Hacksaws	Cut Length, L (m)	Kerf Width, K (m)	Cutting Rate (m/hr)			1.2 x 10 ² 2.2 x 10 ¹	5 5	2.5 x 10 ⁻³ 2.5 x 10 ⁻³	
		8.5 x 10 ³ 9.5 x 10 ²	6.4 x 10 ⁻³ 6.4 x 10 ⁻³	7.2 x 10 ¹ 3.6 x 10 ¹	All HVAC Exhaust Equipment and Only 5% of the Electrical Cable Trays and Bracing are Assumed to be Removed					
11 Final Radiation	N/A	N/A				Assumed to be at Acceptable Release Levels- No Release Calculated	N/A	N/A	N/A	

(a) These numbers refer to the reference radionuclide inventories discussed in Appendix E
(b) N/A = Not Applicable

TABLE N.2-7. Calculated Atmospheric Releases for Immediate Dismantlement of the Reactor Building and Primary Containment

Task	General Method	Operational Time (hr)	Reference Radionuclide Inventory Number(a)	Airborne Generation Rate Inside Building or Control Envelope (Ci/hr)	Total Airborne Radioactivity Inside Building or Control Envelope (Ci)	Contamination Control System Transmission Factor(b)	Total Atmospheric Release (Ci/yr)(c)
1. Install HEPA Filters	N/A(d)	N/A	N/A	N/A	N/A	N/A	N/A
2. Comprehensive Radiation Survey	Radiation Monitors	6.0 x 10 ¹	5	2.2 x 10 ⁻⁴	1.4 x 10 ⁻²	5 x 10 ⁻⁴	6.8 x 10 ⁻⁶
3. Discharge and Ship Fuel	FSAR Procedures	1.2 x 10 ²	N/A	N/A	N/A	5 x 10 ⁻⁴	N/A
4. Remove Dryer and Separator <ul style="list-style-type: none">• Contaminated Dryer• Activated Separator	Plasma Torch and Arc Saw	2.6 x 10 ¹ 3.2 x 10 ¹	4 1	2.0 x 10 ⁻⁷ 5.6 x 10 ⁻⁵	5.1 x 10 ⁻⁶ 1.8 x 10 ⁻³	5 x 10 ⁻⁵ 5 x 10 ⁻⁵	2.6 x 10 ⁻¹⁰ 9.0 x 10 ⁻⁸
5. Drain Suppression Pool to Radwaste; Water-Jet Clean	Draining Water-Jet	N/A 4.2 x 10 ¹	N/A 4	N/A 2.0 x 10 ⁻⁴	N/A 8.5 x 10 ⁻³	5 x 10 ⁻⁴ 5 x 10 ⁻⁴	N/A 4.2 x 10 ⁻⁶
6. Remove Reactor Vessel Internals <ul style="list-style-type: none">• Contaminated• Activated	Plasma Torch Plasma Torch	5.5 x 10 ² 5.2 x 10 ²	4 1	4.0 x 10 ⁻⁸ 7.4 x 10 ⁻²	2.2 x 10 ⁻⁵ 3.9 x 10 ¹	5 x 10 ⁻⁵ 5 x 10 ⁻⁵	1.1 x 10 ⁻⁹ 1.9 x 10 ⁻³
7. Ship Activated Reactor Vessel Internals	Truck Transport	N/A	1	See Section N.5	N/A	N/A	See Section N.5
8. Drain Reactor Well Draining Pool to Radwaste; Water-Jet Clean	Draining Water-Jet	N/A 4.0 x 10 ⁰	N/A 4	N/A 2.0 x 10 ⁻⁴	N/A 8.1 x 10 ⁻⁴	N/A 5 x 10 ⁻⁴	N/A 4.0 x 10 ⁻⁷
9. Chem. Decon Reactor Water Circ. and Cleanup Systems <ul style="list-style-type: none">• Liquid Leak• Spray Leak	Loop Chem. Decon.	6.7 x 10 ² 6.7 x 10 ²	4 4	2.5 x 10 ⁻⁷ 3.8 x 10 ⁻⁶	1.7 x 10 ⁻⁴ 2.5 x 10 ⁻³	5 x 10 ⁻⁴ 5 x 10 ⁻⁴	8.3 x 10 ⁻⁸ 1.3 x 10 ⁻⁶
10. Cleanup, Stage and Shield Hot Spots in Primary Containment	Sweeping Vacuuming	1.5 x 10 ¹ 8.0 x 10 ⁰	5 5	1.1 x 10 ⁻³ 1.1 x 10 ⁻³	1.7 x 10 ⁻² 8.6 x 10 ⁻³	5 x 10 ⁻⁴ 5 x 10 ⁻⁴	8.6 x 10 ⁻⁶ 4.3 x 10 ⁻⁶
11. Enlarge Suppression Chamber Access	N/A	N/A	N/A	N/A	N/A	5 x 10 ⁻⁴	N/A
12. Remove Reactor Vessel <ul style="list-style-type: none">• Remove Activated• Remove Contaminated• Segment Activated• Segment Contaminated	Oxyacetylene Torch/Arc Saw	3.9 x 10 ¹ 9.4 x 10 ¹ 9.2 x 10 ¹ 4.7 x 10 ¹	2 4 2 4	5.5 x 10 ⁻¹ 2.5 x 10 ⁻² 4.1 x 10 ⁻⁶ 6.5 x 10 ⁻⁸	2.2 x 10 ¹ 2.4 x 10 ⁰ 3.8 x 10 ⁻⁴ 3.1 x 10 ⁻⁶	5 x 10 ⁻⁵ 5 x 10 ⁻⁵ 5 x 10 ⁻⁵ 5 x 10 ⁻⁵	1.1 x 10 ⁻³ 1.2 x 10 ⁻⁴ 1.9 x 10 ⁻⁸ 1.5 x 10 ⁻¹⁰
13. Ship Activated Reactor Vessel Segments	Truck Transport	N/A	2	See Section N.5	N/A	N/A	See Section N.5
14. Remove Primary Containment Piping and Equipment(e)	Oxyacetylene and Plasma Torches Power Hacksaws (O.D.)	60 mm 152 mm 356 mm 533 mm 660 mm Total	4 4 4 4 4 5	5.3 x 10 ⁻⁴ 2.2 x 10 ⁻³ 1.4 x 10 ⁻³ 7.3 x 10 ⁻⁴ 1.7 x 10 ⁻⁴ 4.7 x 10 ⁻⁴	1.8 x 10 ⁻¹ 7.0 x 10 ⁻¹ 2.3 x 10 ⁻¹ 7.1 x 10 ⁻¹ 1.3 x 10 ⁻¹ 4.5 x 10 ⁻²	5 x 10 ⁻⁴ 5 x 10 ⁻⁴ 5 x 10 ⁻⁴ 5 x 10 ⁻⁴ 5 x 10 ⁻⁴ 5 x 10 ⁻⁴	9.0 x 10 ⁻⁵ 3.5 x 10 ⁻⁴ 1.1 x 10 ⁻⁴ 3.6 x 10 ⁻⁵ 6.5 x 10 ⁻⁵ 2.2 x 10 ⁻⁵
15. Remove Sacrificial Shield and Radial Beams <ul style="list-style-type: none">• Activated• Contaminated• Radial Beams• Concrete	Oxyacetylene Torch Drilling Blasting	1.3 x 10 ¹ 4.7 x 10 ² 2.5 x 10 ¹ 1.5 x 10 ¹ N/A	2 5 5 3 3	3.4 x 10 ⁻¹ 2.5 x 10 ⁻³ 4.6 x 10 ⁻⁹ 2.3 x 10 ⁻⁹ 1.9 x 10 ⁻⁶	4.5 x 10 ⁰ 1.1 x 10 ⁻¹ 1.2 x 10 ⁻⁸ 3.4 x 10 ⁻⁶ 9.4 x 10 ⁻⁶	5 x 10 ⁻⁴ 5 x 10 ⁻⁴ 5 x 10 ⁻⁵ 5 x 10 ⁻⁵ 5 x 10 ⁻⁵	2.2 x 10 ⁻³ 5.5 x 10 ⁻⁵ 5.8 x 10 ⁻¹² 1.7 x 10 ⁻¹⁰ 4.7 x 10 ⁻¹⁰
16. Remove Contaminated Concrete from Primary Containment	Drilling Spalling	3.8 x 10 ³ 6.6 x 10 ³	5 5	6.0 x 10 ⁻⁶ N/A	2.3 x 10 ⁻² N/A	5 x 10 ⁻⁴ N/A	1.1 x 10 ⁻⁵ N/A
17. Remove HVAC and Electrical Systems from Primary Containment <ul style="list-style-type: none">• HVAC• Electrical Equipment	Oxyacetylene Torch	1.1 x 10 ² 1.2 x 10 ²	5 5	2.9 x 10 ⁻³ 1.0 x 10 ⁻³	3.2 x 10 ⁻¹ 1.2 x 10 ⁻¹	5 x 10 ⁻⁴ 5 x 10 ⁻⁴	1.6 x 10 ⁻⁴ 6.1 x 10 ⁻⁵
18. Drain Contaminated Systems to Radwaste	Draining	N/A	N/A	N/A	N/A	5 x 10 ⁻⁴	N/A
19. Chem. Decon. Residual Heat Removal, Low- and High-Pressure Core Spray Systems <ul style="list-style-type: none">• Liquid Leak• Spray Leak	Loop Chem. Decon.	8.9 x 10 ² 8.9 x 10 ²	4 4	1.4 x 10 ⁻⁷ 2.0 x 10 ⁻⁶	1.2 x 10 ⁻⁴ 1.8 x 10 ⁻³	5 x 10 ⁻⁴ 5 x 10 ⁻⁴	6.0 x 10 ⁻⁸ 9.0 x 10 ⁻⁷
20. Remove Reactor Building Piping	Oxyacetylene and Plasma Torches Power Hacksaws (O.D.)	60 mm 152 mm 356 mm 533 mm 660 mm Total	4 4 4 4 4 4	5.3 x 10 ⁻⁴ 2.2 x 10 ⁻³ 1.4 x 10 ⁻³ 7.1 x 10 ⁻⁴ 1.9 x 10 ⁻⁴ 1.9 x 10 ⁻⁴	3.6 x 10 ⁻¹ 2.2 x 10 ⁰ 4.4 x 10 ⁻¹ 6.0 x 10 ⁻¹ 1.3 x 10 ⁻² 1.3 x 10 ⁻²	5 x 10 ⁻⁴ 5 x 10 ⁻⁴ 5 x 10 ⁻⁴ 5 x 10 ⁻⁴ 5 x 10 ⁻⁴ 5 x 10 ⁻⁴	1.8 x 10 ⁻⁴ 1.1 x 10 ⁻³ 2.2 x 10 ⁻⁴ 3.0 x 10 ⁻⁴ 6.5 x 10 ⁻⁶ 1.8 x 10 ⁻³
21. Drain Dryer and Separator Pool to Radwaste; Water-Jet Clean	Draining Water-Jet	N/A 1.2 x 10 ¹	N/A 4	N/A 2.0 x 10 ⁻⁴	N/A 2.4 x 10 ⁻³	N/A 5 x 10 ⁻⁴	N/A 1.2 x 10 ⁻⁶
22. Chemical Decon Drain Systems <ul style="list-style-type: none">• Liquid Leak• Spray Leak	Single-Pass Chem. Decon.	5.2 x 10 ¹ 5.2 x 10 ¹	4 4	2.8 x 10 ⁻⁷ 4.2 x 10 ⁻⁶	1.4 x 10 ⁻⁵ 2.2 x 10 ⁻⁴	5 x 10 ⁻⁴ 5 x 10 ⁻⁴	1.4 x 10 ⁻¹⁰ 1.1 x 10 ⁻⁷
23. Drain Spent Fuel Pool to Radwaste; Water-Jet Clean	Draining Water-Jet	N/A 1.3 x 10 ¹	N/A 4	N/A 2.0 x 10 ⁻⁴	N/A 1.0 x 10 ⁻⁴	5 x 10 ⁻⁴ 5 x 10 ⁻⁴	N/A 1.3 x 10 ⁻⁶
24. Chemical Decon Fuel Pool Cooling and Cleanup System <ul style="list-style-type: none">• Liquid Leak• Spray Leak	Loop Chemical Decon.	1.2 x 10 ² 1.2 x 10 ²	4 4	5.6 x 10 ⁻⁸ 8.3 x 10 ⁻⁷	6.7 x 10 ⁻⁶ 1.0 x 10 ⁻⁴	4 x 10 ⁻⁴ 5 x 10 ⁻⁴	3.3 x 10 ⁻⁹ 5.0 x 10 ⁻⁸
25. Remove Reactor Building Equipment(e)	Oxyacetylene Torch	3.1 x 10 ²	4	2.1 x 10 ⁻³	6.4 x 10 ⁻¹	5 x 10 ⁻⁴	3.2 x 10 ⁻⁴
26. Remove Liners from Spent Fuel Pool, Reactor Well, and Dryer and Separator Pool	Plasma Torch	1.5 x 10 ²	4	4.7 x 10 ⁻³	7.0 x 10 ⁻¹	5 x 10 ⁻⁴	3.5 x 10 ⁻⁴
27. Remove Reactor Building Contaminated Concrete	Drilling Spalling	1.9 x 10 ³ 1.1 x 10 ³	5 5	6.0 x 10 ⁻⁷ N/A	1.1 x 10 ⁻³ N/A	5 x 10 ⁻⁴ N/A	5.7 x 10 ⁻⁷ N/A
28. Remove HVAC and Electrical Systems from Reactor Building <ul style="list-style-type: none">• HVAC• Electrical Equipment	Oxyacetylene Torch	3.7 x 10 ² 2.4 x 10 ¹	5 5	3.0 x 10 ⁻⁴ 1.1 x 10 ⁻⁴	1.1 x 10 ⁻¹ 2.6 x 10 ⁻³	5 x 10 ⁻⁴ 5 x 10 ⁻⁴	5.5 x 10 ⁻⁴ 1.3 x 10 ⁻⁶
29. Final Radiation Survey	N/A	N/A	N/A	N/A	N/A	N/A	N/A

(a)These numbers refer to the Reference Radionuclide Inventories discussed in Appendix E.
(b)The assumed building HEPA filter transmission factor is 5 x 10⁻⁴. The combined transmission factor for a contamination control envelope and the building HEPA filter is 5 x 10⁻⁵.
(c)Dose calculations are made based on annual releases for comparison purposes.
(d)N/A = Not Applicable.
(e)Primary containment piping is included with reactor building equipment for this analysis.

TABLE N.2-8. Calculated Atmospheric Releases for Immediate Dismantlement of the Turbine Generator Building

Task	General Method	Operational Time (hr)	Reference Radionuclide Inventory Number(a)	Airborne Generation Rate Inside Building or Control Envelope (Ci/hr)	Total Airborne Radioactivity Inside Building or Control Envelope (Ci)	Contamination Control System Transmission Factor(b)	Total Atmospheric Release (Ci/yr)(c)
1. Install HEPA Filters	N/A(d)	N/A	N/A	N/A	N/A	N/A	N/A
2. Comprehensive Radiation Survey	Radiation Monitors	6.0×10^1	5	6.5×10^{-5}	3.9×10^{-3}	5×10^{-4}	1.9×10^{-6}
3. Cleanup and Stage	Sweeping Vacuuming	1.1×10^1 5.0×10^0	5 5	2.2×10^{-4} 2.4×10^{-4}	2.4×10^{-3} 1.2×10^{-3}	5×10^{-4} 5×10^{-4}	1.2×10^{-6} 5.9×10^{-7}
4. Remove Turbine	Oxyacetylene Torch	2.3×10^2	4	5.6×10^{-6}	1.3×10^{-3}	5×10^{-4}	6.4×10^{-7}
5. Drain Contaminated Systems to Radwaste; Water-Jet Clean Condensate Storage Tanks	Draining Water-Jet	N/A 3.8×10^1	N/A 4	N/A 3.6×10^{-4}	N/A 1.4×10^{-2}	N/A 5×10^{-4}	N/A 6.9×10^{-8}
6. Remove Condenser	Oxyacetylene Torch	4.0×10^3	4	1.2×10^{-3}	4.6×10^0	5×10^{-5}	2.3×10^{-4}
7. Drain Condenser to Radwaste; Water-Jet Clean Condensate Storage Tanks	Draining Water-Jet	N/A 9.4×10^2	N/A 4	N/A 3.6×10^{-4}	N/A 3.4×10^{-1}	N/A 5×10^{-5}	N/A 1.7×10^{-5}
8. Chemical Decon. Drain Systems • Liquid Lead • Spray Leak	Single-Pass Chem. Decon.	7.3×10^1 7.3×10^1	4 4	2.7×10^{-7} 4.1×10^{-6}	2.0×10^{-5} 3.0×10^{-4}	5×10^{-4} 5×10^{-4}	1.0×10^{-8} 1.5×10^{-7}
9. Remove Piping (O.D.)	Oxyacetylene and Plasma Torches, Power Hacksaws						
60 mm		4.2×10^2	4	2.6×10^{-4}	1.1×10^{-1}	5×10^{-4}	5.5×10^{-5}
152 mm		1.1×10^2	4	2.4×10^{-4}	2.6×10^{-1}	5×10^{-4}	1.3×10^{-4}
356 mm		9.0×10^1	4	3.9×10^{-4}	3.5×10^{-1}	5×10^{-4}	1.8×10^{-4}
533 mm		1.4×10^2	4	1.8×10^{-4}	2.6×10^{-1}	5×10^{-4}	5.5×10^{-5}
660 mm		4.1×10^2	4	1.5×10^{-4}	6.0×10^{-2}	5×10^{-4}	3.0×10^{-5}
1 220 mm		6.0×10^2	4	2.3×10^{-4}	1.4×10^{-1}	5×10^{-4}	6.9×10^{-5}
Total							5.2×10^{-4}
10. Remove Equipment • Misc. Equipment • LP Feedwater Heaters • HP Feedwater Heaters • Moisture Separator Reheater • Condensate Storage Tanks	Oxyacetylene Torch	2.7×10^1 2.9×10^2 3.4×10^1 1.7×10^2 1.2×10^2	4 4 4 4 4	3.7×10^{-3} 7.4×10^{-4} 2.4×10^{-3} 7.3×10^{-4} 6.7×10^{-3}	1.0×10^{-1} 2.1×10^{-1} 8.0×10^{-3} 1.2×10^{-1} 8.0×10^{-2}	5×10^{-4} 5×10^{-4} 5×10^{-4} 5×10^{-4} 5×10^{-4}	5.0×10^{-5} 1.1×10^{-5} 4.0×10^{-6} 6.2×10^{-5} 4.0×10^{-4}
Total							5.3×10^{-4}
11. Remove Contaminated Concrete	Drilling Spalling	1.3×10^3 7.4×10^2	5 5	5.9×10^{-7} N/A	7.7×10^{-4} N/A	5×10^{-5} N/A	3.9×10^{-8} N/A
12. Remove HVAC and Electrical Systems	Oxyacetylene Torch	1.5×10^2	5	9.8×10^{-4}	1.5×10^{-1}	5×10^{-4}	7.4×10^{-5}
13. Final Radiation Survey	N/A	N/A	N/A	N/A	N/A	N/A	N/A

(a) These numbers refer to the reference radionuclide inventories discussed in Appendix E.

(b) The assumed building HEPA filter transmission factor is 5×10^{-4} . The combined transmission factor for a contamination control envelope and the building HEPA filter is 5×10^{-5} .

(c) Dose calculations are made based on annual releases for comparison purposes.

(d) N/A = Not Applicable.

TABLE N.2-9. Calculated Atmospheric Releases for Immediate Dismantlement of the Radwaste and Control Building

Task	General Method	Operational Time (hr)	Reference Radionuclide Inventory Number(a)	Airborne Generation Rate Inside Building or Control Envelope (Ci/hr)	Total Airborne Radioactivity Inside Building or Control Envelope (Ci)	Contamination Control System Transmission Factor(b)	Total Atmospheric Release (Ci/yr)(c)
1. Comprehensive Radiation Survey	Radiation Monitors	6.0×10^1	5	4.6×10^{-4}	2.8×10^{-2}	5×10^{-4}	1.4×10^{-5}
2. General Cleanup	Sweeping	3.0×10^1	5	1.6×10^{-3}	4.6×10^{-2}	5×10^{-4}	2.3×10^{-5}
3. Chemical Decon. Drain Systems	Single-Pass Chem. Decon.	2.6×10^1	4	3.4×10^{-7}	6.1×10^{-6}	5×10^{-4}	3.0×10^{-9}
• Liquid Leak		2.6×10^1	4	3.5×10^{-6}	9.2×10^{-5}	5×10^{-4}	4.6×10^{-8}
• Spray Leak							
4. Chemical Decon. Equip-ment; Water-Jet Clean	Water-Jet	4.9×10^1	4	3.3×10^{-2}	1.6×10^0	5×10^{-4}	8.1×10^{-4}
• Liquid Leak	Loop Chem.	4.0×10^1	4	1.4×10^{-5}	5.7×10^{-4}	5×10^{-4}	2.8×10^{-7}
• Spray Leak	Decon.	4.0×10^1	4	2.1×10^{-4}	8.6×10^{-3}	5×10^{-4}	4.3×10^{-6}
5. Remove Piping	Oxyacetylene and Plasma Torches, Power Hacksaws (O.D.)						
	60 mm	4.0×10^2	4	8.0×10^{-4}	3.2×10^{-1}	5×10^{-4}	1.6×10^{-4}
	152 mm	1.0×10^3	4	9.7×10^{-4}	9.7×10^{-1}	5×10^{-4}	4.8×10^{-4}
	356 mm	1.6×10^3	4	5.1×10^{-4}	8.2×10^{-2}	5×10^{-4}	4.1×10^{-5}
	533 mm	7.0×10^3	4	8.6×10^{-5}	6.0×10^{-3}	5×10^{-4}	3.0×10^{-6}
	914 mm	3.8×10^1	4	1.4×10^{-3}	5.1×10^{-2}	5×10^{-4}	2.6×10^{-5}
	Total						7.1×10^{-4}
6. Install Temporary Radwaste System	N/A(d)	N/A	N/A	N/A	N/A	N/A	N/A
7. Remove Equipment	Oxyacetylene Torch	3.3×10^2	4	2.7×10^{-2}	9.0×10^0	5×10^{-4}	4.5×10^{-3}
8. Remove Contaminated Concrete	Drilling	6.6×10^2	5	4.3×10^{-6}	2.9×10^{-3}	5×10^{-5}	1.4×10^{-7}
	Spalling	3.8×10^2	5	N/A	N/A	N/A	N/A
9. Remove Misc. Steel Structures	Oxyacetylene Torch and Power Hacksaws	1.3×10^1	5	3.1×10^{-4}	4.0×10^{-3}	5×10^{-4}	2.0×10^{-6}
10. Remove HVAC and Electrical Systems	Oxyacetylene Torch	1.2×10^2	5	1.1×10^{-3}	1.4×10^{-3}	5×10^{-4}	6.8×10^{-5}
	Power Hacksaws	2.2×10^1	5	6.8×10^{-4}	1.5×10^{-2}	5×10^{-4}	7.6×10^{-6}
11. Final Radiation Survey		N/A	N/A	N/A	N/A	N/A	N/A

(a) These numbers refer to the reference radionuclide inventories discussed in Appendix E.

(b) The assumed building HEPA filter transmission factor is 5×10^{-4} . The combined transmission factor for a contamination control envelope and the building HEPA filter is 5×10^{-5} .

(c) Dose calculations are made based on annual releases for comparison purposes.

(d) N/A = Not Applicable.

fifty-year committed dose equivalents to both the maximum-exposed individual and to the population residing within 80 km of the site are calculated for each of the reference BWR buildings analyzed. Each of the atmospheric releases is assumed to be a chronic release; that is, one that occurs at a uniform rate for a period of one year. A uniform chronic release is assumed so that direct comparisons of the impact of immediate dismantlement tasks can be made.

Radiation doses from atmospheric releases of radioactivity are calculated for direct exposure, inhalation, and ingestion radiation exposure pathways. Radiation doses from air submersion are not calculated since they have been shown to be insignificant in previous decommissioning studies.^(12,13,14) For inhalation, radionuclides are assumed to have soluble clearance times (as defined in Appendix F) for the organs of reference, except for the lung where insoluble clearance times are assumed. For this study, the organs of reference for which radiation dose is calculated include: total body, bone, lung, thyroid, and GI-tract (lower large intestine).

Radiation doses to the maximum-exposed individual are shown in Tables N.2-10, N.2-11, and N.2-12 for atmospheric releases from immediate dismantlement tasks in the Reactor Building, Turbine Generator Building, and Rad-waste and Control Building. Population doses resulting from these atmospheric releases are shown in Tables N.2-13, N.2-14, and N.2-15 for these buildings, respectively. The atmospheric dispersion ($\bar{X}/Q'$) values used in the dose calculations for routine atmospheric releases are 7.5×10^{-6} sec/m³ for the maximum-exposed individual and 2.4×10^{-9} sec/m³ for the population. A total population of 3.5×10^6 within 80 km of the site is assumed. More information on the reference site and the radiation dose models used in this study is found in Appendices B and F, respectively.

N.2.2 Postulated Accidents During Immediate Dismantlement

During immediate dismantlement, situations may arise that lead to the accidental atmospheric release of radioactivity. Accident scenarios and calculated airborne radioactivity releases considered for immediate dismantlement in this study are listed in Table N.2-16. The accidents are listed by decreasing atmospheric release. Accident scenarios are based on the technologies used during immediate dismantlement tasks. A general estimate of the frequency of occurrence of the level of the atmospheric release associated with each postulated accident is also given. The frequency of occurrence is arrived at by considering not only the probability of the accident but also the probability of an atmospheric release of the calculated magnitude. The frequency of occurrence is listed as "high" if the occurrence of a release of similar or greater magnitude per year is greater than 10^{-2} , as "medium" if between 10^{-2} and 10^{-5} , and as "low" if less than 10^{-5} .

TABLE N.2-10. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine Immediate Dismantlement Tasks in the Reactor Building and Primary Containment

Task (a)	Reference Radionuclide Inventory Number(b)	Total Atmospheric Release (Ci/yr)(c)	First-Year Dose (rem)					Fifty-Year Committed Dose Equivalent (rem)				
			Total Body	Bone	Lung	Thyroid	GI-LL	Total Body	Bone	Lung	Thyroid	GI-LL
2. Comprehensive Radiation Survey	5	6.8 x 10 ⁻⁶	2.8 x 10 ⁻⁹	5.8 x 10 ⁻⁹	4.9 x 10 ⁻⁹	1.3 x 10 ⁻⁸	1.4 x 10 ⁻⁹	5.2 x 10 ⁻⁹	1.4 x 10 ⁻⁸	1.5 x 10 ⁻⁸	1.4 x 10 ⁻⁸	1.4 x 10 ⁻⁹
4. Remove Dryer and Separator <ul style="list-style-type: none">• Contaminated Dryer• Activated Separator	4 1	2.6 x 10 ⁻¹⁰ 9.0 x 10 ⁻⁸	9.9 x 10 ⁻¹⁴ 6.4 x 10 ⁻¹²	9.1 x 10 ⁻¹⁴ 6.7 x 10 ⁻¹²	3.1 x 10 ⁻¹³ 2.4 x 10 ⁻¹¹	8.1 x 10 ⁻¹⁴ 5.9 x 10 ⁻¹²	8.8 x 10 ⁻¹⁴ 6.5 x 10 ⁻¹²	1.1 x 10 ⁻¹³ 6.7 x 10 ⁻¹²	9.9 x 10 ⁻¹⁴ 1.1 x 10 ⁻¹¹	8.8 x 10 ⁻¹³ 7.4 x 10 ⁻¹¹	8.1 x 10 ⁻¹⁴ 5.9 x 10 ⁻¹²	8.8 x 10 ⁻¹⁴ 6.5 x 10 ⁻¹²
5. Drain Suppression Pool to Radwaste; Water-Jet Clean	4	4.2 x 10 ⁻⁶	1.6 x 10 ⁻⁹	1.5 x 10 ⁻⁹	5.0 x 10 ⁻⁹	1.3 x 10 ⁻⁹	1.4 x 10 ⁻⁹	1.7 x 10 ⁻⁹	1.6 x 10 ⁻⁹	1.4 x 10 ⁻⁸	1.3 x 10 ⁻⁹	1.4 x 10 ⁻⁹
6. Remove Reactor Vessel Internals <ul style="list-style-type: none">• Contaminated• Activated	4 1	1.1 x 10 ⁻⁹ 1.9 x 10 ⁻³	4.2 x 10 ⁻¹³ 1.3 x 10 ⁻⁷	3.9 x 10 ⁻¹³ 1.4 x 10 ⁻⁷	1.3 x 10 ⁻¹² 5.1 x 10 ⁻⁷	3.4 x 10 ⁻¹³ 1.2 x 10 ⁻⁷	3.7 x 10 ⁻¹³ 1.4 x 10 ⁻⁷	4.5 x 10 ⁻¹³ 1.4 x 10 ⁻⁷	4.2 x 10 ⁻¹³ 2.3 x 10 ⁻⁷	3.7 x 10 ⁻¹² 1.6 x 10 ⁻⁶	3.4 x 10 ⁻¹³ 1.2 x 10 ⁻⁷	3.7 x 10 ⁻¹³ 1.4 x 10 ⁻⁷
8. Drain Reactor Well Pool to Radwaste; Water-Jet Clean	4	4.0 x 10 ⁻⁷	1.5 x 10 ⁻¹⁰	1.4 x 10 ⁻¹⁰	4.8 x 10 ⁻¹⁰	1.2 x 10 ⁻¹⁰	1.4 x 10 ⁻¹⁰	1.6 x 10 ⁻¹⁰	1.5 x 10 ⁻¹⁰	1.4 x 10 ⁻⁹	1.2 x 10 ⁻¹⁰	1.4 x 10 ⁻¹⁰
9. Chem. Decon. Reactor Water Recirc. and Cleanup Systems <ul style="list-style-type: none">• Liquid Leak• Spray Leak	4 4	8.3 x 10 ⁻⁸ 1.3 x 10 ⁻⁶	3.2 x 10 ⁻¹¹ 4.9 x 10 ⁻¹⁰	2.9 x 10 ⁻¹¹ 4.6 x 10 ⁻¹⁰	1.0 x 10 ⁻¹⁰ 1.6 x 10 ⁻⁶	2.6 x 10 ⁻¹¹ 4.0 x 10 ⁻¹⁰	2.8 x 10 ⁻¹¹ 4.4 x 10 ⁻¹⁰	3.4 x 10 ⁻¹¹ 5.3 x 10 ⁻¹⁰	3.2 x 10 ⁻¹¹ 4.9 x 10 ⁻¹⁰	2.8 x 10 ⁻¹⁰ 4.4 x 10 ⁻⁹	2.6 x 10 ⁻¹¹ 4.0 x 10 ⁻¹⁰	2.8 x 10 ⁻¹¹ 4.4 x 10 ⁻¹⁰
10. Cleanup, Stage and Shield Hot Spots in Primary Containment	5	1.3 x 10 ⁻⁵	5.3 x 10 ⁻⁹	1.1 x 10 ⁻⁸	9.4 x 10 ⁻⁹	2.5 x 10 ⁻⁸	2.6 x 10 ⁻⁹	9.9 x 10 ⁻⁹	2.6 x 10 ⁻⁸	2.9 x 10 ⁻⁸	2.6 x 10 ⁻⁸	2.6 x 10 ⁻⁹
12. Remove Reactor Vessel <ul style="list-style-type: none">• Remove Activated• Remove Contaminated• Segment Activated• Segment Contaminated	2 4 2 4	1.1 x 10 ⁻³ 1.2 x 10 ⁻⁴ 1.9 x 10 ⁻⁸ 1.5 x 10 ⁻¹⁰	2.0 x 10 ⁻⁸ 4.6 x 10 ⁻⁸ 3.4 x 10 ⁻¹³ 5.7 x 10 ⁻¹⁴	2.2 x 10 ⁻⁸ 4.2 x 10 ⁻⁸ 3.8 x 10 ⁻¹³ 5.3 x 10 ⁻¹⁴	6.5 x 10 ⁻⁸ 1.4 x 10 ⁻⁷ 1.1 x 10 ⁻¹² 1.8 x 10 ⁻¹³	1.7 x 10 ⁻⁸ 3.7 x 10 ⁻⁸ 2.9 x 10 ⁻¹² 4.7 x 10 ⁻¹⁴	1.9 x 10 ⁻⁸ 4.1 x 10 ⁻⁸ 3.2 x 10 ⁻¹³ 5.1 x 10 ⁻¹⁴	2.3 x 10 ⁻⁸ 4.9 x 10 ⁻⁸ 4.0 x 10 ⁻¹³ 6.2 x 10 ⁻¹⁴	2.8 x 10 ⁻⁸ 4.6 x 10 ⁻⁸ 4.8 x 10 ⁻¹³ 5.7 x 10 ⁻¹⁴	1.7 x 10 ⁻⁷ 4.1 x 10 ⁻⁷ 2.9 x 10 ⁻¹² 5.1 x 10 ⁻¹³	1.7 x 10 ⁻⁸ 3.7 x 10 ⁻⁷ 2.8 x 10 ⁻¹³ 4.7 x 10 ⁻¹⁴	2.0 x 10 ⁻⁸ 4.1 x 10 ⁻⁸ 3.4 x 10 ⁻¹³ 5.1 x 10 ⁻¹⁴
14. Remove Primary Containment Piping and Equipment(d)	4 5	9.8 x 10 ⁻⁴ 2.2 x 10 ⁻⁵	3.7 x 10 ⁻⁷ 9.0 x 10 ⁻⁹	3.4 x 10 ⁻⁷ 1.9 x 10 ⁻⁸	1.2 x 10 ⁻⁶ 1.6 x 10 ⁻⁸	3.0 x 10 ⁻⁷ 4.2 x 10 ⁻⁸	3.3 x 10 ⁻⁷ 4.4 x 10 ⁻⁸	4.0 x 10 ⁻⁷ 1.7 x 10 ⁻⁸	3.7 x 10 ⁻⁷ 4.4 x 10 ⁻⁸	3.3 x 10 ⁻⁶ 4.8 x 10 ⁻⁸	3.0 x 10 ⁻⁷ 4.4 x 10 ⁻⁸	3.3 x 10 ⁻⁷ 4.4 x 10 ⁻⁹
15. Remove Sacrificial Shield and Radial Beams <ul style="list-style-type: none">• Activated• Contaminated• Radial Beams• Concrete<ul style="list-style-type: none">- Drilling- Blasting	2 5 5 3 3	2.2 x 10 ⁻³ 5.5 x 10 ⁻⁵ 5.8 x 10 ⁻⁵ 1.7 x 10 ⁻¹² 4.7 x 10 ⁻¹⁰	4.0 x 10 ⁻⁸ 2.3 x 10 ⁻⁸ 2.4 x 10 ⁻⁸ 2.4 x 10 ⁻¹⁷ 6.6 x 10 ⁻¹⁵	4.4 x 10 ⁻⁸ 4.7 x 10 ⁻⁸ 5.0 x 10 ⁻⁸ 2.7 x 10 ⁻¹⁷ 7.5 x 10 ⁻¹⁵	1.3 x 10 ⁻⁷ 4.0 x 10 ⁻⁸ 4.2 x 10 ⁻⁸ 1.9 x 10 ⁻¹⁷ 5.2 x 10 ⁻¹⁵	3.3 x 10 ⁻⁸ 1.0 x 10 ⁻⁷ 1.1 x 10 ⁻⁷ 1.1 x 10 ⁻¹⁶ 2.9 x 10 ⁻¹⁴	3.7 x 10 ⁻⁸ 1.1 x 10 ⁻⁸ 1.2 x 10 ⁻⁸ 2.4 x 10 ⁻¹⁷ 6.6 x 10 ⁻¹⁵	4.6 x 10 ⁻⁸ 4.2 x 10 ⁻⁸ 4.4 x 10 ⁻⁸ 2.9 x 10 ⁻¹⁷ 8.0 x 10 ⁻¹⁵	5.5 x 10 ⁻⁸ 1.1 x 10 ⁻⁷ 1.2 x 10 ⁻⁷ 6.5 x 10 ⁻¹⁷ 1.8 x 10 ⁻¹⁴	3.3 x 10 ⁻⁷ 1.2 x 10 ⁻⁷ 1.3 x 10 ⁻⁷ 3.1 x 10 ⁻¹⁶ 8.5 x 10 ⁻¹⁴	3.3 x 10 ⁻⁸ 1.1 x 10 ⁻⁷ 1.2 x 10 ⁻⁷ 1.9 x 10 ⁻¹⁶ 5.2 x 10 ⁻¹⁴	4.0 x 10 ⁻⁸ 1.1 x 10 ⁻⁸ 1.2 x 10 ⁻⁸ 2.4 x 10 ⁻¹⁷ 6.6 x 10 ⁻¹⁵
16. Remove Contaminated Concrete from Primary Containment	5	1.1 x 10 ⁻⁵	4.5 x 10 ⁻⁹	9.5 x 10 ⁻⁹	7.9 x 10 ⁻⁹	2.1 x 10 ⁻⁸	2.2 x 10 ⁻⁹	8.4 x 10 ⁻⁹	2.2 x 10 ⁻⁸	2.4 x 10 ⁻⁸	2.2 x 10 ⁻⁸	2.2 x 10 ⁻⁹
17. Remove HVAC and Electrical Systems from Primary Containment	5	2.2 x 10 ⁻⁴	9.0 x 10 ⁻⁸	1.9 x 10 ⁻⁷	1.6 x 10 ⁻⁷	4.2 x 10 ⁻⁷	4.4 x 10 ⁻⁸	1.7 x 10 ⁻⁷	4.4 x 10 ⁻⁷	4.8 x 10 ⁻⁷	4.4 x 10 ⁻⁷	4.4 x 10 ⁻⁸
19. Chem. Decon. Residual Heat Removal, Low- and High-Pressure Core Spray Systems <ul style="list-style-type: none">• Liquid Leak• Spray Leak	4 4	6.0 x 10 ⁻⁸ 9.0 x 10 ⁻⁷	2.3 x 10 ⁻¹¹ 3.4 x 10 ⁻¹⁰	2.1 x 10 ⁻¹¹ 3.2 x 10 ⁻¹⁰	7.2 x 10 ⁻¹¹ 1.1 x 10 ⁻⁹	1.9 x 10 ⁻¹¹ 2.8 x 10 ⁻¹⁰	2.0 x 10 ⁻¹¹ 3.1 x 10 ⁻¹⁰	2.5 x 10 ⁻¹¹ 3.7 x 10 ⁻¹⁰	2.3 x 10 ⁻¹¹ 3.4 x 10 ⁻¹⁰	2.0 x 10 ⁻¹⁰ 3.1 x 10 ⁻⁹	1.9 x 10 ⁻¹¹ 2.8 x 10 ⁻¹⁰	2.0 x 10 ⁻¹¹ 3.1 x 10 ⁻¹⁰
20. Remove Reactor Building Piping	4	1.8 x 10 ⁻³	6.8 x 10 ⁻⁷	6.3 x 10 ⁻⁷	2.2 x 10 ⁻⁶	5.6 x 10 ⁻⁷	6.1 x 10 ⁻⁷	7.4 x 10 ⁻⁷	6.8 x 10 ⁻⁷	6.1 x 10 ⁻⁶	5.6 x 10 ⁻⁷	6.1 x 10 ⁻⁷
21. Drain Dryer and Separator Pool to Radwaste; Water-Jet Clean	4	1.2 x 10 ⁻⁶	4.6 x 10 ⁻¹⁰	4.2 x 10 ⁻¹⁰	1.4 x 10 ⁻⁹	3.7 x 10 ⁻¹⁰	4.1 x 10 ⁻¹⁰	4.9 x 10 ⁻¹⁰	4.6 x 10 ⁻¹⁰	4.1 x 10 ⁻⁹	3.7 x 10 ⁻¹⁰	4.1 x 10 ⁻¹⁰
22. Chemical Decon. Drain Systems <ul style="list-style-type: none">• Liquid Leaks• Spray Leaks	4 4	1.4 x 10 ⁻¹⁰ 1.1 x 10 ⁻⁷	5.3 x 10 ⁻¹⁴ 4.2 x 10 ⁻¹¹	4.9 x 10 ⁻¹⁴ 3.9 x 10 ⁻¹¹	1.7 x 10 ⁻¹³ 1.3 x 10 ⁻¹⁰	4.3 x 10 ⁻¹⁴ 3.4 x 10 ⁻¹¹	4.8 x 10 ⁻¹⁴ 3.7 x 10 ⁻¹¹	5.7 x 10 ⁻¹⁴ 4.5 x 10 ⁻¹¹	5.3 x 10 ⁻¹⁴ 4.2 x 10 ⁻¹¹	4.8 x 10 ⁻¹³ 3.7 x 10 ⁻¹⁰	4.3 x 10 ⁻¹⁴ 3.4 x 10 ⁻¹¹	4.8 x 10 ⁻¹⁴ 3.7 x 10 ⁻¹¹
23. Drain Spent Fuel Pool to Radwaste; Water-Jet Clean	4	1.3 x 10 ⁻⁶	4.9 x 10 ⁻¹⁰	4.6 x 10 ⁻¹⁰	1.6 x 10 ⁻⁹	4.0 x 10 ⁻¹⁰	4.4 x 10 ⁻¹⁰	5.3 x 10 ⁻¹⁰	4.9 x 10 ⁻¹⁰	4.4 x 10 ⁻⁹	4.0 x 10 ⁻¹⁰	4.4 x 10 ⁻¹⁰
24. Chem. Decon. Fuel Pool and Cooling and Cleanup System <ul style="list-style-type: none">• Liquid Leak• Spray Leak	4 4	3.3 x 10 ⁻⁹ 5.0 x 10 ⁻⁸	1.3 x 10 ⁻¹² 1.9 x 10 ⁻¹¹	1.2 x 10 ⁻¹² 1.8 x 10 ⁻¹¹	4.0 x 10 ⁻¹² 6.0 x 10 ⁻¹¹	1.0 x 10 ⁻¹² 1.6 x 10 ⁻¹¹	1.1 x 10 ⁻¹² 1.7 x 10 ⁻¹¹	1.4 x 10 ⁻¹² 2.1 x 10 ⁻¹¹	1.3 x 10 ⁻¹¹ 1.9 x 10 ⁻¹¹	1.1 x 10 ⁻¹¹ 1.7 x 10 ⁻¹⁰	1.0 x 10 ⁻¹² 1.6 x 10 ⁻¹¹	1.1 x 10 ⁻¹² 1.7 x 10 ⁻¹¹
25. Remove Reactor Building Equipment(d)	4	3.2 x 10 ⁻⁴	1.2 x 10 ⁻⁷	1.1 x 10 ⁻⁷	3.8 x 10 ⁻⁷	9.9 x 10 ⁻⁸	1.1 x 10 ⁻⁷	1.3 x 10 ⁻⁷	1.2 x 10 ⁻⁷	1.1 x 10 ⁻⁶	9.8 x 10 ⁻⁸	1.1 x 10 ⁻⁷
26. Remove Liners from Spent Fuel Pool, Reactor Well, and Dryer and Separator Pool	4	3.5 x 10 ⁻⁴	1.3 x 10 ⁻⁷	1.2 x 10 ⁻⁷	4.2 x 10 ⁻⁷	1.1 x 10 ⁻⁷	1.2 x 10 ⁻⁷	1.4 x 10 ⁻⁷	1.3 x 10 ⁻⁷	1.2 x 10 ⁻⁶	1.1 x 10 ⁻⁷	1.2 x 10 ⁻⁷
27. Remove Reactor Building Contaminated Concrete	5	5.7 x 10 ⁻⁷	2.3 x 10 ⁻¹⁰	4.9 x 10 ⁻¹⁰	4.1 x 10 ⁻¹⁰	1.1 x 10 ⁻⁹	1.1 x 10 ⁻¹⁰	4.3 x 10 ⁻¹⁰	1.1 x 10 ⁻⁹	1.3 x 10 ⁻⁹	1.1 x 10 ⁻⁹	1.1 x 10 ⁻¹⁰
28. Remove HVAC and Electrical Systems from Reactor Building	5	5.5 x 10 ⁻⁴	2.3 x 10 ⁻⁷	4.7 x 10 ⁻⁷	4.0 x 10 ⁻⁷	1.0 x 10 ⁻⁶	1.1 x 10 ⁻⁷	4.2 x 10 ⁻⁷	1.1 x 10 ⁻⁶	1.2 x 10 ⁻⁶	1.1 x 10 ⁻⁶	1.1 x 10 ⁻⁷
Totals			1.9 x 10 ⁻⁶	2.3 x 10 ⁻⁶	5.7 x 10 ⁻⁶	3.0 x 10 ⁻⁶	1.6 x 10 ⁻⁶	2.4 x 10 ⁻⁶	3.5 x 10 ⁻⁶	1.7 x 10 ⁻⁵	3.2 x 10 ⁻⁶	1.6 x 10 ⁻⁶

(a)Only those tasks that result in an atmospheric release are included in this table.
(b)These numbers refer to the reference radionuclide inventories discussed in Appendix E.
(c)Dose calculations are based on annual releases for comparison purposes.
(d)Primary Containment Equipment is included with reactor building equipment for this analysis.

TABLE N.2-11. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine Immediate Dismantlement Tasks in the Turbine Generator Building

Task (a)	Reference Radionuclide Inventory Number (b)	Total Atmospheric Release (Ci/yr) (c)	First-Year Dose (rem)					Fifty-Year Committed Dose Equivalent (rem)				
			Total Body	Bone	Lung	Thyroid	GI-LLI	Total Body	Bone	Lung	Thyroid	GI-LLI
2. Comprehensive Radiation Survey	5	1.9×10^{-6}	7.8×10^{-10}	1.6×10^{-9}	1.4×10^{-9}	3.6×10^{-9}	3.8×10^{-10}	1.4×10^{-9}	3.8×10^{-9}	4.2×10^{-9}	3.8×10^{-9}	3.8×10^{-10}
3. Cleanup and Stage	5	1.8×10^{-6}	7.4×10^{-10}	1.5×10^{-9}	1.3×10^{-9}	3.4×10^{-9}	3.6×10^{-10}	1.4×10^{-9}	3.6×10^{-9}	4.0×10^{-9}	3.6×10^{-9}	3.6×10^{-10}
4. Remove Turbine	4	6.4×10^{-7}	2.4×10^{-10}	2.2×10^{-10}	7.7×10^{-10}	2.0×10^{-10}	2.2×10^{-10}	2.6×10^{-10}	2.4×10^{-10}	2.2×10^{-9}	2.0×10^{-10}	2.2×10^{-10}
5. Drain Contaminated Systems to Radwaste; Water-Jet Clean Condensate Storage Tanks	4	6.9×10^{-6}	2.6×10^{-9}	2.4×10^{-9}	8.3×10^{-9}	2.1×10^{-9}	2.3×10^{-9}	2.8×10^{-9}	2.6×10^{-9}	2.3×10^{-8}	2.1×10^{-9}	2.3×10^{-9}
6. Remove Condenser	4	2.3×10^{-4}	8.7×10^{-8}	8.1×10^{-8}	2.8×10^{-7}	7.1×10^{-8}	7.8×10^{-8}	9.4×10^{-8}	8.7×10^{-8}	7.8×10^{-7}	7.1×10^{-8}	7.8×10^{-8}
7. Drain Condenser to Radwaste; Water-Jet Clean	4	1.7×10^{-5}	6.5×10^{-9}	6.0×10^{-9}	2.0×10^{-8}	5.3×10^{-9}	5.8×10^{-9}	7.0×10^{-9}	6.5×10^{-9}	5.8×10^{-8}	5.3×10^{-9}	5.8×10^{-9}
8. Chemical Decon. Drain Systems	4	1.0×10^{-8}	3.8×10^{-12}	3.5×10^{-12}	1.2×10^{-11}	3.1×10^{-12}	3.4×10^{-12}	4.1×10^{-12}	3.8×10^{-12}	3.4×10^{-11}	3.1×10^{-12}	3.4×10^{-12}
• Liquid Leak	4	1.5×10^{-7}	5.7×10^{-11}	5.3×10^{-11}	1.8×10^{-10}	4.7×10^{-11}	5.1×10^{-11}	6.2×10^{-11}	5.7×10^{-11}	5.1×10^{-10}	4.7×10^{-11}	5.1×10^{-11}
• Spray Leak	4	1.5×10^{-7}	5.7×10^{-11}	5.3×10^{-11}	1.8×10^{-10}	4.7×10^{-11}	5.1×10^{-11}	6.2×10^{-11}	5.7×10^{-11}	5.1×10^{-10}	4.7×10^{-11}	5.1×10^{-11}
9. Remove Piping	4	5.2×10^{-4}	2.0×10^{-7}	1.8×10^{-7}	6.2×10^{-7}	1.6×10^{-7}	1.8×10^{-7}	2.1×10^{-7}	2.0×10^{-7}	1.8×10^{-6}	1.6×10^{-7}	1.8×10^{-7}
10. Remove Equipment	4	5.3×10^{-4}	2.0×10^{-7}	1.9×10^{-7}	6.4×10^{-7}	1.6×10^{-7}	1.8×10^{-7}	2.2×10^{-7}	2.0×10^{-7}	1.8×10^{-6}	1.6×10^{-7}	1.8×10^{-7}
11. Remove Contaminated Concrete	5	3.9×10^{-8}	1.6×10^{-11}	3.4×10^{-11}	2.8×10^{-11}	7.4×10^{-11}	7.8×10^{-12}	3.0×10^{-11}	7.8×10^{-11}	8.6×10^{-11}	7.8×10^{-11}	7.8×10^{-12}
12. Remove HVAC and Electrical Systems	5	7.4×10^{-5}	3.0×10^{-8}	6.4×10^{-8}	5.3×10^{-8}	1.4×10^{-7}	1.5×10^{-8}	5.6×10^{-8}	1.5×10^{-7}	1.6×10^{-7}	1.5×10^{-7}	1.5×10^{-8}
Totals			5.3×10^{-7}	5.3×10^{-7}	1.6×10^{-6}	5.5×10^{-7}	4.6×10^{-7}	5.9×10^{-7}	6.5×10^{-7}	4.6×10^{-6}	5.6×10^{-7}	4.6×10^{-7}

(a) Only those tasks that result in an atmospheric release are included in this table.
 (b) These numbers refer to the reference radionuclide inventories discussed in Appendix E.
 (c) Dose calculations are based on annual releases for comparison purposes.

TABLE N.2-12. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine Immediate Dismantlement Tasks in the Radwaste and Control Building

Task (a)	Reference Radionuclide Inventory Number (b)	Total Atmospheric Release (Ci/yr) (c)	First-Year Dose (rem)					Fifty-Year Committed Dose Equivalent (rem)				
			Total Body	Bone	Lung	Thyroid	GI-LLI	Total Body	Bone	Lung	Thyroid	GI-LLI
1 Comprehensive Radiation Survey	5	1.4×10^{-5}	5.7×10^{-9}	1.2×10^{-8}	1.0×10^{-8}	2.7×10^{-8}	2.8×10^{-9}	1.1×10^{-8}	2.8×10^{-8}	3.1×10^{-8}	2.8×10^{-8}	2.8×10^{-8}
2 General Cleanup	5	2.3×10^{-5}	9.4×10^{-9}	2.0×10^{-8}	1.7×10^{-8}	4.4×10^{-8}	4.6×10^{-9}	1.7×10^{-8}	4.6×10^{-8}	5.1×10^{-8}	4.6×10^{-8}	4.6×10^{-9}
3 Chemical Decon Drain Systems	4	3.0×10^{-9}	1.1×10^{-12}	1.1×10^{-12}	3.6×10^{-12}	9.3×10^{-13}	1.0×10^{-12}	1.2×10^{-12}	1.1×10^{-12}	1.0×10^{-11}	9.3×10^{-13}	1.0×10^{-12}
• Liquid Leak	4	4.6×10^{-8}	1.7×10^{-11}	1.6×10^{-11}	5.5×10^{-11}	1.4×10^{-11}	1.6×10^{-11}	1.9×10^{-11}	1.7×10^{-11}	1.6×10^{-10}	1.4×10^{-11}	1.6×10^{-11}
• Spray Leak	4											
4 Chemical Decon Equipment, Water-Jet Clean	4	8.1×10^{-4}	3.1×10^{-7}	2.8×10^{-7}	9.7×10^{-7}	2.5×10^{-7}	2.8×10^{-7}	3.3×10^{-7}	3.1×10^{-7}	2.8×10^{-6}	2.5×10^{-7}	2.8×10^{-7}
• Water-Jet	4	2.8×10^{-7}	1.1×10^{-10}	9.8×10^{-11}	3.4×10^{-10}	8.7×10^{-11}	9.5×10^{-11}	1.1×10^{-10}	1.1×10^{-10}	9.5×10^{-10}	8.7×10^{-11}	9.5×10^{-11}
• Liquid Leak	4	4.3×10^{-6}	1.6×10^{-9}	1.6×10^{-9}	5.2×10^{-9}	1.3×10^{-9}	1.5×10^{-9}	1.8×10^{-9}	1.6×10^{-9}	1.5×10^{-8}	1.3×10^{-9}	1.5×10^{-9}
• Spray Leak	4											
5 Remove Piping	4	7.1×10^{-4}	2.7×10^{-7}	2.5×10^{-7}	8.5×10^{-7}	2.2×10^{-7}	2.4×10^{-7}	2.9×10^{-7}	2.7×10^{-7}	2.4×10^{-6}	2.2×10^{-7}	2.4×10^{-7}
7 Remove Equipment	4	4.5×10^{-3}	1.7×10^{-6}	1.6×10^{-6}	5.4×10^{-6}	1.4×10^{-6}	1.5×10^{-6}	1.8×10^{-6}	1.7×10^{-6}	1.5×10^{-5}	1.4×10^{-6}	1.5×10^{-6}
8 Remove Contaminated Concrete	5	1.4×10^{-7}	5.7×10^{-11}	1.2×10^{-10}	1.0×10^{-10}	2.7×10^{-10}	2.8×10^{-11}	1.1×10^{-10}	2.8×10^{-10}	3.1×10^{-10}	2.8×10^{-10}	2.8×10^{-11}
9 Remove Misc Steel Structures	5	2.0×10^{-6}	8.2×10^{-10}	1.7×10^{-9}	1.4×10^{-9}	3.8×10^{-9}	4.0×10^{-10}	1.5×10^{-9}	4.0×10^{-9}	4.4×10^{-9}	4.0×10^{-9}	4.0×10^{-10}
10 Remove HVAC and Electrical Systems	5	7.6×10^{-5}	3.1×10^{-8}	6.5×10^{-8}	5.5×10^{-8}	1.4×10^{-7}	1.5×10^{-8}	5.8×10^{-8}	1.5×10^{-7}	1.7×10^{-7}	1.5×10^{-7}	1.5×10^{-8}
Totals			2.3×10^{-6}	2.2×10^{-6}	7.3×10^{-6}	2.1×10^{-6}	2.0×10^{-6}	2.5×10^{-6}	2.5×10^{-6}	2.0×10^{-5}	2.1×10^{-6}	2.0×10^{-6}

(a) Only those tasks that result in an atmospheric release are included in this table

(b) These numbers refer to the reference radionuclide inventories discussed in Appendix E

(c) Dose calculations are based on annual releases for comparison purposes

TABLE N.2-13. Radiation Doses to the Population from Atmospheric Releases During Routine Immediate Dismantlement Tasks in the Reactor Building and Primary Containment(a)

Task (b)	Reference Radionuclide Inventory Number(c)	*Total Atmospheric Release (Ci/yr)(d)	First-Year Dose (man-rem)					Fifty-Year Committed Dose Equivalent (man-rem)				
			Total Body	Bone	Lung	Thyroid	GI-LLI	Total Body	Bone	Lung	Thyroid	GI-LLI
2. Comprehensive Radiation Survey	5	6.8 x 10 ⁻⁶	2 x 10 ⁻⁶	4 x 10 ⁻⁶	5 x 10 ⁻⁶	1 x 10 ⁻⁵	8 x 10 ⁻⁷	4 x 10 ⁻⁶	1 x 10 ⁻⁵	2 x 10 ⁻⁵	1 x 10 ⁻⁵	1 x 10 ⁻⁶
4. Remove Dryer and Separator	4	2.6 x 10 ⁻¹⁰	6 x 10 ⁻¹¹	5 x 10 ⁻¹¹	3 x 10 ⁻¹⁰	5 x 10 ⁻¹¹	5 x 10 ⁻¹¹	7 x 10 ⁻¹¹	6 x 10 ⁻¹¹	9 x 10 ⁻¹⁰	5 x 10 ⁻¹¹	5 x 10 ⁻¹¹
• Contaminated Dryer	1	9.0 x 10 ⁻⁸	4 x 10 ⁻⁹	3 x 10 ⁻⁹	2 x 10 ⁻⁸	3 x 10 ⁻⁹	4 x 10 ⁻⁹	4 x 10 ⁻⁹	7 x 10 ⁻⁹	8 x 10 ⁻⁹	3 x 10 ⁻⁹	4 x 10 ⁻⁹
• Activated Separator												
5. Drain Suppression Pool to Radwaste; Water-Jet Clean	4	4.2 x 10 ⁻⁶	1 x 10 ⁻⁶	8 x 10 ⁻⁷	5 x 10 ⁻⁶	8 x 10 ⁻⁷	9 x 10 ⁻⁷	1 x 10 ⁻⁶	1 x 10 ⁻⁶	2 x 10 ⁻⁵	8 x 10 ⁻⁷	9 x 10 ⁻⁷
6. Remove Reactor Vessel Internals	4	1.1 x 10 ⁻⁹	3 x 10 ⁻¹⁰	2 x 10 ⁻¹⁰	1 x 10 ⁻⁹	2 x 10 ⁻¹⁰	2 x 10 ⁻¹⁰	3 x 10 ⁻¹⁰	3 x 10 ⁻¹⁰	4 x 10 ⁻⁹	2 x 10 ⁻¹⁰	2 x 10 ⁻¹⁰
• Contaminated	1	1.9 x 10 ⁻³	8 x 10 ⁻⁵	7 x 10 ⁻⁵	5 x 10 ⁻⁴	7 x 10 ⁻⁵	8 x 10 ⁻⁵	8 x 10 ⁻⁵	1 x 10 ⁻⁴	2 x 10 ⁻³	7 x 10 ⁻⁵	8 x 10 ⁻⁵
• Activated												
8. Drain Reactor Well Pool to Radwaste; Water-Jet Clean	4	4.0 x 10 ⁻⁷	9 x 10 ⁻⁸	8 x 10 ⁻⁸	5 x 10 ⁻⁷	7 x 10 ⁻⁸	8 x 10 ⁻⁸	1 x 10 ⁻⁷	9 x 10 ⁻⁸	1 x 10 ⁻⁶	7 x 10 ⁻⁸	8 x 10 ⁻⁸
9. Chem. Decon. Reactor Water Recirc. and Cleanup Systems	4	8.3 x 10 ⁻⁸	2 x 10 ⁻⁸	2 x 10 ⁻⁸	1 x 10 ⁻⁷	1 x 10 ⁻⁸	2 x 10 ⁻⁸	2 x 10 ⁻⁸	2 x 10 ⁻⁸	3 x 10 ⁻⁷	1 x 10 ⁻⁸	2 x 10 ⁻⁸
• Liquid Leak	4	1.3 x 10 ⁻⁶	3 x 10 ⁻⁷	3 x 10 ⁻⁷	2 x 10 ⁻⁶	2 x 10 ⁻⁷	3 x 10 ⁻⁷	3 x 10 ⁻⁷	3 x 10 ⁻⁷	5 x 10 ⁻⁶	2 x 10 ⁻⁷	3 x 10 ⁻⁷
• Spray Leak												
10. Clean Up, Stage and Shield Hot Spots in Primary Containment	5	1.3 x 10 ⁻⁵	3 x 10 ⁻⁶	7 x 10 ⁻⁶	9 x 10 ⁻⁶	2 x 10 ⁻⁵	2 x 10 ⁻⁶	7 x 10 ⁻⁶	2 x 10 ⁻⁵	3 x 10 ⁻⁵	2 x 10 ⁻⁵	2 x 10 ⁻⁶
12. Remove Reactor Vessel	2	1.1 x 10 ⁻³	1 x 10 ⁻⁵	1 x 10 ⁻⁵	6 x 10 ⁻⁵	9 x 10 ⁻⁶	1 x 10 ⁻⁵	1 x 10 ⁻⁵	3 x 10 ⁻⁵	2 x 10 ⁻⁴	9 x 10 ⁻⁶	1 x 10 ⁻⁵
• Remove Activated	4	1.2 x 10 ⁻⁴	3 x 10 ⁻⁵	2 x 10 ⁻⁵	1 x 10 ⁻⁴	2 x 10 ⁻⁵	3 x 10 ⁻⁵	3 x 10 ⁻⁵	3 x 10 ⁻⁵	4 x 10 ⁻⁴	2 x 10 ⁻⁵	7 x 10 ⁻⁵
• Remove Contaminated	2	1.9 x 10 ⁻⁸	2 x 10 ⁻¹⁰	2 x 10 ⁻¹⁰	1 x 10 ⁻⁹	2 x 10 ⁻¹⁰	2 x 10 ⁻¹⁰	2 x 10 ⁻¹⁰	5 x 10 ⁻¹⁰	3 x 10 ⁻⁹	2 x 10 ⁻¹⁰	2 x 10 ⁻¹⁰
• Segment Activated	4	1.5 x 10 ⁻¹⁰	3 x 10 ⁻¹¹	3 x 10 ⁻¹¹	2 x 10 ⁻¹⁰	3 x 10 ⁻¹¹	3 x 10 ⁻¹¹	4 x 10 ⁻¹¹	3 x 10 ⁻¹¹	5 x 10 ⁻¹⁰	3 x 10 ⁻¹¹	3 x 10 ⁻¹¹
• Segment Contaminated												
14. Remove Primary Containment Piping and Equipment(d)	4	9.8 x 10 ⁻⁴	2 x 10 ⁻⁴	2 x 10 ⁻⁴	1 x 10 ⁻³	2 x 10 ⁻⁴	2 x 10 ⁻⁴	2 x 10 ⁻⁴	2 x 10 ⁻⁴	4 x 10 ⁻³	2 x 10 ⁻⁴	2 x 10 ⁻⁴
	5	2.2 x 10 ⁻⁵	6 x 10 ⁻⁶	1 x 10 ⁻⁵	2 x 10 ⁻⁵	3 x 10 ⁻⁵	3 x 10 ⁻⁶	1 x 10 ⁻⁵	3 x 10 ⁻⁵	5 x 10 ⁻⁵	4 x 10 ⁻⁵	3 x 10 ⁻⁶
15. Remove Sacrificial Shield and Radial Beams	2	2.2 x 10 ⁻³	2 x 10 ⁻⁵	3 x 10 ⁻⁵	1 x 10 ⁻⁴	2 x 10 ⁻⁵	2 x 10 ⁻⁵	3 x 10 ⁻⁵	6 x 10 ⁻⁵	4 x 10 ⁻⁴	2 x 10 ⁻⁵	2 x 10 ⁻⁵
• Activated	5	5.5 x 10 ⁻⁵	1 x 10 ⁻⁵	3 x 10 ⁻⁵	4 x 10 ⁻⁵	8 x 10 ⁻⁵	7 x 10 ⁻⁶	3 x 10 ⁻⁵	8 x 10 ⁻⁵	1 x 10 ⁻⁴	9 x 10 ⁻⁵	8 x 10 ⁻⁶
• Contaminated	5	5.8 x 10 ⁻⁵	2 x 10 ⁻⁵	3 x 10 ⁻⁵	4 x 10 ⁻⁵	9 x 10 ⁻⁵	7 x 10 ⁻⁶	3 x 10 ⁻⁵	9 x 10 ⁻⁵	1 x 10 ⁻⁴	9 x 10 ⁻⁵	8 x 10 ⁻⁶
• Radial Beams												
• Concrete:												
- Drilling	3	1.7 x 10 ⁻¹²	1 x 10 ⁻¹⁴	2 x 10 ⁻¹⁴	1 x 10 ⁻¹³	1 x 10 ⁻¹⁴	2 x 10 ⁻¹⁴	2 x 10 ⁻¹⁴	4 x 10 ⁻¹⁴	3 x 10 ⁻¹³	1 x 10 ⁻¹⁴	2 x 10 ⁻¹⁴
- Blasting	3	4.7 x 10 ⁻¹⁰	4 x 10 ⁻¹²	5 x 10 ⁻¹²	3 x 10 ⁻¹¹	3 x 10 ⁻¹²	4 x 10 ⁻¹²	5 x 10 ⁻¹²	1 x 10 ⁻¹¹	9 x 10 ⁻¹¹	6 x 10 ⁻¹²	4 x 10 ⁻¹²
16. Remove Contaminated Concrete from Primary Containment	5	1.1 x 10 ⁻⁵	3 x 10 ⁻⁶	6 x 10 ⁻⁶	8 x 10 ⁻⁶	2 x 10 ⁻⁵	1 x 10 ⁻⁶	6 x 10 ⁻⁶	2 x 10 ⁻⁵	3 x 10 ⁻⁵	2 x 10 ⁻⁵	2 x 10 ⁻⁶
17. Remove HVAC and Electrical Systems from Primary Containment	5	2.2 x 10 ⁻⁴	6 x 10 ⁻⁵	1 x 10 ⁻⁴	2 x 10 ⁻⁴	3 x 10 ⁻⁴	3 x 10 ⁻⁵	1 x 10 ⁻⁴	3 x 10 ⁻⁴	5 x 10 ⁻⁴	4 x 10 ⁻⁴	3 x 10 ⁻⁵
19. Chem. Decon. Residual Heat Removal, Low- and High-Pressure Core Sprays	4	6.0 x 10 ⁻⁸	1 x 10 ⁻⁸	1 x 10 ⁻⁸	7 x 10 ⁻⁸	1 x 10 ⁻⁸	1 x 10 ⁻⁸	2 x 10 ⁻⁸	1 x 10 ⁻⁸	2 x 10 ⁻⁷	1 x 10 ⁻⁸	1 x 10 ⁻⁸
• Liquid Leak	4	9.0 x 10 ⁻⁷	2 x 10 ⁻⁷	2 x 10 ⁻⁷	1 x 10 ⁻⁶	2 x 10 ⁻⁷	2 x 10 ⁻⁷	2 x 10 ⁻⁷	2 x 10 ⁻⁷	3 x 10 ⁻⁶	2 x 10 ⁻⁷	2 x 10 ⁻⁷
• Spray Leak												
20. Remove Reactor Building Piping	4	1.8 x 10 ⁻³	4 x 10 ⁻⁴	4 x 10 ⁻⁴	2 x 10 ⁻³	3 x 10 ⁻⁴	4 x 10 ⁻⁴	5 x 10 ⁻⁴	4 x 10 ⁻⁴	6 x 10 ⁻³	3 x 10 ⁻⁴	4 x 10 ⁻⁴
21. Drain Dryer and Separator Pool to Radwaste; Water-Jet Clean	4	1.2 x 10 ⁻⁶	3 x 10 ⁻⁷	2 x 10 ⁻⁷	1 x 10 ⁻⁶	2 x 10 ⁻⁶	3 x 10 ⁻⁷	3 x 10 ⁻⁷	3 x 10 ⁻⁷	4 x 10 ⁻⁶	2 x 10 ⁻⁷	3 x 10 ⁻⁷
22. Chemical Decon. Drain Systems	4	1.4 x 10 ⁻¹⁰	3 x 10 ⁻¹¹	3 x 10 ⁻¹¹	2 x 10 ⁻¹⁰	3 x 10 ⁻¹¹	3 x 10 ⁻¹¹	4 x 10 ⁻¹¹	3 x 10 ⁻¹¹	5 x 10 ⁻¹⁰	3 x 10 ⁻¹¹	3 x 10 ⁻¹¹
• Liquid Leaks	4	1.1 x 10 ⁻⁷	3 x 10 ⁻⁸	2 x 10 ⁻⁸	1 x 10 ⁻⁷	2 x 10 ⁻⁸	2 x 10 ⁻⁸	3 x 10 ⁻⁸	3 x 10 ⁻⁸	4 x 10 ⁻⁷	2 x 10 ⁻⁸	2 x 10 ⁻⁸
• Spray Leaks												
23. Drain Spent Fuel Pool to Radwaste; Water-Jet Clean	4	1.3 x 10 ⁻⁶	3 x 10 ⁻⁷	3 x 10 ⁻⁷	2 x 10 ⁻⁶	2 x 10 ⁻⁷	3 x 10 ⁻⁷	3 x 10 ⁻⁷	3 x 10 ⁻⁷	5 x 10 ⁻⁶	2 x 10 ⁻⁷	3 x 10 ⁻⁷
24. Chem. Decon. Fuel Pool Cooling and Cleanup System	4	3.3 x 10 ⁻⁹	8 x 10 ⁻¹⁰	7 x 10 ⁻¹⁰	4 x 10 ⁻⁹	6 x 10 ⁻¹⁰	7 x 10 ⁻¹⁰	8 x 10 ⁻¹⁰	8 x 10 ⁻¹⁰	1 x 10 ⁻⁸	6 x 10 ⁻¹⁰	7 x 10 ⁻¹⁰
• Liquid Leak	4	5.0 x 10 ⁻⁸	1 x 10 ⁻⁸	1 x 10 ⁻⁸	6 x 10 ⁻⁸	9 x 10 ⁻⁹	1 x 10 ⁻⁸	1 x 10 ⁻⁸	1 x 10 ⁻⁸	2 x 10 ⁻⁷	9 x 10 ⁻⁹	1 x 10 ⁻⁸
• Spray Leak												
25. Remove Reactor Building Equipment(d)	4	3.2 x 10 ⁻⁴	7 x 10 ⁻⁵	6 x 10 ⁻⁵	4 x 10 ⁻⁴	5 x 10 ⁻⁴	7 x 10 ⁻⁵	8 x 10 ⁻⁵	7 x 10 ⁻⁵	1 x 10 ⁻³	6 x 10 ⁻⁵	7 x 10 ⁻⁵
26. Remove Liners from Spent Fuel Pool, Reactor Well, and Dryer and Separator Pool	4	3.5 x 10 ⁻⁴	8 x 10 ⁻⁵	7 x 10 ⁻⁵	4 x 10 ⁻⁴	6 x 10 ⁻⁵	7 x 10 ⁻⁵	9 x 10 ⁻⁵	8 x 10 ⁻⁵	1 x 10 ⁻³	6 x 10 ⁻⁵	7 x 10 ⁻⁵
27. Remove Reactor Building Contaminated Concrete	5	5.7 x 10 ⁻⁶	1 x 10 ⁻⁷	3 x 10 ⁻⁷	4 x 10 ⁻⁷	9 x 10 ⁻⁷	7 x 10 ⁻⁸	3 x 10 ⁻⁷	9 x 10 ⁻⁷	1 x 10 ⁻⁶	9 x 10 ⁻⁷	8 x 10 ⁻⁸
28. Remove HVAC and Electrical Systems from Reactor Building	5	5.5 x 10 ⁻⁴	1 x 10 ⁻⁴	3 x 10 ⁻⁴	4 x 10 ⁻⁴	8 x 10 ⁻⁴	7 x 10 ⁻⁵	3 x 10 ⁻⁴	8 x 10 ⁻⁴	1 x 10 ⁻³	9 x 10 ⁻⁴	8 x 10 ⁻⁵
Totals			1 x 10 ⁻³	1 x 10 ⁻³	5 x 10 ⁻³	2 x 10 ⁻³	1 x 10 ⁻³	1 x 10 ⁻³	2 x 10 ⁻³	2 x 10 ⁻²	2 x 10 ⁻³	1 x 10 ⁻³

(a) The calculated doses are shown to one significant figure and are to the 3.5 x 10⁶ people residing within a radius of 80 km from the reference site.
(b) Only those tasks that result in an atmospheric release are included in this table.
(c) These numbers refer to the Reference Radionuclide Inventories discussed in Appendix E.
(d) Dose calculations are based on annual releases for comparison purposes.
(e) Primary containment equipment is included with the Reactor Building equipment for this analysis.

TABLE N.2-14. Radiation Doses to the Population from Atmospheric Releases During Routine Immediate Dismantlement Tasks in the Turbine Generator Building^(a)

Task ^(b)	Reference Radionuclide Inventory Number ^(c)	Total Atmospheric Release (Ci/yr) ^(d)	First-Year Dose (man-rem)					Fifty-Year Committed Dose Equivalent (man-rem)				
			Total Body	Bone	Lung	Thyroid	GI-LLI	Total Body	Bone	Lung	Thyroid	GI-LLI
2. Comprehensive Radiation Survey	5	1.9×10^{-6}	5×10^{-7}	1×10^{-6}	1×10^{-6}	3×10^{-6}	2×10^{-7}	1×10^{-6}	3×10^{-6}	4×10^{-6}	3×10^{-6}	3×10^{-7}
3. Clean Up and Stage	5	1.8×10^{-6}	5×10^{-7}	1×10^{-6}	1×10^{-6}	3×10^{-6}	2×10^{-7}	1×10^{-6}	3×10^{-6}	4×10^{-6}	3×10^{-6}	3×10^{-7}
4. Remove Turbine	4	6.4×10^{-7}	1×10^{-7}	1×10^{-7}	8×10^{-7}	1×10^{-7}	1×10^{-7}	2×10^{-7}	1×10^{-7}	2×10^{-6}	1×10^{-7}	1×10^{-7}
5. Drain Contaminated Systems to Radwaste; Water-Jet Clean Condensate Storage Tanks	4	6.9×10^{-6}	2×10^{-6}	1×10^{-6}	8×10^{-6}	1×10^{-6}	1×10^{-6}	2×10^{-6}	2×10^{-6}	2×10^{-5}	1×10^{-6}	1×10^{-6}
6. Remove Condenser	4	2.3×10^{-4}	5×10^{-5}	5×10^{-5}	3×10^{-4}	4×10^{-5}	5×10^{-5}	6×10^{-5}	5×10^{-5}	8×10^{-4}	4×10^{-5}	5×10^{-5}
7. Drain Condenser to Radwaste; Water-Jet Clean	4	1.7×10^{-5}	4×10^{-6}	3×10^{-6}	2×10^{-5}	3×10^{-6}	4×10^{-6}	4×10^{-6}	4×10^{-6}	6×10^{-5}	3×10^{-6}	4×10^{-6}
8. Chemical Decon. Drain Systems												
• Liquid Leak	4	1.0×10^{-8}	2×10^{-9}	2×10^{-9}	1×10^{-8}	2×10^{-9}	2×10^{-9}	3×10^{-9}	2×10^{-9}	4×10^{-8}	2×10^{-9}	2×10^{-9}
• Spray Leak	4	1.5×10^{-7}	3×10^{-8}	3×10^{-8}	2×10^{-7}	3×10^{-8}	3×10^{-8}	4×10^{-8}	3×10^{-8}	5×10^{-7}	3×10^{-8}	3×10^{-8}
9. Remove Piping	4	5.2×10^{-4}	1×10^{-4}	1×10^{-4}	6×10^{-4}	9×10^{-5}	1×10^{-4}	1×10^{-4}	1×10^{-4}	2×10^{-3}	1×10^{-4}	1×10^{-4}
10. Remove Equipment	4	5.3×10^{-4}	1×10^{-4}	1×10^{-4}	6×10^{-4}	1×10^{-4}	1×10^{-4}	1×10^{-4}	1×10^{-4}	2×10^{-3}	1×10^{-4}	1×10^{-4}
11. Remove Contaminated Concrete	5	3.9×10^{-8}	1×10^{-8}	2×10^{-8}	3×10^{-8}	6×10^{-8}	5×10^{-9}	2×10^{-8}	6×10^{-8}	9×10^{-8}	6×10^{-8}	5×10^{-9}
12. Remove HVAC and Electrical Systems	5	7.4×10^{-5}	2×10^{-5}	4×10^{-5}	5×10^{-5}	1×10^{-4}	9×10^{-6}	4×10^{-5}	1×10^{-4}	2×10^{-4}	1×10^{-4}	1×10^{-5}
Totals			3×10^{-4}	3×10^{-4}	2×10^{-3}	3×10^{-4}	3×10^{-4}	3×10^{-4}	4×10^{-4}	5×10^{-3}	4×10^{-4}	3×10^{-4}

(a) The calculated doses are shown to one significant figure and are to the 3.5×10^6 people residing within a radius of 80 km from the reference site.

(b) Only those tasks that result in an atmospheric release are included in this table.

(c) These numbers refer to the reference radionuclide inventories discussed in Appendix E.

(d) Dose calculations are based on annual releases for comparison purposes.

TABLE N.2-15. Radiation Doses to the Population from Atmospheric Releases During Routine Immediate Dismantlement Tasks in the Radwaste and Control Building^(a)

Task ^(b)	Reference Radionuclide Inventory Number ^(c)	Total Atmospheric Release (Ci/yr) ^(d)	First-Year Dose (man-rem)					Fifty-Year Committed Dose Equivalent (man-rem)				
			Total Body	Bone	Lung	Thyroid	GI-LLI	Total Body	Bone	Lung	Thyroid	GI-LLI
1. Comprehensive Radiation Survey	5	1.4×10^{-5}	4×10^{-6}	8×10^{-6}	1×10^{-5}	2×10^{-5}	2×10^{-6}	8×10^{-6}	2×10^{-5}	3×10^{-5}	2×10^{-5}	2×10^{-6}
2. General Cleanup	5	2.3×10^{-5}	6×10^{-6}	1×10^{-5}	2×10^{-5}	3×10^{-5}	3×10^{-6}	1×10^{-5}	3×10^{-5}	5×10^{-5}	4×10^{-5}	3×10^{-6}
3. Chemical Decon. Drain Systems	4	3.0×10^{-9}	7×10^{-10}	6×10^{-10}	4×10^{-9}	5×10^{-10}	6×10^{-10}	8×10^{-10}	7×10^{-10}	1×10^{-8}	5×10^{-10}	6×10^{-10}
• Liquid Leak	4	4.6×10^{-8}	1×10^{-8}	9×10^{-9}	6×10^{-8}	8×10^{-9}	1×10^{-8}	1×10^{-8}	1×10^{-8}	2×10^{-7}	8×10^{-9}	1×10^{-8}
• Spray Leak	4											
4. Chemical Decon. Equip-ment; Water-Jet Clean	4	8.1×10^{-4}	2×10^{-4}	2×10^{-4}	1×10^{-3}	1×10^{-4}	2×10^{-4}	2×10^{-4}	2×10^{-4}	3×10^{-3}	1×10^{-4}	2×10^{-4}
• Water-Jet	4	2.8×10^{-7}	6×10^{-8}	6×10^{-8}	3×10^{-7}	5×10^{-8}	6×10^{-8}	7×10^{-8}	6×10^{-8}	1×10^{-6}	5×10^{-8}	6×10^{-8}
• Liquid Leak	4	4.3×10^{-6}	1×10^{-6}	9×10^{-7}	5×10^{-6}	8×10^{-7}	9×10^{-7}	1×10^{-6}	1×10^{-6}	2×10^{-5}	8×10^{-7}	9×10^{-7}
• Spray Leak	4											
5. Remove Piping	4	7.1×10^{-4}	2×10^{-4}	1×10^{-4}	9×10^{-4}	1×10^{-4}	1×10^{-4}	2×10^{-4}	2×10^{-4}	3×10^{-3}	1×10^{-4}	1×10^{-4}
7. Remove Equipment	4	4.5×10^{-3}	1×10^{-3}	9×10^{-4}	5×10^{-3}	8×10^{-4}	9×10^{-4}	1×10^{-3}	1×10^{-3}	2×10^{-2}	8×10^{-4}	9×10^{-4}
8. Remove Contaminated Concrete	5	1.4×10^{-7}	4×10^{-8}	8×10^{-8}	1×10^{-7}	2×10^{-7}	2×10^{-8}	8×10^{-8}	2×10^{-7}	3×10^{-7}	2×10^{-7}	2×10^{-8}
9. Remove Miscellaneous Electrical Systems	5	2.0×10^{-6}	5×10^{-7}	1×10^{-6}	1×10^{-6}	3×10^{-6}	2×10^{-7}	1×10^{-6}	3×10^{-6}	5×10^{-6}	3×10^{-6}	3×10^{-7}
10. Remove HVAC and Electrical Systems	5	7.6×10^{-5}	2×10^{-5}	4×10^{-5}	5×10^{-5}	1×10^{-4}	9×10^{-6}	4×10^{-5}	1×10^{-4}	2×10^{-4}	1×10^{-4}	1×10^{-5}
Totals			1×10^{-3}	1×10^{-3}	7×10^{-3}	1×10^{-3}	1×10^{-3}	1×10^{-3}	2×10^{-3}	3×10^{-2}	1×10^{-3}	1×10^{-3}

(a) The calculated doses are shown to one significant figure and are to the 3.5×10^6 people residing within a radius of 80 km from the reference site.

(b) Only those tasks that result in an atmospheric release are included in this table.

(c) These numbers refer to the reference radionuclide inventories discussed in Appendix E.

(d) Dose calculations are based on annual releases for comparison purposes.

TABLE N.2-16. Postulated Accidental Atmospheric Releases During Immediate Dismantlement of the Reference BWR

Incident	Comments	Frequency of Occurrence(a)	Reference Radionuclide Inventory Number(b)	Total Airborne Radioactivity Inside Building or Control Envelope (Ci)	Contamination Control System Transmission Factor(c)	Total Atmospheric Release (Ci/hr)(d)
Explosion of LPG Leaked from a Front-End Loader	4.5 kg of LPG leaked in the reactor building during the loading of activated concrete	Low	3	8.6×10^{-3}	1.0	8.6×10^{-3}
Vacuum Filter-Bag Rupture	Rupture of a full filter-bag during surface cleaning	Medium	5	1.7×10^0	5.0×10^{-4}	3.5×10^{-4}
Contamination Control Envelope Rupture	Leakage during reactor vessel removal (Task 12 in the Reactor Building)	High	2	2.8×10^{-1}	5.0×10^{-4}	1.4×10^{-4}
Oxyacetylene Explosion	Explosion during reactor vessel removal (Task 12 in the Reactor Building)	Medium	2	2.4×10^{-1}	5.0×10^{-4}	1.2×10^{-4}
Contaminated Sweeping Compound Fire	A fire consumes 0.5 m^3 of sweeping compound	Medium	5	2.2×10^{-3}	5.0×10^{-4}	1.1×10^{-6}
Gross Leak During Loop Chemical Decontamination	A leak 10 times larger than the routine leak in the Radwaste and Control Building	Low	4	2.1×10^{-3}	5.0×10^{-4}	1.0×10^{-6}
Filter Damage from Blasting Surges	Damage from routine concrete blasting operations (Task 15 in the Reactor Building)	Medium	3	2.6×10^{-4}	5.0×10^{-4}	1.3×10^{-7}
Combustible Waste Fire	1 m^3 of waste is involved in a fire	High	5	1.2×10^{-5}	5.0×10^{-4}	6.0×10^{-9}
Detonation of Unused Explosives	Detonation of a positioned charge after removal of controls (Task 15 in the Reactor Building)	Medium	3	9.5×10^{-7}	5.0×10^{-4}	4.8×10^{-10}
Ion Exchange Resin Accidents	Suspended in liquids with remote operations--no calculation	---(e)	---	---	---	---
Loss of Services	Loss of water supply, electrical power or airflow--no calculation	---	---	---	---	---
Natural Phenomena	Floods, earthquakes, tornadoes or high winds--comparison to design-basis--no calculation	---	---	---	---	---
Aircraft Crashes	Comparison to routine operation probability--no calculation	---	---	---	---	---
Man-Caused Events	Detailed analysis is beyond the scope of study--no calculation	---	---	---	---	---

(a)The frequency of occurrence considers not only the probability of the accident, but also the probability of an atmospheric release of the calculated magnitude. The frequency of occurrence is listed as "High" if the occurrence of a release of similar or greater magnitude per year is $>10^{-2}$, as "Medium" if between 10^{-2} and 10^{-5} , and as "Low" if $<10^{-5}$.

(b)These numbers refer to the reference radionuclide inventories discussed in Appendix E.

(c)The assumed HEPA filter transmission factor for each building is 5.0×10^{-4} .

(d)All releases are assumed to occur in a 1-hr period for comparison purposes.

(e)Dash indicated that no calculations were attempted for the incident listed.

While it is beyond the scope of this study to try to evaluate every potential accident for decommissioning a reference BWR, operational, natural phenomena, and indirect man-related events are discussed in the following subsections. Multiple failure-event accidents are not considered; that is, for each accident, only one failure event is analyzed.

N.2.2.1 Postulated Operational Accidents

Accidental atmospheric releases from incidents such as equipment failure, human error, or service conditions during routine immediate dismantlement tasks are discussed in this section. The accident analyses are based on the assumed technologies for immediate dismantlement tasks discussed in Appendix I.

Explosion of LPG Leaked from a Front-End Loader. The use of a liquified propane gas (LPG) -powered front-end loader for loading concrete rubble and moving light equipment is assumed in this study, since this method is consistent with current industrial methods. It is anticipated that by the time large scale decommissioning operations are required, an alternative method with a safer fuel supply will be available and used instead of the LPG-powered front-end loader.

Flammability limits for LPG range from a mixture of 2.1 volume % to a mixture of 9.5 volume %.^(15,16) Information on the rate of pressure rise from explosions of various mixtures of propane in air are given in Table N.2-17.

TABLE N.2-17. LPG Explosion Information^(a)

<u>Concentration in Air (volume %)</u>	<u>Pre-Detonation Pressure (kPa)</u>	<u>Rate of Pres- sure Rise (kPa/sec)</u>
3	510	8 600
5	660	17 000
7	330	240

(a) Summarized from information in Reference 17.

An accidental leak of LPG is postulated to occur during the loading of concrete rubble in the Reactor Building. The volume of the top level of the Reactor Building is about $3.3 \times 10^4 \text{ m}^3$, and about 1300 kg of propane are required to produce a flammable mixture in this volume of air. This amount of LPG far exceeds the fuel holding capacity of the front-end loader assumed in this study. Therefore, a realistic, smaller LPG leak of 4.5 kg is assumed, which would require about 110 m^3 of air to produce a flammable mixture. This volume is about 0.3% of the total volume of the top level of the Reactor Building. An explosion of this mixture is calculated to produce a maximum pressure surge of less than 240 kPa, which would cause no structural damage to the building HEPA filter system, unless the explosion occurred after the mixture is drawn into the building ventilation ductwork. Thus, the postulated explosion of leaked LPG is assumed to occur in the building ventilation ductwork where the force of the explosion is assumed to cause a failure of the filters and blowers. Loss of the ventilation blowers would aerodynamically isolate the building from the atmosphere. Remedial action, such as sealing off vent outlets, would further limit the atmospheric release.

Radioactive contamination deposited on the HEPA filters and in the ductwork is assumed to be made airborne by the force of the LPG explosion. The explosion is postulated to occur near the end of segmenting the activated concrete in the sacrificial shield, when the filters and vents have collected the maximum quantity of reference radionuclide inventory 3 (see Table E.1-3). Information about the amount of radioactivity on the duct surfaces is not available, thus it is assumed to equal the amount of radioactivity collected on the HEPA filters.

The mass of material deposited on HEPA filters without causing serious operational problems, such as an excessive pressure drop, vary considerably with the filter construction and the particle size of the deposited material.^(18,19) The amount of deposited material per filter assumed for this study is 2.3 kg of reference radionuclide inventory 3, with a contamination level of $4.7 \times 10^{-2} \text{ Ci/m}^3$.

If the pre-filters and filters in both exhaust filter banks rupture simultaneously (with 50 filters per bank), and assuming that the ductwork does not

rupture, about 4.3×10^{-3} Ci are released from the building to the atmosphere. If an equal amount of material is released from the walls of the ductwork, a total of 8.6×10^{-3} Ci of reference inventory 3 is released to the atmosphere. An explosion of leaked LPG resulting in this magnitude of release is estimated to have a "low" frequency of occurrence.

Vacuum Filter-Bag Rupture. Sharp objects, such as metal shards, could rupture a filter-bag during surface cleaning operations involving the use of a vacuum cleaner. When the filter-bag is ruptured, all of the collected material in the bag is assumed to become airborne in the building because of the mechanical and aerodynamic forces of the $1.4\text{-m}^3/\text{min}$ vacuum cleaner air flow. To maximize the calculation of the atmospheric release, the bag rupture is assumed to occur at the time just prior to bag change (i.e., when the filter bag is full). For the Reactor Building, about 34 Ci of reference radionuclide inventory 5 are calculated to cover $2.5 \times 10^3 \text{ m}^2$ of floor surface. If vacuuming is assumed to remove 0.5 of the surface contamination, about 17 Ci are removed from the Reactor Building. Frequent filter-bag changes are anticipated in dusty and dirty areas. Assuming 10 bag changes during the entire task, each filter-bag could contain an average of 1.7 Ci. Assuming all 1.7 Ci becomes airborne during the bag rupture event, a total of 8.5×10^{-4} Ci of reference radionuclide inventory 5 (see Table E.2-9) would be transmitted to the environment through the building HEPA filter. This accident is estimated to have a "medium" frequency of occurrence.

Contamination Control Envelope Rupture. Rupture of the contamination control envelope (greenhouse) during oxyacetylene cutting is assumed to have a "high" frequency of occurrence. Small globules of molten metal have been measured to be projected up to 9 m from a cutting operation.⁽²⁰⁾ It is postulated that such molten particles penetrate the sheet plastic walls and increase the leakage from the contamination control envelope from 0.1 to 0.5. This increase in leakage is assumed to occur during the removal of the reactor vessel (Task 12 in the Reactor Building) when the highest specific activity material is being cut in air. Since this task is remotely controlled, it is assumed that a large leak could occur for one hour of cutting before it is detected. Thus, 5.6×10^{-1} Ci are vaporized and made airborne, and

2.8×10^{-1} Ci leave the control envelope to enter the building ventilation. If the building HEPA system works as designed, the total atmospheric release is calculated to be 1.4×10^{-4} Ci of reference radionuclide inventory 2 (see Table E.1-2).

Oxyacetylene Explosion. Oxyacetylene cutting torches are assumed to be used for removing and segmenting various steel components during immediate dismantlement of the reference BWR. Acetylene gas has a flammability range of 2.5 to 80 volume % mixed in air.^(21,22) 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 rise rates from three acetylene/air mixtures in the explosive range are shown in Table N.2-18.⁽¹⁷⁾

TABLE N.2-18. Example Explosive Data for Acetylene/Air Mixtures

<u>Percent of Acetylene in Mixture</u>	<u>Maximum Explo- sive Pres- sure (kPa)</u>	<u>Maximum Rate of Explosive Pressure Rise (kPa/sec)</u>
5	540	17 000
13	1 000	8 300
20	690	9 700

Oxyacetylene explosions can occur from such causes as flow reversals, nozzle obstructions, and flashbacks (a flare going back up the gas hose).^(23,24) The accident is postulated to occur during the removal of the activated portion of the reactor vessel in air prior to segmenting the removed sections under water. This operation is conducted in a contamination control envelope with an assumed volume of 1000 m^3 . The airborne material generated as a result of the explosion is assumed to fill this volume with an airborne concentration of $1.0 \times 10^{-1} \text{ g/m}^3$.⁽⁵⁾ Thus, a total of 100 g of radioactive carbon-steel dust could fill the contamination control envelope. With a radioactivity level of $1.1 \times 10^2 \text{ Ci/m}^3$ for activated carbon steel of reference radionuclide inventory 2 (see Table E.1-2), and a steel density of $8 \times 10^6 \text{ g/m}^3$, the specific activity of this carbon-steel is calculated to be $1.4 \times 10^{-5} \text{ Ci/g}$. Thus, 100 g of steel dust contains about $1.4 \times 10^{-3} \text{ Ci}$.

It is reasonable to assume that the contamination control envelope HEPA filter system undergoes routine maintenance to ensure its full effectiveness. The mass of material deposited on filters without causing flow restrictions or an excessive pressure drop (>980 Pa of water gauge at rated flows) varies with filter design and the properties of the deposited material.^(18,19) A HEPA filter dust loading of 2.3 kg is assumed (the same amount assumed for the LPG explosion accident). Three pairs of HEPA and roughing filters in the contamination control envelope are calculated to contain about 14 kg of activated carbon-steel dust. If all of this material is made airborne by the oxyacetylene explosion, and using the specific activity of this carbon-steel of 1.4×10^{-5} Ci/g, a total of about 2.0×10^{-1} Ci are made airborne from the filters.

Some radioactivity will be deposited on the inside surfaces of the contamination control envelope, and it is assumed to be made airborne in the Reactor Building by the explosion. The total radioactivity generated during Task 12 oxyacetylene cutting is 22 Ci (see Table N.2-7). Assuming that 0.2% of this radioactivity is deposited on the interior surfaces of the control envelope, and that all of it is made airborne by the explosion, an additional 4.4×10^{-2} Ci could enter the building air.

The total radioactivity made airborne in the Reactor Building by the postulated oxyacetylene explosion is found by summing the three previous airborne generation terms. They are: the radioactivity generated by the explosion inside the contamination control envelope (1.4×10^{-3} Ci), the radioactivity removed from the control envelope HEPA filter system by the explosion (2.0×10^{-1} Ci), and the radioactivity removed from interior surfaces in the control envelope (4.4×10^{-2} Ci). The total airborne radioactivity in the building is 2.4×10^{-1} Ci.

The effect of the transmitted shock wave from the explosion on the building HEPA filters is next investigated. The transmitted shock wave pressure is given in terms of shock wave Mach Number as:⁽²⁵⁾

$$P_1 = \frac{P_t(\gamma-1)}{2\gamma M_t^2 - (\gamma-1)} \quad (N.12)$$

where:

- P_1 • the transmitted shock wave pressure in the gas downstream, Pa
- P_t • the static pressure, Pa
- γ • the specific heat ratio C_p/C_v , unitless
- M_t • the transmitted shock wave Mach Number, unitless.

Assuming, 100 kPa as the static pressure and 2.5 as the Mach Number, the downstream pressure is calculated to be 145 kPa. The Reactor Building is designed to withstand pressure gradients resulting from accident situations. It is felt that the building volume is large enough to reduce the shock wave enough by the time it reaches the building HEPA filter system so that no damage to the HEPA system occurs. Thus, the amount of radioactivity released to the environment is found by multiplying the amount of material airborne in the building by the building HEPA filter transmission factor. The total atmospheric release is 1.2×10^{-4} of reference radionuclide inventory 2. An oxy-acetylene explosion resulting in this magnitude of airborne release is judged to have a "medium" frequency of occurrence.

Contaminated Sweeping Compound Fire. Sweeping compound is composed of sawdust treated with oil or other additives to enhance the collection of loose surface contamination. A fire is postulated to occur in used sweeping compound containing radioactivity removed from floor surfaces. The highest contamination level encountered in sweeping occurs on isolated floor surfaces in the Reactor Building and the Radwaste and Control Building with 2.5×10^{-2} Ci/m² of reference radionuclide inventory 5. It is assumed that an average of 2.5×10^{-4} m³ of sweeping compound is used for each m² of surface swept. It is also assumed that the density of the sweeping compound is about the same as light wood, or about 6.0×10^5 g/m³.⁽²⁶⁾ The total floor surfaces with this contamination level are: 1.2×10^3 m² in the Reactor Building and 3.6×10^{-3} m² in the Radwaste and Control Building. Thus, a total of 1.2 m³, or 7.2×10^2 kg of sweeping compound are required. It is reasonable to assume that this total amount will be stored in smaller quantities, therefore a fire is postulated to occur in 0.5 m³ (or 3.0×10^2 kg) of used sweeping compound.

Assuming that half of the surface contamination is removed by the sweeping compound, a total of 15 Ci of reference radionuclide inventory Number 5 is

contained in 0.5 m^3 of used sweeping compound. The release fraction from the fire is assumed to be similar to that measured from burning waste, 1.5×10^{-4} .⁽²⁷⁾ Thus, about 2.2×10^{-3} Ci of reference radionuclide inventory 5 (see Table E.2-9) are airborne in the Radwaste and Control Building, and 1.1×10^{-6} Ci are released to the atmosphere through the building HEPA filter system. A release of this magnitude from a fire in contaminated sweeping compound is estimated to have a "medium" frequency of occurrence.

Gross Leak During Loop Chemical Decontamination. A massive failure of reactor piping during loop chemical decontamination is judged to have a "low" frequency of occurrence. There have been no such accidents in 30 years of experience with nuclear systems. Massive failures are unknown in high-pressure central steam boilers, for which there is considerably more operating experience than is currently available in the nuclear industry.⁽²⁸⁾

Pipe failures can result from improper system designs, the use of poor quality materials, base metal defects, fabrication defects, or inadequate service conditions.⁽²⁹⁾ The loop chemical decontamination takes place about 40 years after the piping systems were put in service. In chemical decontamination, a caustic chemical is circulated in piping and equipment to remove built-up internal surface contamination. The chemical solution could deteriorate the pipe or the welds and create localized leaks. Accidental leakage could also be introduced because of human error during the chemical decontamination.

Liquid and spray leaks are assumed to occur as part of the routine release analysis. A gross leak about 10 times larger than the spray leak is assumed for this accident. Thus, 1% of the liquid in the system is assumed to be made airborne. Such a leak would be large enough to be easily detected and is assumed to last only 1 hour. The largest routine spray leak is calculated to occur in the Radwaste and Control Building during the chemical decontamination of equipment. A total of 2.1×10^{-4} Ci/hr (see Table N.2-9) are calculated to be made airborne in the building during routine operation. The accidental leak is 10 times this value, or 2.1×10^{-3} Ci/hr. At the end of one hour, 2.1×10^{-3} Ci of material are airborne in the building, or 1.0×10^{-6} Ci of reference radionuclide inventory 4 (see Table E.2-1) are calculated to be released from the building HEPA system.

Filter Damage from Blasting Surges. The Reactor Building and its ventilation system are designed to withstand pressure gradients from accident situations. However, some signs of pressure surge damage from concrete blasting operations were observed during decommissioning of the Elk River Reactor.⁽³⁰⁾ There was a partial collapse of a prefilter in a local contamination control system. Thus, an airborne release is calculated for a postulated level of filter damage resulting from blasting pressure surges.

Using the same contamination control envelope as defined for the oxyacetylene explosion, a total of 14 kg of radioactive dust is assumed to be held on the HEPA and roughing filter. Blasting operations are assumed to be used for removal of activated concrete in the sacrificial shield. Assuming a concrete density of $2.5 \times 10^3 \text{ kg/m}^3$ and a radioactivity level of $4.7 \times 10^{-2} \text{ Ci/m}^3$ of reference radionuclide inventory 3 (see Table E.1-3), the specific activity of the concrete dust on the filters is calculated to be $1.9 \times 10^{-8} \text{ Ci/g}$. A total of $2.6 \times 10^{-4} \text{ Ci}$ is calculated to be in the 14 kg of dust on the filter. If all of this material is made airborne, and if there is no damage to the building HEPA systems from the pressure surge, a total of $1.3 \times 10^{-7} \text{ Ci}$ could be released to the environment.

A pressure surge from blasting that would result in this magnitude of release is judged to have a "medium" frequency of occurrence.

Combustible Waste Fire. Absorbent materials, such as rags or paper wipes, are assumed to be used for a variety of purposes during immediate dismantlement and discarded after use. Materials that have come into contact with contaminated surfaces hold small quantities of radionuclides. Anticontamination clothing (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 contaminated combustible 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 that one cubic meter containing $8.0 \times 10^{-2} \text{ Ci}$ of reference radionuclide inventory 5 (see Table E.2-9) are involved in the fire. The maximum fractional airborne

release measured during the burning of contaminated waste under similar conditions was 1.5×10^{-4} .⁽²⁷⁾ Thus it is estimated that up to 1.2×10^{-5} Ci could be released to the ventilation system from this fire, and that about 6.0×10^{-9} could be released to the environment. A fire that results in a release of this magnitude is estimated to have a "high" frequency of occurrence.

Detonation of Unused Explosives. Removing the sacrificial shield is accomplished by using explosive charges on the concrete after first cutting the steel skin with an oxyacetylene torch. Reinforcing steel in the concrete is cut with an oxyacetylene torch after blasting. The use of explosive charges means that special procedures are required including the use of water sprays, blasting mats, and contamination control envelopes.

If a positioned charge explodes when the water sprays are off and after the contamination control envelope has been removed, a temporary dust loading in a limited air volume will result. For this accident, it is postulated that one charge detonates and fills an air volume of 500 m^3 with a dust loading of $1.0 \times 10^{-1} \text{ g/m}^3$. The radioactivity level of the activated concrete in the sacrificial shield is $4.7 \times 10^{-2} \text{ Ci/m}^3$, and with a density of $2.5 \times 10^{-3} \text{ kg/m}^3$ the specific activity is $1.9 \times 10^{-8} \text{ Ci/g}$. Thus, 9.5×10^{-7} Ci of reference radionuclide inventory 3 (see Table E.1-3) are made airborne in the building and 4.8×10^{-10} Ci are calculated to be released to the environment.

It should be noted that there are additional nonradiological hazards to employees if they are present at the time of the accidental detonation. An atmospheric release from an accidental explosive charge detonation of this magnitude is judged to have a "medium" frequency of occurrence.

Ion Exchange Resin Accidents. Both powdered and beaded resins are used in the liquid radwaste systems at the reference BWR. Expended ion exchange resins are handled by the procedures described in the FSAR.⁽³¹⁾ The spent resins are backwashed to a spent resin tank or to a phase separator where they are allowed to accumulate over a period of time to optimize batch processing and to permit radioactive decay. They are then centrifuged and solidified in shipping containers for offsite shipment. Since the spent resins are hydro-pneumatically transferred and are stored in sealed tanks, there appears to be no danger of combustion. Only accidents involving the handling of spent

resins appear to be likely. Such accidents would lead to little airborne release because of the liquid nature of the wastes involved. Thus, no further analysis is attempted in this study.

Loss of Services. Loss of services, such as water supply, electrical power, or air flow, are immediately apparent to the workers performing the various dismantlement tasks. In the case of manual tasks, the operator would discontinue the task immediately upon the loss of services. Most remotely controlled tasks would be monitored frequently enough to almost immediately detect the loss of a service. Since the loss of services constitutes a lesser magnitude release than other postulated accidents, no further analysis is attempted in this study.

N.2.2.2 Natural Phenomena

The reference BWR is designed to meet a criterion for protection of the buildings and systems against natural phenomena. The following statement of this design criterion is made in the FSAR: "Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami and seiches without loss of capability of perform their safety functions."⁽³²⁾

Furthermore, the designs of structures, systems, and components are based on withstanding the most severe natural phenomena recorded for the site plus appropriate margins to account for uncertainties in the recorded data. It is assumed that the structural integrity of the plant is preserved during the operational lifetime of the plant. It is further assumed that the structural integrity is preserved during decommissioning as long as required for safety. It is estimated that, while natural phenomena can cause severe damage to the reference BWR resulting in severe radiological public safety impacts, they are low probability events. Also, the impact of these events will be less than the impacts calculated in the FSAR design-basis accidents for the operating BWR when nuclear fuel is present in the plant. For this reason, no detailed analysis of the impacts of these phenomena is attempted beyond the brief discussions that follow.

Floods. The FSAR considers runoff floods, surges, seiches, wave action, tsunamis, wind-generated wave action, and combinations of these phenomena. Also, a probable maximum flood is proposed. The plant is designed to be safe from damage by such a flood or other potential flooding.

Earthquakes. All systems and components designated as seismic Category I are designed to exhibit no loss of functions for earthquake ground accelerations of 0.25 g. Category I structures at the reference BWR include the Reactor Building, the control room tower, the radwaste area, and the portions of the Turbine Generator Building that protect the steam lines.

It is estimated that the probability of an earthquake that would occur and cause damage to the plant is 10^{-7} per year.⁽³³⁾ At this probability, earthquake-induced accidents are extremely unlikely.

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.⁽³⁵⁾

Tests at Los Alamos showed that a 0.2-m x 0.2-m x 0.08-m HEPA filter can withstand a 9-sec pressure pulse with the maximum pressure of 21 kPa and hold without visible damage or reduction in efficiency.⁽³⁴⁾ This pressure pulse is substantially greater than the pulse the filter would experience in the event of a design-basis tornado.

High Winds. The design-basis wind velocity for all Category I and Category II structures is 168 km/hr at 9.1 m above nominal ground elevation. For this study it is assumed that immediate dismantlement tasks do not reduce the ability of these structures to withstand high winds, and no releases are anticipated.

N.2.2.3 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 risk of damage for aircraft crashes is

low for most reactor sites.⁽³⁵⁾ Moreover, the risk is not escalated by dismantlement operations. The aircraft crash probability for most plants is very low (10^{-6} to 10^{-8} per year) and no further analysis is attempted.

N.2.2.4 Man-Caused Events

Accidental airborne releases of radioactivity resulting from man-caused events could cover a wide spectrum of magnitudes ranging from releases induced by casual trespassers to releases induced by armed terrorists. A detailed analysis of man-caused events during decommissioning is beyond the scope of this study since an analysis of human motives is difficult to conduct. Thus, no airborne release calculations are made.

N.2.2.5 Radiation Doses from Postulated Immediate Dismantlement Accidents

Radiation doses to the maximum-exposed individual from accidental releases of radionuclides during immediate dismantlement are calculated using the radiation dose models discussed in Appendix F. The first-year doses and the fifty-year committed dose equivalents to the maximum-exposed individual from inhalation for the accidents listed in Table N.2-16 are listed in Table N.2-19. Each accidental release is assumed to occur for one hour so that a comparison of the accidents can be made. The atmospheric dispersion ($\bar{X}/Q'$) value used for the accidents is 6.5×10^{-4} sec/m³.

N.3 PASSIVE SAFE STORAGE

The second decommissioning alternative considered in this study for the reference BWR is passive safe storage. This alternative is divided into three potential phases: preparations, continuing care, and probable deferred dismantlement. Preparations for passive safe storage include tasks designed to immobilize all remaining radioactive contamination and deactivate all unneeded equipment. Continuing care permits decay of short-lived radioactivity, thus reducing the occupational radiation doses associated with probable deferred dismantlement. After a finite period of continuing care, deferred dismantlement is assumed to be required for termination of the nuclear facility license. Details of the tasks and requirements of passive safe storage are discussed in Appendix J.

TABLE N-2.19. Radiation Doses to the Maximum-Exposed Individual from Accidental Atmospheric Releases During Immediate Dismantlement

Incident	Reference Radionuclide Inventory Number(a)	Total Atmospheric Release (Ci/hr)(b)	First-Year Dose (rem)					Fifty-Year Committed Dose Equivalent (rem)				
			Total Body	Bone	Lung	Thyroid	GI-LLI	Total Body	Bone	Lung	Thyroid	GI-LLI
Explosion of LPG Leaked from a Front-End Loader	3	8.6×10^{-3}	7.8×10^{-7}	9.0×10^{-7}	7.9×10^{-5}	3.6×10^{-10}	2.3×10^{-6}	1.5×10^{-6}	4.0×10^{-6}	1.5×10^{-4}	3.0×10^{-10}	3.6×10^{-10}
Vacuum Filter-Bag Rupture	5	8.5×10^{-4}	2.0×10^{-6}	3.2×10^{-6}	8.3×10^{-5}	3.3×10^{-6}	1.3×10^{-6}	9.9×10^{-6}	3.5×10^{-5}	1.8×10^{-4}	3.3×10^{-6}	1.3×10^{-6}
Contamination Control Envelope Rupture	2	1.4×10^{-4}	1.5×10^{-8}	1.0×10^{-8}	1.0×10^{-6}	9.3×10^{-12}	3.8×10^{-8}	2.4×10^{-8}	3.7×10^{-8}	1.9×10^{-6}	9.3×10^{-12}	3.8×10^{-8}
Oxyacetylene Explosion	2	1.2×10^{-4}	1.3×10^{-8}	8.8×10^{-9}	8.7×10^{-7}	7.9×10^{-12}	3.3×10^{-8}	2.1×10^{-8}	3.2×10^{-8}	1.6×10^{-6}	7.9×10^{-12}	3.3×10^{-8}
Contaminated Sweeping Compound Fire	5	1.1×10^{-6}	2.6×10^{-9}	4.2×10^{-9}	1.1×10^{-7}	4.3×10^{-9}	1.7×10^{-9}	1.3×10^{-8}	4.6×10^{-8}	2.3×10^{-7}	4.3×10^{-9}	1.7×10^{-9}
Gross Leak During Loop Chemical Decontamination	4	1.0×10^{-6}	1.4×10^{-9}	1.0×10^{-9}	9.8×10^{-8}	2.8×10^{-13}	3.1×10^{-9}	1.4×10^{-9}	1.4×10^{-9}	2.1×10^{-7}	2.8×10^{-13}	3.1×10^{-9}
Filter Damage from Blasting Surges	3	1.3×10^{-7}	1.2×10^{-11}	1.4×10^{-11}	1.2×10^{-11}	5.4×10^{-15}	3.5×10^{-11}	2.2×10^{-11}	6.0×10^{-11}	2.3×10^{-9}	5.4×10^{-15}	3.5×10^{-11}
Combustible Waste Fire	5	6.0×10^{-9}	1.4×10^{-11}	2.3×10^{-11}	5.9×10^{-10}	2.4×10^{-11}	9.0×10^{-12}	7.0×10^{-11}	2.5×10^{-10}	1.2×10^{-9}	2.4×10^{-11}	9.0×10^{-12}
Detonation of Unused Explosives	3	4.8×10^{-10}	4.3×10^{-14}	5.0×10^{-14}	4.4×10^{-12}	2.0×10^{-17}	1.3×10^{-13}	8.1×10^{-14}	2.2×10^{-13}	8.6×10^{-12}	2.0×10^{-17}	1.3×10^{-13}

(a) These numbers refer to the reference radionuclide inventories discussed in Appendix E.

(b) All releases are assumed to occur in a 1-hr period for comparison purposes.

In this section, the atmospheric releases of radioactivity and the resulting radiation doses to the public from both routine tasks and postulated accidents during preparations for passive safe storage are presented. This analysis uses the same methods described in Section N.2 for immediate dismantlement.

N.3.1 Routine Tasks During Preparations for Passive Safe Storage

The task schedule and sequence during preparations for passive safe storage of the reference BWR are shown in Figure J.4-1. Several of the tasks during preparations for passive safe storage are identical to those analyzed previously in Section N.2 for immediate dismantlement. In the cases where the tasks for these two decommissioning modes are identical, the atmospheric releases and resulting radiation doses from the task during preparations for passive safe storage are assumed to be identical to those reported in Section N.2. A listing of tasks for preparations for passive safe storage that shows whether a new safety analysis is required is given in Table N.3-1.

The following subsections contain discussions of the methods used to calculate the atmospheric releases and resulting public radiation doses during preparations for passive safe storage of the reference BWR. The data for and the results of new analyses for those tasks that are unique to preparations for passive safe storage are presented; summed doses for all tasks similar to immediate dismantlement are also given.

N.3.1.1 Methods and Data for Calculating Atmospheric Releases of Radioactivity

The tasks for preparations for passive safe storage are analyzed using the same airborne release mechanisms discussed previously in Section N.2.1.1 for immediate dismantlement. The summary of the analyses is shown previously in Table N.2-2, including the general activities analyzed. Four general activities that are unique to preparations for passive safe storage (i.e., manual decontamination, sealing building areas, isolating equipment, and spray painting to immobilize surface radioactive contamination) are not anticipated to result in an atmospheric release; thus, they are not analyzed further in this section.

TABLE N.3-1. Safety Analysis Requirements for Preparations for Passive Safe Storage

Building/Task	Analyzed in Section N.2	New Analysis Required
Reactor Building		
1. Comprehensive Radiation Survey	✓	
2. Discharge and Ship Fuel	✓	
3. Drain Suppression Pool to Radwaste; Water-Jet Clean	✓	
4. Drain Reactor Well Pool to Radwaste; Water-Jet Clean	✓	
5. Chemical Decon Reactor Water Recirculation and Cleanup Systems	✓	
6. General Cleanup		✓
7. Drain Contaminated Systems to Radwaste	✓	
8. Chemical Decon Residual Heat Removal, Low- and High-Pressure Core Spray Systems	✓	
9. Drain Dryer and Separator Pool to Radwaste; Water-Jet Clean	✓	
10. Chemical Decon Drain Systems	✓	
11. Drain Spent Fuel Pool to Radwaste; Water-Jet Clean	✓	
12. Chemical Decon Spent Fuel Pool Cooling and Cleanup Systems	✓	
13. Cover and Seal Spent Fuel Pool; Dryer and Separator Pool		✓
14. Seal Equipment and Personnel Hatches into Primary Containment		✓
15. Decontaminate HVAC, Electrical, Misc. Steel Structures and Equipment and Concrete; Apply Protective Paint		✓
16. Isolate and Seal Equipment, Piping, Rooms, Stacks, HVAC Ducts, Rail and Steam Tunnels		✓
17. Seal Drywell Top Head and Unneeded Reactor Building Doors		✓
18. Install HEPA-Filtered Vents		✓
19. Deactivate Unnecessary Utilities		✓
20. Install Intrusion, Radiation Monitoring, and Fire Alarm Systems		✓
21. Final Radiation Survey	✓	
Turbine Generator Building		
1. Comprehensive Radiation Survey	✓	
2. General Cleanup	✓	
3. Drain Contaminated Systems to Radwaste; Water-Jet Clean Condensate Storage Tanks	✓	
4. Drain Condenser to Radwaste; Water-Jet Clean		✓
5. Chemical Decon Drain Systems	✓	
6. Decontaminate HVAC, Electrical, Misc. Steel Structures and Concrete; Apply Protective Paint		✓
7. Isolate and Seal Equipment, Piping, Rooms, Stack, and HVAC Ducts		✓
8. Install HEPA-Filtered Vents		✓
9. Deactivate Unnecessary Utilities		✓
10. Install Intrusion, Radiation Monitoring and Fire Alarm Systems		✓
11. Final Radiation Survey	✓	
Radwaste and Control Building		
1. Comprehensive Radiation Survey	✓	
2. General Cleanup	✓	
3. Decontaminate Equipment External Surfaces; Apply Protective Paint		✓
4. Decontaminate Electrical Equipment, Misc. Steel Structures and HVAC; Apply Protective Paint		✓
5. Decontaminate Concrete; Apply Protective Paint		✓
6. Chemical Decon Drain Systems	✓	
7. Install Temporary Radwaste System	✓	
8. Chemical Decon Equipment	✓	
9. Install HEPA-Filtered Vents		✓
10. Deactivate Unnecessary Equipment and Utilities		✓
11. Isolate and Seal Equipment and Areas		✓
12. Install Intrusion, Radiation Monitoring and Fire Alarm Systems		✓
13. Final Radiation Survey	✓	

N.3.1.2 Atmospheric Releases from Preparations for Passive Safe Storage

Public safety analysis data for the task preparations for passive safe storage are listed in Table N.3-2. The calculated atmospheric releases from the tasks are listed in Table N.3-3. Radiation doses to the public resulting from these releases are discussed in the following subsection.

N.3.1.3 Public Radiation Doses from Routine Tasks During Preparations for Passive Safe Storage

First-year doses and fifty-year committed dose equivalents to both the maximum-exposed individual and the population residing within 80 km of the site are calculated using the radiation dose models discussed in Appendix F. Each routine atmospheric release during preparations for passive safe storage is assumed to occur at a constant rate for a 1-year period, so direct comparisons of the public radiation doses from the various tasks can be made. The organs of reference for which radiation doses are calculated include: total body, bone, lung, thyroid, and GI-tract (lower large intestine [GI-LLI]).

Radiation doses to the maximum-exposed individual are listed in Table N.3-4; those to the population are listed in Table N.3-5. These tables list the doses from the tasks that are unique to preparations for passive safe storage and that result in an atmospheric release as well as the total doses from the tasks that are identical to immediate dismantlement, which are summed and listed separately for each building.

N.3.2 Postulated Accidents During Preparation for Passive Safe Storage

A comparison of the postulated accidents for immediate dismantlement and for passive safe storage is presented in Table N.3-6. The accidents are listed by decreasing magnitude of the calculated atmospheric release. A general estimate of the frequency of occurrence of the level of the atmospheric release associated with each postulated accident is also given. The frequency of occurrence is based on both the probability of the accident and on the probability of an atmospheric release of the calculated magnitude. The frequency of occurrence is listed as "high" if the occurrence of a release of similar or greater magnitude per year is greater than 10^{-2} , as "medium" if between

TABLE N.3-2. Public Safety Analysis Data for Preparations for Passive Safe Storage^(a)

Building/Task	General Method	Operational Data			Comments	Operational Time (hr)	Reference Radionuclide Inventory Number ^(b)	Contamination Levels, C _s (Ci/m ²)	
Reactor Building		Surface Area (m ²)						% of Surface Area	
								0.25 x 10 ⁻⁵ Ci/m ²	0.25 x 10 ⁻² Ci/m ²
6 General Cleanup	Sweeping Vacuuming	1.6 x 10 ³			Assumed to be 65% of effort required for Immediate Dismantlement	1.0 x 10 ¹	5	50	
		1.6 x 10 ³				5.0 x 10 ⁰	5	50	50
15 Decontaminate HVAC, Electrical, Misc Steel Structures and Equip and Concrete, Apply Protective Paint		Cleaning Rate (m ² /hr)	Surface Area (m ²)	Liquid Volume (m ³)					
	Water Jet Manual Operating Spray Painting	4.6 x 10 ¹	2.1 x 10 ³ N/A(c) N/A	6.3 x 10 ¹	Water-Jet used on equipment and Concrete, No airborne release anticipated	5.3 x 10 ¹ N/A N/A	5 N/A N/A	2.5 x 10 ⁻³ N/A N/A	
Turbine Generator Building									
4 Drain Condenser to Radwaste, Water-Jet Clean	Draining		N/A		No airborne release anticipated	N/A	N/A	N/A	
		Cleaning Rate (m ² /hr)	Surface Area (m ²)	Liquid Volume (m ³)					
	Water Jet	4.6 x 10 ¹	1.0 x 10 ⁴	3.0 x 10 ²	25% of surface assumed cleaned for Immediate Dismantlement	2.2 x 10 ²	4	5.0 x 10 ⁻²	
6 Decontaminate HVAC, Electrical, Misc Steel Structures and Concrete, Apply Protective Paint		Cleaning Rate (m ² /hr)	Surface Area (m ²)	Liquid Volume (m ³)					
	Water Jet Manual Operations Spray Paint	4.6 x 10 ¹	1.7 x 10 ³ N/A N/A	5.1 x 10 ¹	Water-Jet used on equipment and concrete, No airborne release anticipated	3.7 x 10 ¹ N/A N/A	5 N/A N/A	2.5 x 10 ⁻³ N/A N/A	
Radwaste and Control Building									
3 Decontaminate Equipment External Surfaces, Apply Protective Paint	Water Jet Manual Operations Spray Paint	Cleaning Rate (m ² /hr)	Surface Area (m ²)	Liquid Volume (m ³)	Water-Jet used on equipment and concrete, No airborne release anticipated	5.0 x 10 ¹ N/A N/A	5 N/A N/A	2.5 x 10 ⁻² N/A N/A	
		4.6 x 10 ¹	2.3 x 10 ³ N/A N/A	6.9 x 10 ¹					
4 Decontaminate Electrical Equipment, Misc Steel Structures, HVAC, Apply Protective Paint	Water Jet Manual Operations Spray Paint	Cleaning Rate (m ² /hr)	Surface Area (m ²)	Liquid Volume (m ³)	Water-Jet used on surfaces No airborne release anticipated	7.0 x 10 ¹ N/A N/A	5 N/A N/A	2.5 x 10 ⁻² N/A N/A	
		4.6 x 10 ¹	3.2 x 10 ³ N/A N/A	9.6 x 10 ¹					
5 Decontaminate Concrete, Apply Protective Paint	Water Jet Manual Operations Spray Paint	Cleaning Rate (m ² /hr)	Surface Area (m ²)	Liquid Volume (m ³)	Water-Jet used on concrete No airborne release anticipated	4.4 x 10 ¹ N/A N/A	5 N/A N/A	2.5 x 10 ⁻² N/A N/A	
		4.6 x 10 ¹	2.0 x 10 ³ N/A N/A	6.0 x 10 ¹					

(a) Only tasks that are unique to preparations for passive safe storage and that result in atmospheric releases are included (A complete list of the tasks for this mode is given in Table N.3-1)

(b) Numbers refer to the reference radionuclide inventories presented in Appendix E

(c) N/A = not applicable

TABLE N.3-3. Calculated Atmospheric Releases During Preparations for Passive Safe Storage^(a)

Building/Task	General Method	Operational Time (hr)	Reference Radionuclide Inventory Number ^(b)	Airborne Generation Rate Inside Building or Control Envelope (Ci/hr)	Total Airborne Radioactivity Inside Building or Control Envelope (Ci)	Contamination Control System Transmission Factor ^(c)	Annual Atmospheric Release (Ci) ^(d)
<u>Reactor Building</u>							
6. General Cleanup	Sweeping	1.0×10^1	5	1.1×10^{-3}	1.1×10^{-2}	5×10^{-4}	5.5×10^{-6}
	Vacuuming	5.0×10^0	5	1.1×10^{-3}	5.5×10^{-3}	5×10^{-4}	2.8×10^{-6}
15. Decon HVAC, Electrical, Misc. Steel Structures and Concrete; Apply Protective Paint	Water Jet	5.3×10^1	5	1.1×10^{-5}	5.9×10^{-4}	5×10^{-4}	3.0×10^{-7}
<u>Turbine Generator Building</u>							
4. Drain Condenser to Rad-waste; Water-Jet Cleaning	Water Jet	2.2×10^2	4	3.6×10^{-4}	7.9×10^{-2}	5×10^{-5}	4.0×10^{-6}
6. Decon HVAC, Electrical, Misc. Steel Structures and Concrete; Apply Protective Paint	Water Jet	3.7×10^1	5	1.8×10^{-5}	6.7×10^{-4}	5×10^{-4}	3.3×10^{-7}
<u>Radwaste and Control Building</u>							
3. Decon Equipment External Surfaces; Apply Protective Paint	Water Jet	5.0×10^1	5	3.6×10^{-4}	1.8×10^{-2}	5×10^{-4}	9.0×10^{-6}
4. Decon Electrical Equipment, Misc. Steel Structures, HVAC; Apply Protective Paint	Water Jet	7.0×10^1	5	3.6×10^{-4}	2.5×10^{-2}	5×10^{-4}	1.2×10^{-5}
5. Decon Concrete; Apply Protective Paint	Water Jet	4.4×10^1	5	3.6×10^{-4}	1.6×10^{-2}	5×10^{-4}	8.0×10^{-6}

(a) Only tasks that are unique to preparations for passive safe storage and that result in atmospheric releases are included (a complete list of the tasks for this mode is given in Table N.3-1).

(b) Numbers refer to the reference radionuclide inventories presented in Appendix E.

(c) The building HEPA filter transmission factor is 5×10^{-4} ; the combined transmission factor for a contamination control envelope and the building HEPA filter is 5×10^{-5} .

(d) Annual releases are given for comparison purposes.

TABLE N.3-4. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine Preparations for Passive Safe Storage

Building/Task (a)	Reference Radionuclide Inventory Number(b)	Annual Atmospheric Release (Ci)(c)	First-Year Dose (rem)					Fifty-Year Committed Dose Equivalent (rem)				
			Total Body	Bone	Lung	Thyroid	GI-LLI	Total Body	Bone	Lung	Thyroid	GI-LLI
Reactor Building												
Tasks Identical to Immediate Dismantlement(d)	N/A(e)	N/A	1.2×10^{-8}	2.0×10^{-8}	2.6×10^{-8}	4.1×10^{-8}	7.2×10^{-9}	1.9×10^{-8}	4.4×10^{-8}	7.6×10^{-8}	4.3×10^{-8}	7.2×10^{-9}
6. General Cleanup	5	8.3×10^{-6}	3.5×10^{-9}	7.4×10^{-9}	6.2×10^{-9}	1.6×10^{-8}	1.7×10^{-9}	6.3×10^{-9}	1.7×10^{-8}	1.8×10^{-8}	1.7×10^{-8}	1.7×10^{-9}
15. Decon HVAC, Electrical, Misc. Steel Structures and Equipment and Concrete, Apply Protective Paint	5	3.0×10^{-7}	1.2×10^{-10}	2.4×10^{-10}	2.2×10^{-10}	5.7×10^{-10}	6.0×10^{-11}	2.3×10^{-10}	6.0×10^{-10}	6.6×10^{-10}	6.0×10^{-10}	6.0×10^{-11}
Totals	N/A	N/A	1.5×10^{-8}	2.8×10^{-8}	3.2×10^{-8}	5.8×10^{-8}	9.0×10^{-9}	2.6×10^{-8}	6.1×10^{-8}	9.5×10^{-8}	6.1×10^{-8}	9.0×10^{-9}
Turbine Generator Building												
Tasks Identical to Immediate Dismantlement(d)	N/A	N/A	4.2×10^{-9}	5.6×10^{-9}	1.1×10^{-8}	9.2×10^{-9}	3.1×10^{-9}	5.7×10^{-9}	1.0×10^{-8}	3.2×10^{-8}	9.6×10^{-9}	3.1×10^{-9}
4. Drain Condenser to Rad-waste; Water-Jet Cleaning	4	4.0×10^{-6}	1.5×10^{-9}	1.4×10^{-9}	4.8×10^{-9}	1.2×10^{-9}	1.4×10^{-9}	1.6×10^{-9}	1.5×10^{-9}	1.4×10^{-8}	1.2×10^{-9}	1.4×10^{-9}
6. Decon HVAC, Electrical, Misc. Steel Structures and Concrete; Apply Protective Paint	5	3.3×10^{-7}	1.4×10^{-10}	2.8×10^{-10}	2.4×10^{-10}	6.3×10^{-10}	6.6×10^{-11}	2.5×10^{-10}	6.6×10^{-10}	7.3×10^{-10}	6.6×10^{-10}	6.6×10^{-11}
Totals	N/A	N/A	5.8×10^{-9}	7.2×10^{-9}	1.6×10^{-8}	1.1×10^{-8}	4.6×10^{-9}	7.6×10^{-9}	1.2×10^{-8}	4.6×10^{-8}	1.1×10^{-8}	4.6×10^{-9}
Radwaste and Control Building												
Tasks Identical to Immediate Dismantlement(d)	N/A	N/A	3.3×10^{-7}	3.1×10^{-7}	1.0×10^{-6}	3.2×10^{-7}	2.9×10^{-7}	3.6×10^{-7}	3.9×10^{-7}	2.9×10^{-6}	3.3×10^{-7}	2.9×10^{-7}
3. Decon Equipment External Surfaces; Apply Protective Paint	5	9.0×10^{-6}	3.7×10^{-9}	7.7×10^{-9}	6.5×10^{-9}	1.7×10^{-8}	1.8×10^{-9}	6.8×10^{-9}	1.8×10^{-8}	2.0×10^{-8}	1.8×10^{-8}	1.8×10^{-9}
4. Decon Electrical Equipment, Misc. Steel Structures, HVAC; Apply Protective Paint	5	1.2×10^{-5}	4.9×10^{-9}	1.0×10^{-8}	8.6×10^{-9}	2.3×10^{-8}	2.4×10^{-9}	9.1×10^{-9}	2.4×10^{-8}	2.6×10^{-8}	2.4×10^{-8}	2.4×10^{-9}
5. Decon Concrete; Apply Protective Paint	5	8.0×10^{-6}	3.3×10^{-9}	6.9×10^{-9}	5.8×10^{-9}	1.5×10^{-8}	1.6×10^{-9}	6.1×10^{-9}	1.6×10^{-8}	1.8×10^{-8}	1.6×10^{-8}	1.6×10^{-9}
Totals	N/A	N/A	3.4×10^{-7}	3.4×10^{-7}	1.0×10^{-6}	3.8×10^{-7}	2.9×10^{-7}	3.8×10^{-7}	4.4×10^{-7}	3.0×10^{-6}	3.8×10^{-7}	2.9×10^{-7}

(a) A listing of preparations for passive safe storage tasks and a determination of whether they are the same as immediate dismantlement tasks is given in Table N.3-1. Only tasks that are unique to preparation for passive safe storage that result in atmospheric release are listed in detail here.

(b) Numbers refer to the reference radionuclide inventories discussed in Appendix E.

(c) For comparison purposes, dose calculations are made based on a continuous release for 1 year.

(d) A listing of the radiation doses from immediate dismantlement tasks are found in Tables N.2-10, N.2-11, and N.2-12. Radiation doses from preparation for passive safe storage that are identical to immediate dismantlement tasks are summed and listed here.

(e) N/A = not applicable. The doses shown are summed over releases of the different inventories.

TABLE N.3-5. Radiation Doses to the Population from Atmospheric Releases During Preparations for Passive Safe Storage^(a)

Building/Task ^(b)	Reference Radionuclide Inventory Number ^(d)	Annual Atmospheric Release (Ci) ^(d)	First-Year Dose (rem)					Fifty-Year Committed Dose Equivalent (rem)				
			Total Body	Bone	Lung	Thyroid	GI-LLI	Total Body	Bone	Lung	Thyroid	GI-LLI
Reactor Building												
Tasks Identical to Immediate Dismantlement ^(e)	N/A ^(f)	N/A	7×10^{-6}	1×10^{-5}	2×10^{-5}	3×10^{-5}	5×10^{-6}	1×10^{-5}	3×10^{-5}	9×10^{-5}	3×10^{-5}	5×10^{-6}
6. General Cleanup	5	8.3×10^{-6}	2×10^{-6}	5×10^{-6}	6×10^{-6}	1×10^{-5}	1×10^{-6}	4×10^{-6}	1×10^{-5}	2×10^{-5}	1×10^{-5}	1×10^{-6}
15. Decon HVAC, Electrical, Misc. Steel Structures and Equipment and Concrete; Apply Protective Paint	5	3.0×10^{-7}	8×10^{-8}	2×10^{-7}	2×10^{-7}	5×10^{-7}	4×10^{-8}	2×10^{-7}	5×10^{-7}	7×10^{-7}	5×10^{-7}	4×10^{-8}
Totals	N/A	N/A	9×10^{-6}	2×10^{-5}	3×10^{-5}	4×10^{-5}	6×10^{-6}	2×10^{-5}	4×10^{-5}	1×10^{-4}	4×10^{-5}	6×10^{-6}
Turbine Generator Building												
Tasks Identical to Immediate Dismantlement ^(e)	N/A	N/A	3×10^{-6}	3×10^{-6}	1×10^{-5}	7×10^{-6}	1×10^{-6}	4×10^{-6}	8×10^{-6}	3×10^{-5}	7×10^{-6}	2×10^{-6}
4. Drain Condenser to Rad- waste; Water-jet Cleaning	4	4.0×10^{-6}	9×10^{-7}	8×10^{-7}	5×10^{-6}	7×10^{-7}	8×10^{-7}	1×10^{-6}	9×10^{-7}	1×10^{-5}	7×10^{-7}	8×10^{-7}
6. Decon HVAC, Electrical, Misc. Steel Structures and Concrete; Apply Protective Paint	5	3.3×10^{-7}	9×10^{-8}	2×10^{-7}	2×10^{-7}	5×10^{-7}	4×10^{-8}	2×10^{-7}	5×10^{-7}	8×10^{-7}	5×10^{-7}	5×10^{-8}
Totals	N/A	N/A	4×10^{-6}	4×10^{-6}	2×10^{-5}	8×10^{-6}	2×10^{-6}	5×10^{-6}	9×10^{-6}	4×10^{-5}	8×10^{-6}	2×10^{-6}
Radwaste and Control Building												
Tasks Identical to Immediate Dismantlement ^(e)	N/A	N/A	2×10^{-4}	2×10^{-4}	1×10^{-3}	2×10^{-4}	2×10^{-4}	2×10^{-4}	3×10^{-4}	3×10^{-3}	2×10^{-4}	2×10^{-4}
3. Decon Equipment External Surfaces; Apply Protective Paint	5	9.0×10^{-6}	2×10^{-8}	5×10^{-8}	6×10^{-8}	1×10^{-7}	1×10^{-8}	5×10^{-6}	1×10^{-5}	2×10^{-5}	1×10^{-5}	1×10^{-6}
4. Decon Electrical Equipment, Misc. Steel Structures, HVAC; Apply Protective Paint	5	1.2×10^{-5}	3×10^{-6}	7×10^{-6}	8×10^{-6}	2×10^{-5}	1×10^{-6}	6×10^{-6}	2×10^{-5}	3×10^{-5}	2×10^{-5}	2×10^{-6}
5. Decon Concrete; Apply Protective Paint	5	8.0×10^{-6}	2×10^{-6}	4×10^{-6}	6×10^{-6}	1×10^{-5}	1×10^{-6}	4×10^{-6}	1×10^{-5}	2×10^{-5}	1×10^{-5}	1×10^{-6}
Totals	N/A	N/A	2×10^{-4}	2×10^{-4}	1×10^{-3}	2×10^{-4}	2×10^{-4}	2×10^{-4}	3×10^{-4}	3×10^{-3}	2×10^{-4}	2×10^{-4}

(a) The calculated doses are shown to one significant figure and are to the 3.5×10^6 people residing within a radius of 80 km.

(b) A listing of preparation for safe storage tasks and a determination of whether they are the same as immediate dismantlement tasks is given in Table N.3-1. Only tasks that are unique to preparation for safe storage that result in an atmospheric release are listed in detail in this table.

(c) Numbers refer to the reference radionuclide inventories discussed in Appendix E.

(d) For comparison purposes, dose calculations are made based on a continuous release for 1 year.

(e) A listing of the radiation doses from immediate dismantlement tasks is found in Tables N.2-13, N.2-14, and N.2-15. Radiation doses from preparations for passive safe storage tasks that are identical to the doses for immediate dismantlement tasks are summed and listed here.

(f) N/A = not applicable.

TABLE N.3-6. Comparison of Postulated Accidents for Immediate Dismantlement and Passive Safe Storage

Incident	Comments	Frequency of Occurrence(a)	Reference Radionuclide Inventory Number(b)	Immediate Dismantlement Atmospheric Release (Ci/hr)(c)	Passive Safe Storage Atmospheric Release (Ci/hr)(c)
Explosion of LPG Leaked from a Front-End Loader	4.5 kg of LPG leaked in the reactor building during the loading of activated concrete	Low	3	8.6×10^{-3}	N/A ^(d)
Vacuum Filter-Bag Rupture	Rupture of a full filter-bag during surface cleaning	Medium	5	8.5×10^{-4}	8.5×10^{-4}
Contamination Control Envelope Rupture	Leakage during reactor vessel removal (Task 12 in the Reactor Building)	High	2	1.4×10^{-4}	N/A
Oxyacetylene Explosion	Explosion during reactor vessel removal (Task 12 in the Reactor Building)	Medium	2	1.2×10^{-4}	N/A
Contaminated Sweeping Compound Fire	A fire consumes 0.5 m^3 of sweeping compound	Medium	5	1.1×10^{-6}	1.1×10^{-6}
Gross Leak During Loop Chemical Decontamination	A leak 10 times larger than the routine leak in the Radwaste and Control Building)	Low	4	1.0×10^{-6}	1.0×10^{-6}
Filter Damage from Blasting Surges	Damage from routine concrete blasting operations (Task 14 in the Reactor Building)	Medium	3	1.3×10^{-7}	N/A
Combustible Waste Fire	1 m^3 of waste is involved in a fire	High	5	6.0×10^{-9}	6.0×10^{-9}
Detonation of Unused Explosives	Detonation of a positioned charge after removal of controls (Task 15 in the Reactor Building)	Medium	3	4.8×10^{-10}	N/A

(a)The frequency of occurrence considers not only the probability of the accident, but also the probability of an atmospheric release of the calculated magnitude. The frequency of occurrence is listed as "High" if the occurrence of a release of similar or greater magnitude per year is $>10^{-2}$, as "Medium" if between 10^{-2} and 10^{-5} , and as "Low" if $<10^{-5}$.

(b)These numbers refer to the reference radionuclide inventories discussed in Appendix E.

(c)All releases are assumed to occur in a 1-hr period for comparison purposes.

(d)N/A = not applicable.

10^{-2} and 10^{-5} , and as "low" if less than 10^{-5} . Only accidents with a calculated atmospheric release are compared in Table N.3-6.

No accidents are postulated for preparations for passive safe storage that are different from the accidents that are postulated for immediate dismantlement. None of the tasks defined for passive safe storage involve removal of the activated steel or concrete, thus the accidents defined for immediate dismantlement involving these materials do not apply. The calculated radiation doses to the maximum-exposed individual from the accidents considered for preparations for passive safe storage are the same as the doses calculated for immediate dismantlement, as listed in Table N.2-19.

N.3.3 Continuing Care

Radioactive contamination during the continuing care period is assumed to be fixed firmly in place and is not readily available for airborne release. The tasks during continuing care involve surveillance, maintenance, and security with no operational components or equipment except for automatic monitoring or security systems. Thus, no routine airborne releases are anticipated during continuing care.

Only low probability accidents with causes external to the plant can result in an accidental release of radioactive materials during continuing care. Such causes include earthquakes or intrusion into the facility. The combination of the low probability of these events and the immobility of the radionuclide inventories will reduce the impact of accidents during continuing care. Thus, no analysis of postulated continuing care accidents is attempted in this study.

N.3.4 Deferred Dismantlement

Deferred dismantlement is assumed to be required after a period of continuing care to permit termination of the nuclear facility license. Since a period of continuing care will permit radioactive decay of the short-lived activation products and fission products, the inventories within the reference BWR will be greatly reduced. The time dependence of the reference radionuclide inventories is discussed in Appendix E. Thus, if the same procedures are used, the atmospheric releases during deferred dismantlement should be less than

those during immediate dismantlement. The public radiation doses from either routine deferred dismantlement tasks or postulated accidents should also be less than the radiation doses calculated for immediate dismantlement. Therefore, public safety during deferred dismantlement is not further analyzed, since the radiation doses are assumed to be equal to or less than the radiation doses from immediate dismantlement, depending on the length of the continuing care period.

N.4 ENTOMBMENT

The third decommissioning alternative considered in this study is entombment. This alternative results in removal of the radioactively contaminated portions of the buildings and equipment and utilizes onsite storage of waste by converting the primary containment vessel in the Reactor Building into a waste storage facility. Onsite storage reduces the volume of waste shipped off-site for disposal, but requires a continued license as a waste storage facility. Details of the tasks and requirements of entombment are discussed in Appendix K.

The inventories of radioactive contamination used in this public safety evaluation of entombment tasks are the shutdown inventories discussed in Appendix E. Shutdown inventories are used to maximize the calculated atmospheric releases.

In the following sections, the atmospheric releases of radioactivity from entombment are described for both routine tasks and postulated accidents. The atmospheric release analysis uses the same calculational methods described in Section N.2.1.1 for immediate dismantlement.

N.4.1 Routine Entombment Tasks

The chronological sequence of entombment tasks is listed in Table K.2-1 in Appendix K. Entombment tasks are listed with their operational time requirements for the Reactor Building, the Turbine Generator Building, and the Radwaste and Control Building. The majority of the tasks defined for entombment are identical to the tasks defined for immediate dismantlement. For the Turbine Generator Building and the Radwaste and Control Building, the only difference between entombment and immediate dismantlement tasks is the chronological

sequence of the tasks. Therefore, no new analysis of the atmospheric releases from entombment of these buildings is attempted. Furthermore, the atmospheric releases and resulting radiation doses to the public are assumed to be identical with those calculated previously for immediate dismantlement (see Section N.2). For the Reactor Building, a list of the entombment tasks considered and a determination of whether a new analysis is required is given in Table N.4-1.

Two scenarios of entombment are considered in Appendix K. In the first, it is assumed that the reactor internals are removed and shipped to a waste disposal site. In the second, the reactor internals are assumed to be left in-place. Atmospheric releases and the resulting public radiation doses are calculated only for Scenario 1. Scenario 2, with fewer tasks and smaller atmospheric releases, is not analyzed further.

The following sections contain a discussion of the methods used to calculate the atmospheric releases and resulting public radiation doses for entombment of the reference BWR. A new analysis is required only for those tasks in the Reactor Building that are unique to entombment. Dose results for the tasks that are similar to immediate dismantlement are based on the results from immediate dismantlement, with no new analysis.

N.4.1.1 Methods for Calculating Atmospheric Releases of Radioactivity

The reference entombment tasks defined in Appendix K are analyzed using the same airborne release mechanisms discussed in Section N.2.1.1 for immediate dismantlement. A summary of the reference tasks considered in this study is shown in Table N.2-2. The only task used during entombment that is not defined for immediate dismantlement is the use of a jackhammer for removing concrete. For the airborne release analysis, it is assumed that the jackhammer produces airborne radioactivity in the same manner as drilling contaminated concrete. The airborne release from drilling is defined by Equation N.11 in Section N.2.1.1. Since the jackhammer works on a small area of concrete, a vacuum exhaust system is defined to reduce the quantity of airborne radioactive concrete dust. This system is assumed to have an airflow of $1000 \text{ m}^3/\text{hr}$, and reduces the air concentration of dust by a factor of 0.3, the same factor as used for drilling. The radioactivity released to the atmosphere is found by

TABLE N.4-1. Comparison Between Entombment and Immediate Dismantlement Tasks

Building/Task ^(a)	Analyzed in Section N.2	New Analysis Required
<u>Reactor Building and Primary Containment</u>		
1. Install HEPA Filters	✓	
2. Comprehensive Radiation Survey	✓	
3. Discharge and Ship Fuel	✓	
4. Remove Dryer and Separator	✓	
5. Drain Suppression Pool to Radwaste; Water-Jet Cleaning	✓	
6. Remove Reactor Vessel Internals	✓	
7. Ship Activated Components		✓
8. Cut Suppression Pool Downcomers and Bracing		✓
9. Chemical Decon Residual Heat Removal, Low- and High-Pressure Core Spray Systems	✓	
10. Drain Contaminated Systems to Radwaste	✓	
11. Drain Reactor Well Pool Cavity to Radwaste; Water-Jet Cleaning	✓	
12. Clean Up, Stage, and Shield Hot Spots in Primary Containment	✓	
13. Chemical Decon Reactor Water Recirculating and Clean-up Systems	✓	
14. Drain Dryer and Separator Pool to Radwaste; Water-Jet Cleaning	✓	
15. Cut Suppression Chamber Accesses Through Drywell Floor (2)		✓
16. Chemical Decon Drain Systems	✓	
17. Cut Primary Containment Piping Penetrations and Seal		✓
18. Cut Drywell Bellows Access Openings		✓
19. Remove Reactor Building Piping	✓	
20. Drain Spent Fuel Pool to Temporary Radwaste; Water-Jet Cleaning	✓	
21. Chemical Decon Spent Fuel Pool Cooling and Cleanup Systems	✓	
22. Remove Liners from Spent Fuel Pool and Dryer and Separator Pool	✓	
23. Remove Reactor Building Equipment	✓	
24. Seal Equipment and Personnel Hatch Openings into Primary Containment		✓
25. Remove Reactor Building Contaminated Concrete	✓	
26. Seal Rail Tunnel, Steam Tunnel and Biological Shield Penetrations		✓
27. Seal Drywell Top Head and Reactor Building External Doors		✓
28. Remove HVAC and Disable Crane	✓	
29. Final Radiation Survey	✓	
30. Install Security and Surveillance Monitoring Equipment--Disconnect Unnecessary Utilities		✓

(a) Only the Reactor Building is considered since entombment tasks in the other buildings are identical to the immediate dismantlement tasks previously defined. Only entombment Scenario 1 is considered in this analysis.

multiplying the quantity of airborne radioactivity generated by the leakage fraction from the contamination control envelope (0.1) and by the building HEPA filter transmission factor.

N.4.1.2 Atmospheric Releases from Entombment

Radioactivity releases to the atmosphere resulting from entombment tasks at the reference BWR are calculated using the general methods and equations

described in Sections N.2.1.1 and N.4.1.1. The operational data used in this analysis is listed in Table N.4-2. Again, only those tasks assumed to result in an atmospheric release that are unique to entombment are listed, and these all occur in the Reactor Building. The calculated atmospheric releases of radioactivity are listed in Table N.4-3, together with the calculated airborne radioactivity generated inside the Reactor Building. The atmospheric releases are used in radiation dose calculations to estimate the radiation doses to the public from routine entombment tasks. The results of the radiation dose calculations are contained in the following section.

N.4.1.3 Public Radiation Doses from Routine Entombment

First-year doses and fifty-year committed dose equivalents to both the maximum-exposed individual and the population residing within 80 km of the site are calculated using the radiation dose models discussed in Appendix F. Each routine atmospheric release during entombment is assumed to occur at a constant rate for a 1-year period so that direct comparisons of the impact of entombment tasks can be made. Public radiation doses are listed for entombment Scenario 1 only. For this study, the organs of reference for which radiation doses are calculated include: total body, bone, lung, thyroid, and GI-tract (lower large intestine).

Radiation doses to the maximum-exposed individual are listed in Table N.4-4, and radiation doses to the population are listed in Table N.4-5 for entombment Scenario 1. These tables list the tasks unique to entombment Scenario 1, and the results from the tasks previously calculated for immediate dismantlement that are identical to entombment tasks.

N.4.2 Postulated Accidents During Entombment

No entombment accidents are postulated that are different from the accidents that are postulated for immediate dismantlement in Section N.2.2. The only task defined for entombment Scenario 1 that involves removal of highly activated materials is the removal of the reactor internals. This task is performed by underwater cutting methods, and no accidents resulting in an atmospheric release are postulated. Thus, entombment accidents are less severe than those postulated for immediate dismantlement. The same accidents considered

TABLE N.4-2. Public Safety Analysis Data for Entombment of the Reactor Building and Primary Containmentment

Task (a)	General Method	Safety Analysis Data			Comments	Operational Time (hr)	Reference Radionuclide Inventory Number (b)	Contamination Levels, C_s , (Ci/m ²)												
8. Cut Suppression Pool Downcomers and Bracing	Oxyacetylene Torch	Average O.D. (mm)	Cut Length L_s (m)	Kerf Width K_s (m)	Downcomers and bracing cut to fit into the lower cavity in the reactor pedestal	3.5×10^2 8.7×10^1	4 4	5×10^{-2} 5×10^{-2}												
		610	4.8×10^2	6.4×10^{-3}																
		710	1.2×10^2	6.4×10^{-3}																
15. Cut Suppression Chamber Access Through Drywell Floor	Jackhammer	Surface Area (m ²)	Concrete Volume (m ³)	Both top and bottom surfaces of concrete are contaminated. Jackhammer is used in a contamination control envelope	8.0×10^1	5	2.5×10^{-2}													
		• Concrete	1.6×10^1					1.6×10^0												
	• Steel	Oxyacetylene Torch	Cut Length L_s (m)		Kerf Width K_s (m)	Cutting Rate (m/hr)	4.0×10^1	5	2.5×10^{-2}											
			1.1×10^1		6.4×10^{-3}	2.8×10^{-1}														
17. Cut Primary Containment Piping Penetrations and Seal	Oxyacetylene Torch	Cut Length L_s (m)	Kerf Width K_s (m)	Cutting Rate (m/hr)	Pipe Penetrations are cut and covers are welded in place	2.3×10^2	4	<table><tr><td></td><td>% of Cuts</td><td></td></tr><tr><td>R. Water</td><td>Steam</td><td>Concrete</td></tr><tr><td>@ 1.1 Ci/m^2</td><td>@ $5 \times 10^{-3} \text{ Ci/m}^2$</td><td>@ $5 \times 10^{-2} \text{ Ci/m}^2$</td></tr><tr><td>19</td><td>14</td><td>67</td></tr></table>		% of Cuts		R. Water	Steam	Concrete	@ 1.1 Ci/m^2	@ $5 \times 10^{-3} \text{ Ci/m}^2$	@ $5 \times 10^{-2} \text{ Ci/m}^2$	19	14	67
			% of Cuts																	
		R. Water	Steam	Concrete																
		@ 1.1 Ci/m^2	@ $5 \times 10^{-3} \text{ Ci/m}^2$	@ $5 \times 10^{-2} \text{ Ci/m}^2$																
19	14	67																		
3.3×10^2	6.4×10^{-3}	1.4×10^0																		
18. Cut Drywell Bellows Access Openings	Oxyacetylene Torch	Cut Length L_s (m)	Kerf Width K_s (m)	Cutting Rate (m/hr)	Remote cutting inside a contamination control envelope	1.2×10^2	4	5×10^{-2}												
		2.8×10^1	6.4×10^{-3}	2.3×10^{-1}																

(a) Only tasks unique to entombment scenario 1 that results in an atmospheric release are listed. Tasks that involve welded closures, cast-in-place reinforced concrete, or other methods of sealing are not anticipated to result in an atmospheric release and are not included here.

(b) These numbers refer to the reference radionuclide inventories discussed in Appendix E.

TABLE N.4-3. Calculated Atmospheric Releases for Entombment of the Reactor Building and Primary Containment

Task (a)	General Method	Operational Time (hr)	Reference Radionuclide Inventory Number (b)	Airborne Generation Rate Inside Building or Control Envelope (Ci/hr)	Total Airborne Radioactivity Inside Building or Control Envelope (Ci)	Contamination Control System Transmission Factor (c)	Annual Atmospheric Release (Ci) (d)
8. Cut Suppression Pool Downcomers and Bracing	Oxyacetylene Torch	4.4×10^2	4	8.6×10^{-4}	3.8×10^{-1}	5×10^{-4}	1.9×10^{-4}
15. Cut Suppression Chamber Access Through Drywell Floor							
• Concrete	Jackhammer	8.0×10^1	5	9.0×10^{-7}	7.2×10^{-5}	5×10^{-5}	3.6×10^{-9}
• Steel	Oxyacetylene Torch	4.0×10^1	5	4.4×10^{-5}	1.8×10^{-3}	5×10^{-5}	8.8×10^{-8}
17. Cut Primary Containment Piping Penetrations and Seal	Oxyacetylene Torch	2.3×10^2	4	2.2×10^{-3}	5.1×10^{-1}	5×10^{-4}	2.6×10^{-4}
18. Cut Drywell Bellows Access Openings	Oxyacetylene Torch	1.2×10^2	4	7.5×10^{-5}	9.0×10^{-3}	5×10^{-4}	4.5×10^{-6}

(a) Only tasks unique to entombment scenario 1 that result in an atmospheric release are listed. Tasks that involve welded closures, cast-in-place reinforced concrete, or other methods of sealing are not anticipated to result in an atmospheric release and are not included here.

(b) These numbers refer to the reference radionuclide inventories discussed in Appendix E.

(c) The assumed building HEPA transmission factor is 5×10^{-4} . The combined transmission for a contamination control envelope and the building HEPA filter is 5×10^{-5} .

(d) For comparison purposes, dose calculations are made based on a continuous release for 1 year.

TABLE N.4-4. Radiation Doses to the Maximum-Exposed Individual From Atmospheric Releases During Routine Entombment Tasks

Building/Task ^(a)	Reference Radionuclide Inventory Number ^(b)	Annual Atmospheric Release (Ci) ^(c)	First-Year Dose (rem)					Fifty-Year Committed Dose Equivalent (rem)				
			Total Body	Bone	Lung	Thyroid	GI-LLI	Total Body	Bone	Lung	Thyroid	GI-LLI
<u>Reactor Building</u>												
Tasks Identical to Immediate Dismantlement ^(d)	N/A ^(e)	N/A	1.3×10^{-6}	1.5×10^{-6}	2.3×10^{-6}	1.9×10^{-6}	1.1×10^{-6}	1.6×10^{-6}	2.3×10^{-6}	1.2×10^{-5}	2.1×10^{-6}	1.1×10^{-6}
8. Cut Suppression Pool Down- comers and Bracing	4	1.9×10^{-4}	7.2×10^{-8}	6.7×10^{-8}	2.3×10^{-7}	5.9×10^{-8}	6.5×10^{-8}	7.8×10^{-8}	7.2×10^{-8}	6.5×10^{-7}	5.9×10^{-8}	6.5×10^{-8}
15. Cut Suppression Chamber Access Through Drywell Floor	5	9.2×10^{-9}	3.8×10^{-12}	7.9×10^{-12}	6.6×10^{-12}	1.8×10^{-11}	1.8×10^{-12}	7.0×10^{-12}	1.8×10^{-11}	2.0×10^{-11}	1.8×10^{-11}	1.8×10^{-12}
17. Cut Primary Containment Piping Penetrations and Seal	4	2.6×10^{-4}	9.9×10^{-8}	9.1×10^{-8}	3.1×10^{-7}	8.1×10^{-8}	8.8×10^{-8}	1.1×10^{-7}	9.9×10^{-8}	8.8×10^{-7}	8.1×10^{-8}	8.8×10^{-8}
18. Cut Drywell Bellows Access Openings	4	4.5×10^{-6}	1.7×10^{-9}	1.6×10^{-9}	5.4×10^{-9}	1.4×10^{-9}	1.5×10^{-9}	1.8×10^{-9}	1.7×10^{-9}	1.5×10^{-8}	1.4×10^{-9}	1.5×10^{-9}
Totals for the Reactor Building	N/A	N/A	1.5×10^{-6}	1.7×10^{-6}	2.8×10^{-6}	2.0×10^{-9}	1.2×10^{-6}	1.9×10^{-6}	2.5×10^{-6}	1.4×10^{-5}	2.2×10^{-6}	1.2×10^{-6}
<u>Turbine Generator Building^(f)</u>	N/A	N/A	5.3×10^{-7}	5.3×10^{-7}	1.6×10^{-6}	5.5×10^{-7}	5.9×10^{-7}	5.9×10^{-7}	6.5×10^{-7}	4.6×10^{-7}	5.6×10^{-7}	4.6×10^{-7}
<u>Radwaste and Control Building^(f)</u>	N/A	N/A	2.3×10^{-6}	2.2×10^{-6}	7.3×10^{-6}	2.1×10^{-5}	2.0×10^{-6}	2.5×10^{-6}	2.5×10^{-6}	2.0×10^{-5}	2.1×10^{-6}	2.0×10^{-6}

(a) A listing of entombment tasks and a determination of whether they are the same as immediate dismantlement tasks is given in Table N.4-1. Only tasks unique to entombment scenario 1 that result in an atmospheric release are listed in detail here.

(b) These numbers refer to the reference radionuclide inventories discussed in Appendix E.

(c) For comparison purposes, dose calculations are based on a continuous release for 1 year.

(d) A listing of the radiation doses from immediate dismantlement tasks is found in Table N.2-10. Radiation doses from entombment tasks that are identical to immediate dismantlement tasks are summed and listed here.

(e) N/A = Not Applicable.

(f) All entombment and immediate dismantlement tasks are the same for this building. Complete listings of the radiation dose per task are found in Tables N.2-11 and N.2-12.

TABLE N.4-5. Radiation Doses to the Population from Atmospheric Releases During Routine Entombment Tasks(a)

Building/Task ^(b)	Reference Radionuclide Inventory Number ^(c)	Annual Atmospheric Release (Ci) ^(d)	First-Year Dose (man-rem)					Fifty-Year Committed Dose Equivalent (man-rem)				
			Total Body	Bone	Lung	Thyroid	GI-LLI	Total Body	Bone	Lung	Thyroid	GI-LLI
Reactor Building												
Tasks Identical to Imme- diate Dismantlement ^(e)	N/A ^(f)	N/A	2 x 10 ⁻⁴	6 x 10 ⁻⁴	3 x 10 ⁻³	1 x 10 ⁻³	7 x 10 ⁻⁴	2 x 10 ⁻⁴	1 x 10 ⁻³	1 x 10 ⁻²	1 x 10 ⁻³	7 x 10 ⁻⁴
8. Cut Suppression Pool Downcomers and Bracing	4	1.9 x 10 ⁻⁴	4 x 10 ⁻⁵	4 x 10 ⁻⁵	2 x 10 ⁻⁴	3 x 10 ⁻⁵	4 x 10 ⁻⁵	5 x 10 ⁻⁵	4 x 10 ⁻⁵	7 x 10 ⁻⁴	3 x 10 ⁻⁵	4 x 10 ⁻⁵
15. Cut Suppression Chamber Access Through Drywell Floor	5	9.2 x 10 ⁻⁹	2 x 10 ⁻⁹	5 x 10 ⁻⁹	6 x 10 ⁻⁹	1 x 10 ⁻⁸	1 x 10 ⁻⁹	5 x 10 ⁻⁹	1 x 10 ⁻⁸	2 x 10 ⁻⁸	1 x 10 ⁻⁸	1 x 10 ⁻⁹
17. Cut Primary Contain- ment Piping Pene- trating and Seal	4	2.6 x 10 ⁻⁴	6 x 10 ⁻⁵	5 x 10 ⁻⁵	3 x 10 ⁻⁴	5 x 10 ⁻⁵	5 x 10 ⁻⁵	7 x 10 ⁻⁵	6 x 10 ⁻⁵	9 x 10 ⁻⁴	5 x 10 ⁻⁵	5 x 10 ⁻⁵
18. Cut Drywell Bellows Access Openings	4	4.5 x 10 ⁻⁶	1 x 10 ⁻⁶	9 x 10 ⁻⁷	5 x 10 ⁻⁶	8 x 10 ⁻⁷	9 x 10 ⁻⁷	1 x 10 ⁻⁶	1 x 10 ⁻⁶	2 x 10 ⁻⁵	8 x 10 ⁻⁷	9 x 10 ⁻⁷
Totals for the Reactor Building	N/A	N/A	3 x 10 ⁻⁴	7 x 10 ⁻⁴	4 x 10 ⁻³	1 x 10 ⁻³	8 x 10 ⁻⁴	3 x 10 ⁻⁴	1 x 10 ⁻³	1 x 10 ⁻²	1 x 10 ⁻³	8 x 10 ⁻⁴
Turbine Generator Building ^(g)	N/A	N/A	3 x 10 ⁻⁴	3 x 10 ⁻⁴	2 x 10 ⁻³	3 x 10 ⁻⁴	3 x 10 ⁻⁴	3 x 10 ⁻⁴	4 x 10 ⁻⁴	5 x 10 ⁻⁴	4 x 10 ⁻⁴	3 x 10 ⁻⁴
Radwaste and Control Building ^(g)	N/A	N/A	1 x 10 ⁻³	1 x 10 ⁻³	7 x 10 ⁻³	1 x 10 ⁻³	1 x 10 ⁻³	1 x 10 ⁻³	2 x 10 ⁻³	3 x 10 ⁻²	1 x 10 ⁻³	1 x 10 ⁻³

(a) The calculated doses are shown to one significant figure and are to the 3.5×10^6 people residing within a radius of 80 km from the reference site.

(b) A listing of entombment tasks and a determination of whether they are the same as immediate dismantlement tasks is given in Table N.4-1. Only tasks unique to entombment scenario 1 that result in an atmospheric release are listed in detail here.

(c) These numbers refer to the reference radionuclide inventories discussed in Appendix E.

(d) For comparison purposes, dose calculations are based on a continuous release for 1 year.

(e) A listing of the radiation doses from immediate dismantlement tasks is found in Table N.2-13. Radiation doses from entombment tasks that are identical to immediate dismantlement tasks are summed and listed here.

(f) N/A = not applicable.

(g) All entombment and immediate dismantlement tasks are the same for this building. Complete listings of the radiation dose per task are found in Tables N.2-14 and N.2-15.

for preparations for passive safe storage are assumed to apply for entombment. These accidents are compared to immediate dismantlement accidents in Table N.3-6. The radiation doses to the maximum-exposed individual are listed in Table N.2-19.

N.5 RADIATION DOSES FROM ROUTINE TRANSPORTATION TASKS

Calculated radiation doses to the public and to the exposed transport workers resulting from shipments of radioactive materials during decommissioning are discussed in the following subsections. Spent fuel from the final BWR core loading is assumed to be shipped by rail in an IF-300 shipping cask to a repository located 2400 km away. Radioactive waste materials collected during decommissioning tasks are assumed to be shipped by truck to a disposal site located 800 km away. The method used to estimate routine radiation doses to transportation workers and to members of the general public from these shipments is based on the method given in Reference 36. Radiation doses received by workers unloading the radioactive materials at the repository or disposal site are not considered in this study since they are assumed to occur at separate licensed facilities.

N.5.1 Technical Approach

The following assumptions are made about rail and truck shipments of radioactive materials:

- 1: Each shipment contains enough radioactive material (either spent fuel or waste) to result in the maximum exposure rates allowable by regulations. Department of Transportation (DOT) regulations set the following exposure limits:
 - 1000 mR/hr at 1 m from the external surface of any package transported in a closed vehicle
 - 200 mR/hr at the external surface of the vehicle
 - 10 mR/hr at any point 2 m from the vehicle
 - 2 mR/hr at any normally occupied position in the vehicle. (37)

2. For spent-fuel shipments, two train brakemen spend 10 minutes during each of 15 stops (one every 160 km) at an average distance of 1 m from the shipping cask. The average exposure rate at 1 m is assumed to be 25 mR/hr.
3. For each truck shipment of radioactive waste, two truck drivers spend 12 hours inside the cab (with an exposure rate of 2 mR/hr) and 1 hour outside of the cab at a distance of 2 m from the cargo (with an exposure rate of 10 mR/hr).
4. For each truck shipment of radioactive waste, two garagement each spend 10 minutes at an average distance of 2 m from the payload (at an exposure rate of 10 mR/hr).
5. For each truck shipment of radioactive waste, the maximum-exposed individual is located 30 m from the route. His dose rate is calculated to be 1.2×10^{-7} rem per shipment.
6. On each spent-fuel or radioactive waste shipment, 10 onlookers from the general public each spend 3 minutes at an average distance of 2 m from the payload (at an exposure rate of 10 mR/hr).
7. The population density along the transport corridors is 120 persons/km².
8. All shipments maintain an average speed of 65 km/hr; thus, the cumulative dose to the public is 2.3×10^{-6} man-rem/km.

Using these assumptions, the following subsections contain a discussion of the radiation doses from transportation of radioactive materials during each of the three decommissioning alternatives considered in this study. Both routine and accident conditions are discussed.

N.5.1.1 Rail Shipments of Spent Fuel

Using the aforementioned assumptions, the calculated radiation doses to the railway brakemen and to the general public resulting from the transport of the final reactor core are listed in Table N.5-1. For the 43 cask shipments, 5.4 man-rem to the train brakemen and 0.45 man-rem to onlookers and the general public are calculated to result. These values are assumed to be identical for all decommissioning alternatives considered, since all require the spent fuel to be shipped offsite.

TABLE N.5-1. Calculated Radiation Doses from Rail Transport of Spent Fuel^(a)

<u>Group</u>	<u>Radiation Dose Per Shipment, (man-rem)</u>	<u>Total Radiation Dose, (man-rem)^(b)</u>
<u>Train Brakemen</u>	0.12	<u>5.4</u>
Total Occupational		5.4
Onlookers	0.005	0.22
<u>General Public</u>	0.006	<u>0.24</u>
Total Public Dose		0.46

(a) All doses are rounded to two significant figures.

(b) Based on a total of 43 shipments of 2400 km each.

N.5.1.2 Truck Shipments of Radioactive Waste

The number of radioactive waste shipments required for the three decommissioning alternatives, found in Tables I.3-2, J.5-4, and K.3-2, are: 1495 for immediate dismantlement, 318 for passive safe storage, and 985 for entombment scenario 1 and 728 for entombment scenario 2. Using these numbers of shipments and the previous assumptions, radiation doses to transport workers and to the general public are calculated and are listed in Table N.5-2.

N.5.2 Transportation Accidents

Transportation accidents have a wide range of severities. Most accidents occur at low vehicle speeds and have relatively minor consequences. In general, as speed increases, accident severity also increases. However, accident severity is not a function of vehicle speed only. Other factors, (e.g., the type of accident, the kind of equipment involved, and the location of the accident) can have an important bearing on accident severity.

Furthermore, damage to a package in a transportation accident is not directly related to accident severity. In a series of accidents of the same severity, or in a single accident involving a number of packages, damage to packages may vary from none to extensive. In relatively minor accidents, serious damage to packages

TABLE N.5-2. Calculated Radiation Doses from Routine Radioactive Waste Transportation^(a)

Mode/Group	Radiation Dose per Shipment (man-rem) ^(a)	Number of Shipments	Total Population Dose per Group (man-rem) ^(b)
<u>Immediate Dismantlement</u>			
Truck Drivers	6.7×10^{-2}	1 495	100
Garagemen	3.3×10^{-3}	1 495	<u>5.0</u>
Total Transportation Worker Dose			110
Onlookers	5.0×10^{-3}	1 495	7.5
General Public	1.8×10^{-3}	1 495	<u>2.8</u>
Total Public Dose			10
<u>Preparations for Passive Safe Storage</u>			
Truck Drivers	6.7×10^{-2}	318	21
Garagemen	3.3×10^{-3}	318	<u>1.1</u>
Total Transportation Worker Dose			22
Onlookers	5.0×10^{-3}	318	1.6
General Public	1.8×10^{-3}	318	<u>0.6</u>
Total Public Dose			2.2
<u>Entombment</u>			
(Scenario 1)			
Truck Drivers	6.7×10^{-2}	985	66
Garagemen	3.3×10^{-3}	985	<u>3.2</u>
Total Transportation Worker Dose			69
Onlookers	5.0×10^{-3}	985	4.9
General Public	1.8×10^{-3}	985	<u>1.8</u>
Total Public Dose			6.7
(Scenario 2)			
Truck Drivers	6.7×10^{-2}	728	49
Garagemen	3.3×10^{-3}	728	<u>2.4</u>
Total Transportation Worker Dose			51
Onlookers	5.0×10^{-3}	728	3.6
General Public	1.8×10^{-3}	728	<u>1.3</u>
Total Public Dose			4.9

(a)Based on one-way trips of 800 km.

(b)All doses are rounded to two significant figures.

can occur from impacts on sharp objects or from being struck by other cargo. Conversely, even in very severe accidents, damage to packages may be minimal.

The probabilities of rail and truck accidents in this study are based on accident data supplied in Reference 36. Accidents are classified into five severity categories as functions of vehicle speed and fire duration. The five categories and their associated probabilities for both rail and truck accidents are shown in Table N.5-3.

TABLE N.5-3. Transportation Accident Severity Categories^(a)

Severity	Vehicle Speed (km/hr)	Fire Duration (hr)	Probability/Vehicle-km	
			Rail	Truck
Minor	0-50	0	3×10^{-7}	2×10^{-7}
	0-50	<0.5	4×10^{-9}	4×10^{-9}
	50-80	0	2×10^{-7}	6×10^{-7}
Moderate	0-50	0.5-1.0	6×10^{-10}	3×10^{-11}
	50-80	<0.5	2×10^{-9}	6×10^{-9}
	80-110	0	5×10^{-8}	2×10^{-7}
	80-110	<0.5	6×10^{-10}	3×10^{-9}
Severe	0-50	>1.0	4×10^{-11}	3×10^{-12}
	50-80	>1.0	2×10^{-11}	6×10^{-12}
	50-80	0.5-1.0	3×10^{-10}	4×10^{-12}
	80-110	0.5-1.0	9×10^{-11}	4×10^{-12}
	>110	0	5×10^{-10}	6×10^{-11}
	>110	<0.5	6×10^{-12}	6×10^{-11}

(a) Summarized from material in Reference 36.

The following subsections contain a discussion of the method of calculating the airborne radioactivity concentrations resulting from transportation accidents, and discussions of spent-fuel and radioactive waste transportation accidents. Radiation doses to the maximum-exposed individual from inhalation are presented.

N.5.2.1 Calculations of the Airborne Radioactive Concentration from a Transportation Accident

The estimation of air concentrations for short-term airborne releases at distances within 1.0 km of the source requires special attention. Reference 38

contains a discussion of acceptable assumptions for evaluating design-basis, loss-of-coolant accidents at a PWR. For accidental releases with a duration of less than 8 hours, a Pasquill Type F atmospheric stability, and a unidirectional 1-m/sec wind are assumed. In Figure 2(A) of Reference 38, the diffusion factor (χ/Q) is shown as a function of both the distance from a point source and the duration of the release. For a release of less than 8 hours, the χ/Q value at a distance of about 200 m is about 10^{-2} sec/m³. Extension of the curve beyond the border of the figure results in an estimated χ/Q value of about 1×10^{-1} sec/m³ at a distance of 100 m.⁽³⁸⁾ Because of the difficulty in modeling the air concentrations at close distances, for this study the maximum-exposed individual is assumed to be located at 100 m from transportation accidents.

At distances close to the release source, the diffusion of the downwind plume is influenced by the cross-sectional area of the release source. For this reason, a comparison of the extrapolated χ/Q values is calculated using a building-wake model.⁽³⁹⁾ Assuming complete reflection of the plume by the ground plane, the ground-level centerline air concentration from a ground-level release is calculated using Equation N.13:

$$\chi = Q' (\pi \Sigma_y \Sigma_z \bar{u})^{-1} \quad (N.13)$$

where:

- χ • the ground-level centerline air concentration, Ci/m³
- Q' • the rate of release from the source, corrected for decay during transit to the exposure point, Ci/sec
- $\bar{u}$ • the average ground-level wind speed in the direction of travel, m/s
- Σ_y, Σ_z • the building-wake diffusion coefficients, in meters, calculated by:

$$\Sigma_y = (cA/\pi + \sigma_y^2)^{1/2}$$

$$\Sigma_z = (cA/\pi + \sigma_z^2)^{1/2}$$

where:

- c • a factor estimating the relation of the cross-sectional area to pressure wakes (0.5 for this study)
- σ_y • the crosswind lateral standard deviation of the cloud concentration, m
- σ_z • the crosswind vertical standard deviation of the cloud concentration, m

Minimum values of σ_y and σ_z of $\sqrt{A}/6$ are assumed for this study.

Using Equation N.13, Pasquill Type F centerline air concentrations for various distances and selected release source cross-sectional areas are shown in Figure N.5-1. At about 500 m downwind, the curves converge. At close distances to the release source, the dependence of the building-wake modification is illustrated by the divergence of the area-dependent curves.

In Figure N.5-1, it is seen that the value of χ/Q taken from Reference 38 corresponds to an exposed cross-sectional area of about 30 m². This area appears to be reasonable; therefore, a value of 3 x 10⁻² sec/m³ is used as the transportation accident χ/Q located 100 m downwind from the accident.

N.5.2.2 Accidents During Rail Transportation of Spent-Fuel

The IF-300 cask is a water-filled railroad cask designed to transport 18 fuel assemblies with an approximate loaded weight of 63 Mg.^(40,41) Gamma shielding is provided by 0.1 m of depleted uranium sandwiched between stainless steel walls. Neutron shielding and heat removal are provided by the water in the cask. This cask is a rugged thick-walled container designed to provide safe transport of spent fuel, with design integrity to withstand most transport accident situations. The cask is licensed to withstand Type B package tests. Recent results of cask test programs show that this cask will withstand all

FIGURE N.5-1. Pasquill Type F Centerline χ/Q Values for Building-Wake Modification Factors Versus Downwind Distance

but the most severe, highly unusual types of accidents. (42,43,44) For a release to occur during spent fuel transport, radioactive material must leave both the fuel cladding and the cask containment.

Postulated accidents involving spent fuel are derived from those presented in Reference 45. Since the transportation of spent fuel is not unique to decommissioning, and since the probabilities of accidents that lead to atmospheric releases of radionuclides from spent fuel transportation accidents are so low,

no further analysis or dose calculations are presented in this study. A more complete discussion of the impact of spent fuel transportation accidents on public safety is given in Reference 45.

N.5.2.3 Accidents During Truck Transportation of Radioactive Wastes

Transportation accidents involving trucks carrying radioactive waste from decommissioning the reference BWR may result in the atmospheric release of radioactive material. Accidents could occur with shipments of either Type B or A packages as discussed in the following paragraphs.

Type B Package Accidents. A number of accident classes are identified and discussed in Reference 45 for Type B packages. The accident protection provided by Type B overpacks and containers enables a shipment to withstand all but very severe, highly unusual accidents. Only improper packaging, severe impact, or fire is assumed to provide a mechanism for atmospheric release. The material being transported is solid and, for the most part, noncombustible; hence, even the fraction of respirable material released in a severe accident with ruptured Type B containers is expected to be small.⁽⁴⁵⁾ The most highly radioactive wastes are the neutron-activated portion of the reactor vessel and its internals, which are solid and noncombustible, and an airborne release from an accident involving this material is considered to have an extremely low probability. Therefore, no accident analysis of radioactive waste shipped in Type B containers is attempted in this study. A detailed discussion of the public safety impacts of transportation accidents involving Type B containers is found in Reference 46.

Type A Package Accidents. A more realistic transportation accident is one involving wastes shipped in Type A packages. In Appendix I, shipments of compacted combustible material in Type A packages are described. Accidents descriptions involving Type A packages are given in Table N.5-4 for minor accidents and in Table N.5-5 for severe accidents. Both minor and severe accidents are assumed to occur to truck shipments of combustible waste containing the mixture of radionuclides in reference radionuclide inventory 5 (Table E.2-9 in Appendix E). This inventory is composed of radionuclides in transport groups III and IV.⁽⁴⁷⁾ The dominant radionuclide in this inventory is ^{60}Co , a transport-group-III radionuclide.⁽¹²⁾ Type A packages containing

TABLE N.5-4. Minor Accidents for Type A Packages
During Truck Transportation

<u>Accident Description</u>	<u>Sequence of Events</u>	<u>Atmospheric Release</u>
Truck Collision or Overturn Involves Waste Containers	<ol style="list-style-type: none"> 1. Collision or overturn accident occurs. 2. Truck leaves roadway and may overturn. 3. Type A packages may rupture, but no fire occurs. 4. Accident is reported to local and federal officials. 5. Packages recovered. 	None
Truck Collision or Overturn and a Minor Fire (1/2 Hour or Less) Involves 1 Type A Waste Container.	<ol style="list-style-type: none"> 1. Collision or overturn accident occurs. 2. Truck leaves roadway and may overturn. 3. One Type A waste container is involved in a fire with 1/2 hour or less duration. 4. Accident is reported to local and federal officials. 5. Packages recovered. 	Release fraction of 5×10^{-4} of the contents of one waste package is assumed. (27)

TABLE N.5-5. Severe Accidents for Type A Packages
During Truck Transportation

<u>Accident Description</u>	<u>Sequence of Events</u>	<u>Atmospheric Release</u>
Truck Collision or Overturn and a Major Fire (1 Hour or Longer) Involving 40 Type A Waste Containers.	<ol style="list-style-type: none"> 1. Collision or overturn accident occurs. 2. Truck leaves roadway at high speed and overturns. 3. Type A packages rupture and 40 waste containers are involved in a fire. 4. Accident is reported to local and federal officials. 5. Packages recovered. 	Release fraction of 5×10^{-4} of the contents of 40 waste packages is assumed. (27)

transport group III material are limited to a maximum of 3 Ci per package.⁽⁴⁸⁾ Since most operations that involve combustible waste are manual operations conducted in relatively low dose rate areas, the quantity of radioactivity contained in each package is assumed to be 1 Ci for this study. For the minor accident, one package is assumed to burn and release 5×10^{-4} of the radioactivity in the package, or 5×10^{-4} Ci.⁽²⁷⁾ The resulting 50-year committed dose equivalent to the maximum-exposed individual located 100 m downwind is 2.2 mrem to the lung. The minor transportation accident is estimated to have a low frequency of occurrence. For the severe accident, the contents of 40 waste containers (each containing 1 Ci of reference radionuclide inventory 5) are assumed to burn, releasing a total of 2×10^{-2} Ci. The resulting 50-year committed dose equivalent to the maximum-exposed individual is 90 mrem to the lung. The severe accident is estimated to have a low frequency of occurrence. A complete listing of the doses to the organs of the maximum-exposed individual from these accidents is given Table N.5-6.

TABLE N.5-6. Radiation Doses to the Maximum-Exposed Individual from Accidental Atmospheric Releases During Transportation Accidents

Incident (a)	Reference Radionuclide Inventory Number (b)	Total Atmospheric Release (Ci/hr) (c)	First-Year Dose (rem)					Fifty-Year Committed Dose Equivalent (rem)				
			Total Body	Bone	Lung	Thyroid	GI-LLI	Total Body	Bone	Lung	Thyroid	GI-LLI
Minor Transportation Accident	5	5×10^{-4}	5.4×10^{-5}	8.7×10^{-5}	2.2×10^{-3}	9.0×10^{-5}	3.5×10^{-5}	2.7×10^{-4}	9.5×10^{-4}	2.2×10^{-3}	9.0×10^{-5}	3.5×10^{-5}
Severe Transportation Accident	5	2×10^{-2}	2.2×10^{-3}	3.5×10^{-3}	9.0×10^{-2}	3.6×10^{-3}	1.4×10^{-3}	1.1×10^{-2}	3.8×10^{-2}	9.0×10^{-2}	3.6×10^{-3}	1.4×10^{-3}

(a) Both transportation accidents are felt to have a "low" frequency of occurrence.

(b) These numbers refer to the reference radionuclide inventories discussed in Appendix E.

(c) All releases are assumed to occur in a 1-hr period for comparison purposes.

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APPENDIX 0

DETAILS OF ALTERNATE STUDY BASES

The results presented previously in this report are based on the characteristics of the reference BWR, on the conditions postulated to exist at the end of its operating life, and on the assumption that the operating utility becomes the prime contractor for decommissioning. This appendix contains detailed information concerning the sensitivity of decommissioning costs and/or radiation doses to the following "alternate bases":

- different BWR power plant sizes
- increased radiation exposures
- different contractual arrangements
- increased nuclear waste disposal charges
- different BWR containment designs.

These alternate bases are discussed in turn in the following sections.

0.1 SENSITIVITY OF DECOMMISSIONING COSTS AND RADIATION DOSES TO DIFFERENT BWR POWER PLANT SIZES

In this section, a scaling factor methodology is used to determine the costs and the radiation doses for decommissioning different size BWR power plants. In the first two of the following subsections, a scaling factor relationship is first developed and then applied to determine the following aspects of decommissioning any BWR regardless of its size:

- costs for immediate dismantlement
- costs for passive safe storage
- costs for entombment
- radiation doses for these three decommissioning modes.

In the third subsection, a relationship is developed and applied to determine the cost for demolition and site restoration of any BWR.

0.1.1 Development of the Scaling Factor Relationship

This subsection presents the details for and the development of a scaling factor relationship that will be applied later in Section 0.1.2 to determine the decommissioning costs and radiation doses for any BWR power plant. The following matters are presented: the BWR power plants, the BWR plant components, and the scaling factor methodology used in developing, and the actual development of the scaling factor relationship.

0.1.1.1 BWR Power Plants Used

Seven BWR power plants, including the reference BWR and ranging in power rating from 1593 MWt to 3320 MWt, are used in developing the scaling factor relationship. The BWR power plants and their power ratings are: Vermont Yankee, 1593 MWt; Oyster Creek; 1600 MWt; Monticello, 1670 MWt; Cooper, 2381 MWt; Dresden 2 or 3, 2527 MWt; Peach Bottom 2 or 3, 3293 MWt; and the reference plant, WNP-2, 3320 MWt.

The scaling factor relationship is based on information obtained from the final safety analysis reports (FSAR) for the seven plants.⁽¹⁻⁷⁾ It is developed by comparing the dimensional or other physical characteristics of components of the other BWR power plants with the corresponding features of the reference BWR.

It should be noted that the BWR plants with power ratings less than 1500 MWt are early designs and are generally markedly different from the later, larger plants. Since they are generally one-of-a-kind plants, these small, early BWR plants are not included in this analysis.

It should also be noted that there are some significant differences among the later, large plants, specifically with regard to reactor containment design (i.e., the design of the drywell and the pressure suppression chamber). The three different BWR containment designs include the Mark I (light bulb-shaped drywell and toroidal-shaped pressure suppression chamber), the Mark II (a truncated-cone drywell atop a cylindrical suppression chamber), and the Mark III (a cylindrical drywell with an annular, weir-type suppression chamber). The effect of these three different containment designs on dismantlement costs is discussed later in Section 0.5.

0.1.1.2 BWR Plant Components Used

Three major BWR plant components are used in developing the scaling factor relationship: 1) the reactor vessel, 2) the reactor vessel internals, and 3) the Reactor Building (combination of selected contents excluding the first two components). These components are described in most of the individual FSARs in sufficient detail to permit meaningful comparisons to be made with the corresponding components of the reference BWR. The components that are not described in the FSARs sufficiently well to permit comparisons with the reference plant and with each other are not examined in this analysis. It is assumed that the dismantlement costs for the unanalyzed components have the same relationship to the plant power rating as have the components that are examined.

0.1.1.3 Scaling Factor Methodology Used

The methodology used to develop the scaling factor relationship is based on comparisons of the individual BWR plant components and on the estimated costs presented in Section I.3 of Appendix I for immediate dismantlement of each of the reference BWR components. The component comparisons are made on the physical parameter (i.e., mass, surface area, volume, or some combination thereof) that is most appropriate for the particular component.

The component comparisons for a given BWR power plant determine component-specific scaling factors for that plant, which are weighted by cost factors based on the dismantlement costs for the respective reference-plant components and summed to create a plant-specific scaling factor. The various plant-specific scaling factors are used to develop the scaling factor relationship.

The component-specific scaling factor, SF_{Cj} , is calculated by dividing the physical parameter of the specific component, P_{Cj} , from the j th plant by that from the reference plant, as follows:

$$SF_{Cj} = P_{Cj} / P_{Cr} \quad (0.1)$$

Component-specific scaling factors are calculated in this manner for all plants whose FSARs provide the appropriate component parameters. For those plants whose FSARs do not provide the appropriate component parameters, these calculated

component-specific scaling factors are fitted (by the method of least squares) to a linear function that has as its independent variable the plant power rating (PPR) in thermal megawatts [i.e., $SF_{Cj} = A_C + B_C (PPR_j)$]. This function is used to estimate component-specific scaling factors for those plants.

The cost factor, CF_i , used to weight the component-specific scaling factor for the i th component of each plant is determined by dividing each individual cost for dismantling the reference-plant i th component, C_{ir} , by the sum of the individual costs, as follows:

$$CF_{ir} = C_{ir} / \sum_{i=1}^n C_{ir} \quad (0.2)$$

The plant-specific scaling factor for the j th plant, SF_{pj} , is found by summing the product of the component-specific scaling factor and the respective cost factor for component i with those for the other components:

$$SF_{pj} = \sum_{i=1}^n (SF_{ij} CF_{ir}) \quad (0.3)$$

The desired scaling factor relationship is developed by fitting the plant-specific scaling factors for all the plants with a linear function that again has the plant power rating as its independent variable:

$$OSF = A + B (PPR) \quad (0.4)$$

OSF is the "overall scaling factor" for estimating a cost or a radiation dose for any decommissioning mode applied to any BWR power plant. For example, the cost and the radiation dose for immediate dismantlement of the j th power plant, C_{dj} and D_{dj} , can be estimated by multiplying OSF evaluated at PPR_j (OSF_j) by the cost and the radiation dose, respectively, for immediate dismantlement of the reference BWR, as follows:

$$C_{dj} = OSF_j C_{dr} \quad (0.5a)$$

$$D_{dj} = OSF_j D_{dr} \quad (0.5b)$$

The values for passive safe storage and for entombment of the jth power plant can be estimated in the same manner.

OSF is developed in the following subsection.

0.1.1.4 Development of the Scaling Factor Relationship

The development of the scaling factor relationship for decommissioning costs and radiation is broken down into the following parts:

- component-specific scaling factors and reference-plant dismantlement costs for the reactor vessel, the reactor vessel internals, and the Reactor Building, in turn (basically Equation 0.1)
- plant-specific scaling factors (Equations 0.2 and 0.3)
- scaling factor relationship (Equation 0.4).

0.1.1.4.1 Component-Specific Scaling Factors and Reference-Plant Dismantlement Costs for the Reactor Vessel. Dismantlement of the reactor vessel encompasses removing and segmenting the vessel, packaging the vessel segments, and transporting the packaged segments and emplacing them in a nuclear waste disposal facility. It was determined previously that the dismantlement cost for a reactor vessel is essentially proportional to the mass of the vessel.⁽⁸⁾ Therefore, in this study, the component-specific scaling factor for a given reactor vessel dismantlement is the ratio of the mass of the given vessel to the mass of the reference BWR vessel.

The component-specific scaling factors for the reactor vessels of the selected BWR power plants are listed in Table 0.1-1, together with the plant power ratings and the reported reactor vessel masses. Of the five plants whose reactor vessel masses are reported in the table, only four component-specific scaling factors are included in the least-squares fitting analysis to determine the scaling factors for the other two plants. Oyster Creek is excluded

TABLE 0.1-1. Component-Specific Scaling Factors for BWR Reactor Vessels

Power Plant	Plant Power Rating (MWt)	Reactor Vessel Mass (Mg)	Component-Specific Scaling Factor ^(a)
Vermont Yankee	1 593	428	0.572
Oyster Creek	1 600	592 ^(b)	0.791 ^(c)
Monticello	1 670	-- ^(d)	0.588 ^(e)
Cooper	2 381	566	0.757 ^(e)
Dresden 2 or 3	2 527	-- ^(d)	0.782 ^(e)
Peach Bottom 2 or 3	3 293	681	0.910
WNP-2	3 320	748	1.000

(a) Calculated as the ratio of the reactor vessel mass for the jth plant to that for the reference plant, WNP-2 (Equation 0.1).

(b) Calculated from dimensional data obtained from Reference 2.

(c) Value excluded from least-squares analysis.

(d) Not available in FSAR.

(e) Value inferred from least-squares fitted equation.

because its mass is not given in the FSAR and is calculated from dimensional data obtained therefrom. It is also somewhat atypical in that its vessel wall is thicker than the walls of the other plants. The equation of the "best-fit" line through the included component-specific scaling factors is:

$$SF_{Vj} = 0.211 + (2.26 \times 10^{-4}) PPR_j$$

Using this relationship, component-specific scaling factors are calculated for those plants whose reactor vessel mass data are unavailable or not calculable (values footnoted (e) in Table 0.1-1).

The use of a straight-line relationship can be justified on the grounds that all of the reactor vessels are designed to the same code requirements (i.e., the configurations and the design temperatures and pressures are the same for all of the power plants).

The factors considered in estimating the cost for dismantling the reactor vessel in the reference plant include removal, sectioning, packaging, transporting, and disposal at an approved waste storage facility. In Section I.3

of Appendix I, dismantlement of the vessel is treated as a portion of the total dismantlement project, and some of the costs are not isolated (e.g., sectioning of the non-activated portions). A crew of workmen is assumed active throughout the dismantlement operation, and their activities are spread over the entire project as required to maintain a smooth progression of the overall job. As a result, some of the continuing operations that are not critical path items and that do not require special skills or equipment are not defined individually, but are considered in their entirety. The costs of these operations are lumped together as part of the total immediate dismantlement cost. It is assumed that the costs for dismantling the reactor vessel that are included in the total cost are small in relation to those costs that are individually defined.

The costs for dismantling the reference-plant reactor vessel that are defined and treated individually are listed in Tables I.3-3, I.3-4, and I.3-7 in Appendix I and are summarized in Table 0.1-2.

TABLE 0.1-2. Identifiable Costs for Immediate Dismantlement of the WNP-2 Reactor Vessel

<u>Cost Category</u>	<u>Immediate Dismantlement Cost (\$) (a)</u>
Activated Material Disposal	71 530 ^(b)
Contaminated Material Disposal	159 650 ^(c)
<u>Dedicated Staff Labor</u>	<u>385 290^(d)</u>
Total	616 470

(a) Early 1978 dollars.

(b) Based on information presented in Table I.3-3 in Appendix I.

(c) Based on information presented in Table I.3-4.

(d) Based on information presented in Table I.3-7.

0.1.1.4.2 Component-Specific Scaling Factors and Reference-Plant Dismantlement Costs for the Reactor Vessel Internals. Dismantlement of the reactor vessel internals requires that they be removed from the reactor vessel and placed under water in the dryer and separator pool. The internals

are then sectioned and packaged remotely (under water) and shipped to an approved nuclear waste storage facility for disposal. It was determined previously that the cost of dismantling the reactor vessel internals is roughly proportional to their mass.⁽⁸⁾ Therefore, in this study, the component-specific scaling factor for the reactor vessel internals is the ratio of the mass of the given-plant vessel internals mass to the mass of the reference BWR vessel internals.

The component-specific scaling factors for the reactor vessel internals of the selected BWR power plants are listed in Table 0.1-3, together with the plant power ratings and the reported reactor vessel internals masses. Only the four scaling factors based on FSAR information are included in the least-squares fitting analysis to determine the component-specific scaling factors for the three other plants. The equation of the "best-fit" line through the component-specific scaling factors is:

$$SF_{Ij} = 0.283 + (1.772 \times 10^{-4}) PPR_j$$

Using this relationship, component-specific scaling factors are calculated for those plants whose reactor vessel internals mass data are unavailable (values footnoted (c) in Table 0.1-3).

TABLE 0.1-3. Component Specific Scaling Factors for BWR Reactor Vessel Internals

Power Plant	Plant Power Rating (MWt)	Reactor Vessel Internals Mass (Mg)	Component-Specific Scaling Factor(a)
Vermont Yankee	1 593	151.7	0.531
Oyster Creek	1 600	--(b)	0.567(c)
Monticello	1 670	176.4	0.617
Cooper	2 381	--(b)	0.705(c)
Dresden 2 or 3	2 527	--(b)	0.731(c)
Peach Bottom 2 or 3	3 293	209.6	0.734
WNP-2	3 320	285.7	1.000

(a) Calculated as the ratio of the reactor vessel internals mass for the jth plant to that for the reference plant, WNP-2 (Equation 0.1).

(b) Not available in FSAR.

(c) Value inferred from least-squares fitted equation.

The identifiable dismantlement costs for the reactor vessel internals of the reference BWR involve radioactive material disposal and staff labor. These costs are given in Tables I.3-3, I.3-4, and I.3-7 in Appendix I and are summarized in Table 0.1-4.

TABLE 0.1-4. Identifiable Costs for Immediate Dismantlement of the WNP-2 Reactor Vessel Internals

<u>Cost Category</u>	<u>Immediate Dismantlement Cost (\$) (a)</u>
Activated Material Disposal	2 102 680 ^(b)
Contaminated Material Disposal	90 580 ^(c)
<u>Dedicated Staff Labor</u>	<u>1 020 270^(d)</u>
Total	3 213 530

(a) Early 1978 dollars.

(b) Based on information presented in Table I.3-3 in Appendix I.

(c) Based on information presented in Table I.3-4.

(d) Based on information presented in Table I.3-7.

0.1.1.4.3 Component-Specific Scaling Factors and Reference-Plant Dismantlement Costs for the Reactor Building. Besides the reactor vessel and its internals, dismantlement of the Reactor Building involves the removal, packaging, and disposal of other activated and contaminated materials. The building components that produce these materials are:

- sacrificial shield
- pressure suppression chamber
- piping systems
- HVAC and electrical equipment
- reactor refueling pools
- contaminated concrete surfaces
- miscellaneous equipment.

The appropriate physical characteristics of these building components are used to calculate the component-specific scaling factor for the Reactor Building.

Scaling factors for all the individual building components in a given power plant are calculated, weighted, and summed in accordance with Equations 0.1 and 0.3 in Section 0.1.1.3 to determine the component-specific scaling factor for the Reactor Building of the given plant.

The identifiable reference-plant dismantlement costs for each of these building components are summarized and totaled in Table 0.1-5. The cost factors for weighting the scaling factors for the building components are also presented.

TABLE 0.1-5. Identifiable Costs for Immediate Dismantlement of the WNP-2 Reactor Building

<u>Building Component</u>	<u>Immediate Dismantlement Cost (\$) (a)</u>	<u>Cost Factor (b)</u>
Sacrificial Shield	616 070	0.1454
Pressure-Suppression Chamber	672 780	0.1588
Piping Systems	1 800 860	0.4250
HVAC and Electrical Equipment	269 040	0.0635
Reactor Refueling Pools	349 970	0.0826
Contaminated Concrete from Surfaces	119 660	0.0282
Miscellaneous Equipment	408 740	0.0965
Totals	4 237 120	1.0000

(a) Early 1978 dollars.

(b) Calculated as the ratio of the component cost and the total of the seven component costs (Equation 0.2).

The component-specific scaling factors for the BWR Reactor Building are presented in Table 0.1-6, together with a summary of the scaling factors and cost factors for the building components, which are used to determine the component-specific scaling factors.

The scaling factors and the identifiable reference-plant dismantlement costs for each building component are discussed and presented in the following subsections.

TABLE 0.1-6. Component-Specific Scaling Factors for BWR Reactor Buildings

Power Plant	Plant Power Rating (Mwt)	Scaling Factors for Building Components							Component-Specific Scaling Factor(a)
		Sacrificial Shield	Pressure Suppression Chamber	Piping Systems	HVAC and Electrical	Reactor Refueling Pools	Contaminated Concrete Surfaces	Miscellaneous Equipment	
Vermont Yankee	1 593	0 693	0 531	0 480	0 934	0 964	0 941	0 586	0 611
Oyster Creek	1 600	0 725	0 607	0 482	0 785	0 938	0 865	0 632	0 620
Monticello	1 670	0 691	0 568	0 503	0 841	0 846	0 876	0 612	0 612
Cooper	2 381	0 806	0 638	0 720	0 799	0 974	0 802	0 724	0 748
Dresden 2 or 3	2 527	0 824	0 817	0 760	0 875	1 151	0 880	0 808	0 826
Peach Bottom 2 or 3	3 293	0 990	0 990	0 990	0 876	1 167	0 878	0 995	0 995
WNP-2	3 320	1 000	1 000	1 000	1 000	1 000	1 000	1 000	1 000
Cost Factor for Reference-Plant Building Component(b)		0 1454	0 1588	0 4250	0 0635	0 0826	0 0282	0 0965	

(a) Calculated as the sum of the products of the individual scaling factors and the corresponding cost factors (Equation 0 3 applicable)

(b) Same as the fractional cost presented in Table 0 1-5

Sacrificial Shield. Dismantlement of the sacrificial shield involves removal of the shield itself as well as the surrounding radial beams and floor gratings.

The sacrificial shields in BWRs are of approximately the same thickness and relative height. Therefore, the volume of the shield is proportional to the outer surface area of the reactor vessel it surrounds. Since the dismantlement cost for the shield is proportional to the volume of the shield, the scaling factor for the sacrificial shield is the product of the reactor vessel length, L, and diameter, D, for the jth plant divided by the same product for the reference plant, as developed in Table 0.1-7.

TABLE 0.1-7. Scaling Factor for BWR Sacrificial Shields

Power Plant	Plant Power Rating (MWt)	Reactor Inside Dimensions		Scaling Factor ^(a)
		Vessel Length (m)	Diameter (m)	
Vermont Yankee	1 593	19.24	5.23	0.692
Oyster Creek	1 600	19.46	5.41	0.724
Monticello	1 670	19.25	5.21	0.690
Cooper	2 381	21.12	5.54	0.805
Dresden 2 or 3	2 527	--(b)	--(b)	0.854 ^(c)
Peach Bottom 2 or 3	3 293	22.53	6.38	0.989
WNP-2	3 320	22.78	6.38	1.000

(a) Calculated as the ratio of the product of the reactor vessel length and diameter for the jth plant and the same product for the reference plant, WNP-2 (Equation 0.1 applicable).

(b) Not available in FSAR.

(c) Value inferred from least-squares fitted equation.

The physical geometry of the radial beams and floor gratings surrounding the sacrificial shield suggests that the dismantlement cost for these items will vary in the same manner as those for the sacrificial shield. Therefore, the scaling factor for the sacrificial shield is assumed to apply to the radial beams and floor gratings as well.

The equation of the "best-fit" line through the scaling factors, which is used to infer one of the scaling factors, is given below:

$$SF_{Sj} = 0.418 + (1.726 \times 10^{-5}) PPR_j$$

The costs estimated for dismantlement of the sacrificial shield (and the radial support beams and floor gratings) from the reference BWR are listed in Table 0.1-8. These costs are summarized from Tables I.3-3, I.3-4, and I.3-7 in Appendix I.

TABLE 0.1-8. Identifiable Costs for Dismantlement of the WNP-2 Sacrificial Shield^(a)

<u>Cost Category</u>	<u>Immediate Dismantlement Cost (\$)^(b)</u>
Activated Material Disposal	126 010 ^(c)
Contaminated Material Disposal	272 270 ^(d)
<u>Staff Labor</u>	<u>217 790^(d)</u>
Total	616 070

(a) Including the surrounding radial beams and floor gratings.

(b) Early 1978 dollars.

(c) Based on information presented in Table I.3-3 in Appendix I.

(d) Based on information presented in Table I.3-4.

(e) Based on information presented in Table I.3-7.

Pressure Suppression Chamber. In selecting an appropriate parameter to develop the scaling factor for the pressure suppression chamber, it is necessary to determine which parameters most strongly influence the dismantlement cost. The individual cost components are listed in Table 0.1-9, extracted from Tables I.3-4 and I.3-7 in Appendix I. From the data in Table 0.1-9, it is seen that draining and cleaning the pool (including disposing of the contaminated water) constitutes the major share of the cost for dismantlement of the pressure suppression chamber. Since this item is proportional to the volume of the suppression pool, the scaling factor for this pressure suppression chamber is the pool volume of the jth plant divided by the pool volume of the reference plant, as shown in Table 0.1-10.

TABLE 0.1-9. Identifiable Costs for Immediate Dismantlement of the WNP-2 Pressure Suppression Chamber

<u>Cost Category</u>	<u>Immediate Dismantlement Cost (\$) (a)</u>
Disposal of Piping and Equipment (0.35 x \$315 960)	110 590 ^(b)
Disposal of Contaminated Concrete (0.805 x \$105 930)	85 270 ^(b)
Drain and Clean Pool	241 790 ^(c)
Enlarge Wetwell Access	6 570 ^(c)
Remove Piping and Equipment (35% is downcomer piping) (0.35 x \$533 920)	186 870 ^(c)
Remove Contaminated Concrete (80.5% is from wetwell) (0.805 x \$51 790)	41 690 ^(c)
<u>Total</u>	<u>672 780</u>

(a) Early 1978 dollars.

(b) Based on information presented in Table I.3-4 in Appendix I.

(c) Based on information presented in Table I.3-7.

TABLE 0.1-10. Scaling Factors for BWR Pressure Suppression Chambers

<u>Power Plant</u>	<u>Plant Power Rating (MWt)</u>	<u>Pressure Suppression Pool Volume (m³)</u>	<u>Scaling Factors (a)</u>
Vermont Yankee	1 593	2 067	0.531
Oyster Creek	1 600	2 362	0.607
Monticello	1 670	2 208	0.568
Cooper	2 381	2 482	0.638
Dresden 2 or 3	2 527	3 177	0.817
Peach Bottom 2 or 3	3 293	3 851	0.990
WNP-2	3 320	3 890	1.000

(a) Calculated as the ratio of the pool volume for the jth plant and the pool volume for the reference plant, WNP-2 (Equation 0.1).

Piping Systems. Dismantlement costs for the piping systems are proportional to the mass of piping removed, packaged, and emplaced at a nuclear waste disposal facility. The mass of the piping is proportional to the volume of the piping walls, which, in turn, is proportional to the product of the wall thickness, t , the pipe diameter, D , and the pipe length, L . Since the wall thickness is proportional to the pipe diameter for the same design conditions of pressure and stress, the mass of the piping is proportional to the product $D^2 \times L$, which is also proportional to the volume of the piping. The lengths are determined largely by the size of the Reactor Building, which, for the plants used, are very nearly the same. Since the piping is used primarily for the transfer of heat from the reactor under various operating conditions, the volume of the piping is generally proportional to the power rating of the plant. Thus, the scaling factor for the piping is assumed to be the ratio of the power rating of the j th plant and the power rating of the WNP-2 plant, as listed in Table 0.1-11. This scaling factor obviously reduces to zero at zero power rating, an unrealistic result. However, over the range of power ratings examined, this assumption is reasonable and adequate.

TABLE 0.1-11. Scaling Factors for BWR Reactor Building Piping Systems

Power Plant	Plant Power Rating (MWt)	Scaling Factor ^(a)
Vermont Yankee	1 593	0.480
Oyster Creek	1 600	0.482
Monticello	1 670	0.503
Cooper	2 381	0.720
Dresden 2 or 3	2 527	0.760
Peach Bottom 2 or 3	3 293	0.990
WNP-2	3 320	1.000

(a) Calculated as the ratio of the plant power rating for the j th plant and that of the reference plant, WNP-2 (Equation 0.1).

The costs for dismantlement of the piping systems in the Reactor Building of the reference BWR are listed in Table 0.1-12. These costs are summarized from Tables I.3-4 and I.3-7 in Appendix I.

TABLE 0.1-12. Identifiable Costs for Dismantlement of the WNP-2 Reactor Building Piping Systems

<u>Cost Category</u>	<u>Immediate Dismantlement Cost (\$) (a)</u>
<u>Primary Containment Disposal (b)</u>	
Mirror Insulation	17 990
Pipe Hangers and Restraints	74 070
Piping (65% of Total)	205 370
<u>Reactor Building Disposal (b)</u>	
Piping and Valves	176 000
<u>Staff Labor (c)</u>	
Containment Piping and Equipment	533 920
<u>Reactor Building Piping</u>	<u>793 510</u>
Total	1 800 860

(a) Early 1978 dollars.

(b) Based on information presented in Table I.3-4 in Appendix I.

(c) Based on information presented in Table I.3-7.

HVAC and Electrical Equipment. The HVAC system in the Reactor Building consists of ducting, blowers, heaters, coolers, and filters. The bulk of the material and most of the labor for dismantlement are associated with the ducting. For purposes of estimation, the length of ducting is assumed to be proportional to the surface area of the building. It was shown previously that the surface area of a building is proportional to the volume of the building raised to the two-thirds power, $(V_B)^{2/3}$.⁽⁸⁾ Thus, the scaling factor for the HVAC system is the total Reactor Building volume of the jth plant divided by the same volume of the reference plant, that ratio raised to the two-thirds power, $(V_{Bj}/V_{Br})^{2/3}$.

Similarly, the amount of electrical equipment is assumed to be proportional to the surface area of the building. Hence, the scaling factor for the

contaminated electrical equipment is also $(V_{Bj}/V_{Br})^{2/3}$. However, it is assumed that only the electrical equipment in the lower section of the Reactor Building is contaminated, while the HVAC exhaust ducting throughout the entire building is contaminated. Thus, the scaling factor is a weighted average of the volume ratios of the total building and of the lower section of the building. The weighting is determined from the fractions of the dismantlement costs attributable to the HVAC and the electrical systems.

The costs estimated for the dismantlement of the HVAC and electrical systems in the reference-plant Reactor Building are given in Table 0.1-13 (obtained from Tables I.3-4 and I.3-7 in Appendix I). These costs that are uniquely identifiable with either HVAC or electrical are divided 65% and 35%, respectively. Thus, the weightings for calculating the weighted scaling factor are 65% HVAC and 35% electrical.

TABLE 0.1-13. Identifiable Costs for Dismantlement of the WNP-2 Reactor Building HVAC and Electrical Systems

<u>Cost Category</u>	<u>Immediate Dismantlement Cost (\$) (a)</u>	<u>Fraction of Uniquely Identifiable Cost (b)</u>
HVAC Disposal	175 710 ^(c)	0.653
Electrical Disposal	93 330 ^(c)	0.347
<u>Dedicated Staff Labor</u>	<u>44 160^(d)</u>	
Total Cost	313 200	

(a) Early 1978 dollars.

(b) Total of uniquely identifiable costs is \$269,040.

(c) Based on information presented in Table I.3-4 in Appendix I.

(d) Based on information presented in Table I.3-7.

The scaling factors for the HVAC and electrical systems are presented in Table 0.1-14. The equation of the "best-fit" line through the scaling factors (with which one scaling factor is determined) is:

$$SF_{HEj} = 0.761 + (4.795 \times 10^{-5}) PPR_j$$

**TABLE 0.1-14. Scaling Factors for BWR Reactor Building
HVAC and Electrical Systems**

Power Plant	Plant Power Rating (MWt)	Building Volumes (m ³)		HVAC Scaling Factor(a)	Electrical Scaling Factor(a)	Scaling Factor(b)
		Total	Lower			
Vermont Yankee	1 593	114 682	86 649	0.930	0.941	0.934
Oyster Creek	1 600	101 750	76 483	0.796	0.865	0.820
Monticello	1 670	--(c)	--(c)	--	--	0.841 ^(d)
Cooper	2 381	91 250	68 280	0.798	0.802	0.799
Dresden 2 or 3	2 527	104 353	78 383	0.873	0.880	0.875
Peach Bottom 2 or 3	3 293	104 679	78 175	0.875	0.878	0.876
WNP-2	3 320	127 899	94 994	1.000	1.000	1.000
Cost Factor ^(e)				0.653	0.347	

(a) Calculated respectively as the ratio of the lower- or total-building volume from the jth plant and the corresponding values from the reference plant, WNP-2, the ratios raised to the two-thirds power (Equation 0.1 applicable).

(b) Calculated as the sum of the products of the HVAC and electrical scaling factors with their respective cost factors (Equation 0.3 applicable).

(c) Not available in FSAR.

(d) Value inferred from the least-squares fitted equation.

(e) Fractional costs from Table 0.1-13.

Reactor Refueling Pools. Dismantlement of the reactor refueling pools (i.e., the spent fuel storage pool, the reactor well, and the dryer and separator storage pool) involves the draining and cleaning of the pools and the fuel pool cooling and cleanup system, and the removal and disposal of the spent fuel storage racks, the stainless steel pool liners, and about 50 mm of assumed contaminated concrete from behind the pool liners. The volume of water and the spent fuel storage racks (to a degree) are proportional to the volumetric capacities of the pools, while the pool liners and the contaminated concrete are proportional to the surface areas of the pool cavities. Therefore, it is necessary to determine the scaling factors for the reactor refueling pools as the sum of the weighted volumetric scaling factor and the weighted areal scaling factor.

The identifiable costs for immediate dismantlement of the reference-plant reactor refueling pools are summarized and categorized in Table 0.1-15, as obtained from Tables I.3-4 and I.3-7 in Appendix I. The table also presents the fractional costs (cost factors) for weighting the volumetric scaling factor and the areal scaling factor. It can be seen that about 63% of the identifiable costs are volume-related and 37% are area-related. Thus, the scaling factor for the reactor refueling pools is the sum of 63% of the volumetric scaling factor and 37% of the areal scaling factor.

TABLE 0.1-15. Identifiable Costs for Immediate Dismantlement
of the WNP-2 Reactor Refueling Pools

<u>Cost Category</u>	<u>Immediate Dismantlement Cost (\$) (a)</u>	<u>Cost Factor</u>
<u>Volume-Related Costs</u>		
<u>Material Disposal (b)</u>		
Fuel Pool Cooling and Cleanup System	4 150	0.012
Spent Fuel Storage Racks	44 790	0.128
<u>Dedicated Staff Labor (c)</u>		
Drain and Clean Pools	132 980	0.380
Remove and Package Spent Fuel Racks	17 660	0.050
Decontaminate Fuel Pool Cooling and Cleanup System	20 630	0.059
<u>Subtotals, Volume-Related Costs</u>	<u>220 210</u>	<u>0.629</u>
<u>Area-Related Costs</u>		
<u>Material Disposal (b)</u>		
Pool Liners	44 790	0.128
Contaminated Concrete	51 770 (d)	0.148
<u>Dedicated Staff Labor (c)</u>		
Remove Pool Liners	17 660	0.050
Remove Contaminated Concrete	15 540 (d)	0.044
<u>Subtotals, Area-Related Costs</u>	<u>129 760</u>	<u>0.370</u>
<u>Totals</u>	<u>349 970</u>	<u>1.000</u>

(a) Early 1978 dollars.

(b) Based on information presented in Table I.3-4 in Appendix I.

(c) Based on information presented in Table I.3-7.

(d) 36% of values presented in the respective tables in Appendix I.

The scaling factors for the reactor refueling pools are presented in Table 0.1-16, together with the related information.

TABLE 0.1-16. Scaling Factors for the Reactor Refueling Pools

Power Plant	Plant Power Rating (Mwt)	Reactor Refueling Pools		Volume-Related Scaling Factor(a)	Area-Related Scaling Factor(a)	Scaling Factor(b)
		Volume (m ³)	Surface Area (m ²)			
Vermont Yankee	1 593	1 178	584	0.889	1.086	0.964
Oyster Creek	1 600	1 133	573	0.855	1.065	0.938
Monticello	1 670	994	545	0.750	1.013	0.846
Cooper	2 381	1 197	594	0.903	1.104	0.974
Dresden 2 or 3	2 527	1 474	657	1.113	1.221	1.151
Peach Bottom 2 or 3	3 293	1 502	663	1.134	1.232	1.167
WNP-2	3 320	1 325	538	1.000	1.000	1.000
Cost Factor(c)				0.629	0.370	

(a) Calculated, respectively, as the ratio of the reactor refueling pools volume or surface area from the jth plant and the corresponding values from the reference plant, WNP-2 (Equation 0.1).

(b) Calculated as the sum of the products of the volume-related and area-related scaling factors with their respective cost factors (Equation 0.3 applicable).

(c) Fractional cost subtotals from Table 0.1-15.

Contaminated Concrete Surfaces. Those concrete surfaces (excluding surfaces in the reactor refueling pools and the pressure suppression chamber) into which radioactive contamination has penetrated are removed, packaged, and emplaced at a nuclear waste disposal facility. Since about 50 mm of surface is assumed removed, the amount of material is proportional to the surface areas involved. The surface areas involved are assumed to be the lower section of the Reactor Building. It was shown previously that the surface area of a building is proportional to its volume raised to the two-thirds power.⁽⁸⁾ Therefore, the scaling factor for the contaminated concrete from the jth plant is the lower section volume of that plant divided by the lower section volume of the reference plant, the ratio raised to the two-thirds power, $(V_{Lj}/V_{Lr})^{2/3}$, as presented in Table 0.1-17, together with plant power ratings and the volumes of the building lower sections.

TABLE 0.1-17. Scaling Factors for BWR Reactor Building Contaminated Concrete Surfaces

<u>Power Plant</u>	<u>Plant Power Rating (Mwt)</u>	<u>Volume of Building Lower Section (m³)</u>	<u>Scaling Factor^(a)</u>
Vermont Yankee	1 593	86 649	0.941
Oyster Creek	1 600	76 483	0.865 ^(c)
Monticello	1 670	-- ^(b)	0.876 ^(c)
Cooper	2 381	68 280	0.802
Dresden 2 or 3	2 527	78 383	0.880
Peach Bottom 2 or 3	3 293	78 175	0.878
WNP-2	3 320	94 996	1.000

(a) Calculated as the ratio of building lower section volume from the jth plant and that value from the reference plant, WNP-2, the ratio raised to the two-thirds power (Equation 0.1 applicable).

(b) Not available in the FSAR.

(c) Value inferred from least-squares analysis.

The equation of the "best-fit" line through the scaling factors is:

$$SF_{Cj} = 0.837 + (2.334 \times 10^{-5}) PPR_j$$

The scaling factor for the Monticello plant is calculated using the above equation.

The identifiable costs estimated for immediate dismantlement of the contaminated concrete surfaces in the WNP-2 plant are again obtained from Tables I.3-4 and I.3-7 in Appendix I. They are summarized in Table 0.1-18.

Miscellaneous Equipment. The majority of the identifiable items of equipment removed during immediate dismantlement of the Reactor Building are closely related to the reactor pressure vessel. Therefore, it is assumed that the costs of dismantling this miscellaneous equipment are proportional to the volume of the reactor pressure vessel. Thus, the scaling factor for the miscellaneous equipment is the reactor vessel volume of the jth plant divided by the reactor vessel volume of the reference plant. These scaling factors, together with the plant power ratings and the plant reactor vessel volumes, are presented in Table 0.1-19.

TABLE 0.1-18. Identifiable Costs for Immediate Dismantlement of the WNP-2 Reactor Building Contaminated Concrete Surfaces

<u>Cost Category</u>	<u>Immediate Dismantlement Cost (\$) (a)</u>
Material Disposal ^(b)	92 038 ^(c)
Dedicated Staff Labor ^(d)	27 622 ^(c)
Total	119 660

(a) Early 1978 dollars.

(b) Costs based on information presented in Table I.3-4 in Appendix I.

(c) Calculated as 64% of the cost shown in the respective table.

(d) Costs based on information presented in Table I.3-7.

TABLE 0.1-19. Scaling Factors for BWR Reactor Building Miscellaneous Equipment

<u>Power Plant</u>	<u>Plant Power Rating (MWt)</u>	<u>Volume of Reactor Vessel (m³)</u>	<u>Scaling Factor^(a)</u>
Vermont Yankee	1 593	376	0.586
Oyster Creek	1 600	406	0.632
Monticello	1 670	-- ^(b)	0.612 ^(c)
Cooper	2 381	465	0.724
Dresden 2 or 3	2 527	-- ^(b)	0.808 ^(c)
Peach Bottom 2 or 3	3 293	639	0.995
WNP-2	3 320	642	1.000

(a) Calculated as the ratio of the reactor vessel volume from the jth plant and that value from the reference plant, WNP-2 (Equation 0.1).

(b) Not available in the FSAR.

(c) Value inferred from least-squares fitted equation.

The equation of the "best-fit" line through the scaling factors is given below and is used to infer the scaling factors for the miscellaneous equipment.

$$SF_{MEj} = 0.231 + (2.284 \times 10^{-4}) PPR_j$$

The identifiable costs for immediate dismantlement of the miscellaneous Reactor Building equipment from WNP-2 are given in Table 0.1-20, abstracted from Tables I.3-4 and I.3-7 in Appendix I.

TABLE 0.1-20. Identifiable Costs for Immediate Dismantlement of the WNP-2 Reactor Building Miscellaneous Equipment

<u>Cost Category</u>	<u>Immediate Dismantlement Cost (\$) ^(a)</u>
<u>Equipment Disposal ^(b)</u>	
Control Rod Drive	14 350
Atmosphere Control	8 520
Condensate	1 910
Core Spray	10 680
Reactor Recirculation	32 220
Residual Heat Removal	20 350
Drain	2 830
In-core Instrumentation	1 720
<u>Dedicated Staff Labor ^(c)</u>	
Radiation Survey	12 550
Reactor Coolant System Decontamination	102 740
Containment Cleanup	12 950
Drain Systems Decontamination	51 260
Remove Equipment	132 480
Final Radiation Survey	4 180
<u>Total</u>	<u>408 740</u>

(a)Early 1978 dollars.

(b)Based on information presented in Table I.3-4 in Appendix I.

(c)Based on information presented in Table I.3-7.

0.1.1.4.4 Plant-Specific Scaling Factors for the Selected BWR Power Plants. In the preceding subsections, component-specific scaling factors are developed for the three selected components in the selected BWR power plants. The component-specific scaling factors for a given plant must be weighted by their respective cost factors and summed to calculate the plant-specific scaling factor for that plant. Once the plant-specific scaling factors for all the selected plants are calculated, the desired scaling factor relationship for estimating decommissioning costs and radiation doses can be derived.

The total identifiable costs for immediate dismantlement of the three selected components from the reference plant are summarized in Table 0.1-21, together with the cost factors for weighting the component-specific scaling factors.

TABLE 0.1-21. Summary of Identifiable Costs for Immediate Dismantlement of the WNP-2 Selected Components

<u>Component</u>	<u>Immediate Dismantlement Cost (\$) (a)</u>	<u>Cost Factor (b)</u>
Reactor Vessel	616 470	0.0764
Reactor Vessel Internals	3 213 530	0.3983
<u>Reactor Building</u>	<u>4 237 120</u>	<u>0.5253</u>
Totals	8 067120	1.0000

(a)Early 1978 dollars.

(b)Calculated as the ratio of the component cost and the total of the three component costs (Equation 0.2).

The plant-specific scaling factors for the selected BWR power plants are calculated using Equation 0.3 and are presented in Table 0.1-22, together with the supporting data.

0.1.1.4.5 Scaling Factor Relationship. The scaling factor relationship that can be used to estimate the costs and radiation doses of decommissioning any BWR is derived by a least-squares analysis of the plant-specific scaling

TABLE 0.1-22. Plant-Specific Scaling Factors for the Selected BWR Power Plants

Power Plant	Plant Power Rating (MWt)	Component-Specific Scaling Factors			Plant-Specific Scaling Factor(a)
		Reactor Vessel	Vessel Internals	Reactor Building	
Vermont Yankee	1 593	0.572	0.531	0.611	0.576
Oyster Creek	1 600	0.791	0.567	0.620	0.612
Monticello	1 670	0.588	0.617	0.612	0.612
Cooper	2 381	0.757	0.705	0.748	0.732
Dresden 2 or 3	2 527	0.782	0.731	0.826	0.785
Peach Bottom 2 or 3	3 293	0.910	0.734	0.995	0.885
WNP-2	3 320	1.000	1.000	1.000	1.000
Cost Factor		0.0764	0.3983	0.5253	

(a) Calculated as the sum of the component-specific scaling factors multiplied by their respective cost factors (Equation 0.3).

factors and their respective plant power ratings, as indicated previously by Equation 0.4 in Section 0.1.1.3.

The equation of the "best-fit" line through the plant-specific scaling factors is:

$$OSF = 0.267 + (2.035 \times 10^{-4}) PPR$$

where OSF is the overall scaling factor for determining decommissioning costs and radiation doses and PPR is the plant power rating of the BWR power plant in question. This equation must be normalized to yield the total estimated cost and radiation dose for a given decommissioning mode at the reference BWR when evaluated at $PPR_r = 3320$ MWt. Thus, the desired scaling factor relationship is:

$$OSF = 0.324 + (2.035 \times 10^{-4}) PPR \quad (0.6)$$

Equation 0.6 is used to determine specific decommissioning costs and radiation doses for certain BWR power plants as follows in Section 0.1.2.

0.1.2 Application of the Scaling Factor Relationship

The scaling factor relationship (Equation 0.6) developed in Section 0.1.1 is applied in this section to determine the following aspects of decommissioning BWR power plants of different sizes:

- cost for immediate dismantlement
- cost for passive safe storage
- cost for entombment
- radiation doses for these three decommissioning modes.

These aspects are discussed in turn in the following subsections.

0.1.2.1 Cost for Immediate Dismantlement of a Different Size BWR

Immediate dismantlement is the removal from the site, immediately following plant shutdown, of all the materials having radioactivity levels greater than permitted for unrestricted use of the property. A complete discussion of immediate dismantlement for the reference BWR is contained in Appendix I.

The cost for immediate dismantlement of the j th BWR power plant, C_{IDj} , can be estimated by multiplying the cost for immediate dismantlement of the reference BWR, C_{IDr} , by OSF_j (Equation 0.6 evaluated at PPR_j), as follows:

$$C_{IDj} = C_{IDr} OSF_j$$

For illustration, Table 0.1-23 presents the estimated costs for immediate dismantlement of two different size BWR power plants as well as those for the reference BWR, WNP-2.

0.1.2.2 Cost for Passive Safe Storage of a Different Size BWR

As an alternative to immediate dismantlement, a nuclear power plant may be placed in passive safe storage, with possible deferred dismantlement at some future time. Continuing care (surveillance and maintenance) of the power plant facilities is required throughout the safe storage period. A complete discussion of passive safe storage for the reference BWR is contained in Appendix J.

Preparations for passive safe storage and deferred dismantlement after a finite period of time include many of the same operations as immediate

TABLE 0.1-23. Estimated Costs for Immediate Dismantlement
of Different Size BWR Power Plants

Power Plant	WNP-2	Cooper	Vermont Yankee
Plant Power Rating, PPR_j (MWt)	3 320	2 381	1 593
OSF_j (a)	1.000	0.809	0.648
Immediate Dismantlement Cost (\$ millions) (b)	43.6 (c)	35.3	28.3

(a) Overall Scaling Factor; Equation 0.6 evaluated at PPR_j .

(b) Early 1978 dollars; rounded to the nearest \$100,000; does not imply accuracy to that many significant figures.

(c) Based on information presented in Table I.3-1 in Appendix I; does not include other possible costs that are delineated in Table I.3-1 and range from \$23.1 to 29.0 million.

dismantlement. Therefore, the cost of preparing a particular BWR power plant for safe storage, when compared with that cost for the reference BWR, are assumed to be in roughly the same proportion as the costs for immediate dismantlement of the respective plants. Thus, the cost for preparations for passive safe storage of the j th plant, C_{pj} , can again be estimated by multiplying the cost for preparations for passive safe storage of the reference BWR, C_{pr} , by OSF_j (Equation 0.6 evaluated at PPR_j), as follows:

$$C_{pj} = C_{pr} OSF_j$$

Similarly, for deferred dismantlement,

$$C_{DDj} = C_{DDr} OSF_j$$

Continuing care for passive safe storage involves caretaking operations, the costs of which are not expected to vary a great deal with the size of the power plant, with the possible exception of the cost for energy. The cost for energy is only a small portion of the total annual continuing care cost, AC_{CC} , and a variation in that cost does not markedly affect the total; thus, no allowance is made for power plant size in the estimated cost of continuing care.

Based on the above information and definitions, the relationship for estimating the total cost for passive safe storage of the jth BWR power plant, including deferred dismantlement after n years of continuing care, C_{PSSj} , is:

$$C_{PSSj} = (C_{Pr} + C_{DDr}) OSF_j + (AC_{CC}) n$$

For illustration, Table 0.1-24 lists the estimated costs for passive safe storage of two different size BWR power plants as well as those for the reference BWR.

TABLE 0.1-24. Estimated Costs for Passive Safe Storage of Different Size BWR Power Plants

<u>Power Plant</u>	<u>WNP-2</u>	<u>Cooper</u>	<u>Vermont Yankee</u>
<u>Plant Power Rating, PPR_j (Mwt)</u>	3 320	2 381	1 593
<u>OSF_j (a)</u>	1.000	0.809	0.648
<u>Preparations Cost (\$ millions)^(b,c)</u>	21.3	17.2	13.8
<u>Continuing Care Cost (\$ millions)^(b,d)</u>			
for 10 years	0.6	0.6	0.6
for 30 years	2.0	2.0	2.0
for 50 years	3.4	3.4	3.4
for 100 years	6.9	6.9	6.9
<u>Deferred Dismantlement Cost (\$ millions)^(b,e)</u>			
after 10 years	35.5	28.7	23.0
after 30 years	35.5	28.7	23.0
after 50 years	26.4	21.4	17.1
after 100 years	<u>26.3</u>	<u>21.3</u>	<u>17.0</u>
<u>Totals, Passive Safe Storage (\$ millions)^(b,f)</u>			
Deferred Dismantlement after			
10 years	57.4	46.5	37.4
30 years	58.8	47.9	38.8
50 years	51.1	42.0	34.3
100 years	54.5	45.4	37.7

(a)Overall Scaling Factor (Equation 0.6 evaluated at PPR_j).

(b)Early 1978 dollars; rounded to the nearest \$100,000; does not imply accuracy to that many significant figures.

(c)Cost for WNP-2 based on information presented in Table J.5-1 in Appendix J; does not include other possible costs that are delineated in Table J.5-1 and range from \$5.5 to \$9.7 million.

(d)Cost for WNP-2 based on information presented in Table J.7-3.

(e)Cost for WNP-2 based on information presented in Table J.7-2.

(f)Sum of costs for preparations, continuing care for n years, and deferred dismantlement after n years.

0.1.2.3 Cost for Entombment of a Different Size BWR

Another alternative to immediate dismantlement is entombment, which is the encasement of all radioactive materials remaining on the site in concrete or other structural material that is sufficiently strong and structurally long-lived to assure retention of the radioactivity until it has decayed to levels that permit unconditional release of the site. Continuing care of the entombment structure is required until such time or until all radioactivity is removed from the site. Entombment of the reference power plant is discussed in detail in Appendix K.

Basically, the operations required for preparations for entombment of a BWR include dismantlement and storage, inside the Primary Containment, of as much as possible of the contaminated material located elsewhere in the plant. The Primary Containment is then sealed to prevent the release of any radioactivity. The rest of the contaminated material is dismantled and shipped to an offsite nuclear waste disposal facility.

The entombment preparations are similar in nature to the immediate dismantlement operations; many of the same tasks are included. Therefore, it is again assumed that the ratio of the entombment costs for a particular BWR power plant to those of the reference BWR is the same as the ratio of the immediate dismantlement costs for the same plants. Thus, the estimated entombment preparation cost for the j th plant, C_{Ej} , is equal to the product of the estimated entombment preparation cost for the reference BWR, C_{Er} , and OSF_j (Equation 0.6 evaluated at PPR_j), as follows:

$$C_{Ej} = C_{Er} OSF_j$$

Again, no change in total annual continuing care cost during entombment, AC_{CCE} , can be assumed for a different size BWR power plant. Therefore, the total cost for entombment of the j th BWR power plant after n years of continuing care, C_{TE} , is defined by:

$$C_{TEj} = C_{Er} OSF_j + (AC_{CCE}) n$$

Not included are the costs associated with whatever actions are taken at the time the facility owner decides to terminate the nuclear license. Since those actions depend on the length of the continuing care period and on the extant regulations, no attempt is made in this analysis to predict what costs may be incurred. It is important to realize, however, that some costs will be incurred at that time, and that those costs could approach the cost for deferred dismantlement after a period of passive safe storage (see Table 0.1-24).

For illustration, Table 0.1-25 presents the estimated costs for entombment of two different size BWR power plants as well as those for the reference BWR. Two types of entombment are shown in the table: 1) "without internals" assumes the removal and offsite disposal of the reactor vessel internals, and 2) "with internals" assumes the intact retention of the internals inside the entombment structure.

TABLE 0.1-25. Estimated Costs for Entombment of Different Size BWR Power Plants

<u>Power Plant</u>	<u>WNP-2</u>	<u>Cooper</u>	<u>Vermont Yankee</u>
<u>Plant Power Rating, PPR_j (MWt)</u>	3 320	2 381	1 593
<u>OSF_j (a)</u>	1.000	0.809	0.648
<u>Preparations Cost (\$ millions) (b)</u>			
without internals (c)	40.6	32.9	26.3
with internals (d)	35.0	28.3	22.7
<u>Continuing Care Cost (\$ millions) (b,e)</u>			
for 100 years	4.0	4.0	4.0
for 200 years	8.0	8.0	8.0
<u>Totals, Entombment (\$ millions) (b)</u>			
without internals for 100 years	44.6	36.9	30.3
with internals for 200 years	43.0	36.3	30.7

(a) Overall Scaling Factor; Equation 0.6 evaluated at PPR_j.

(b) Early 1978 dollars; rounded to the nearest \$100,000; does not imply accuracy to that many significant figures.

(c) Cost for WNP-2 based on Entombment Scenario-1 information presented in Table K.3-1 in Appendix K; does not include other possible costs delineated in Table K.3-1 for this scenario and ranging from \$16.2 to \$21.4 million.

(d) Cost for WNP-2 based on Entombment Scenario-2 information presented in Table K.3-1 for this scenario and ranging from \$15.6 to \$19.5 million.

(e) Cost based on information present in Table K.3-1 for both entombment scenarios.

0.1.2.4 Radiation Doses for Decommissioning a Different Size BWR

The occupational and public radiological doses resulting from the different decommissioning modes considered in this report (immediate dismantlement, passive safe storage, and entombment) are discussed in detail for the reference power plant in Appendices I, J, K, and N.

The radiation produced by the activated/contaminated materials during a decommissioning operation is directly proportional to the amount of similar material types in the different power plants. The radiation doses to the decommissioning workers, to the transport workers, and to the public are roughly proportional to the mass of such materials handled. Since the cost for immediate dismantlement is largely a function of the mass of these materials handled and the amount of radioactivity present in them, the radiation doses are proportional to the costs of dismantlement.

It follows that the radiation doses to the workers and to the public are proportional to the power rating of the BWR power plant being decommissioned. Thus, an expected radiation dose for a particular decommissioning mode at the j th plant, D_{dj} , can be estimated by multiplying the corresponding radiation dose estimated for the reference BWR, D_{dr} , by OSF_j (Equation 0.6 evaluated at PPR $_j$), as follows:

$$D_{dj} = D_{dr} OSF_j$$

For illustration, Table 0.1-26 presents the estimated radiation doses for decommissioning two different size BWR plants as well as those for the reference BWR.

0.1.3 Cost For Demolition of a Different Size BWR

Following a BWR's decommissioning, it may be desirable to demolish the decontaminated and uncontaminated plant structures and restore the site to pre-facility conditions. In this study, demolition includes removing above-grade structures to a depth of 1 m below grade and fracturing base floors and sub-structures sufficiently to permit normal drainage of surface water, and site

**TABLE 0.1-26. Estimated Radiation Doses for Decommissioning
Different Size BWR Power Plants**

Power Plant		WNP-2	Cooper	Vermont Yankee
Plant Power Rating, PPR_j (Mwt)		3 320	2 381	1 593
OSF_j (a)		1.000	0.809	0.648
Immediate Dismantlement Doses (man-rem) (b)				
	Occupational	1 845 ^(c)	1 493	1 196
	Transportation	110 ^(d)	89	71
	Public	10 ^(d)	9	6
Passive Safe Storage Doses (man-rem) (b)				
Preparations				
	Occupational	375 ^(e)	303	243
	Transportation	22 ^(d)		
	Public	2 ^(d)		
Continuing Care (f)				
for 10 years	Occupational	1.3	1.3	1.3
for 30 years	Occupational	6.5	6.5	6.5
for 50 years	Occupational	10	10	10
for 100 years	Occupational	10	10	10
Deferred Dismantlement (g)				
after 10 years	Occupational	495	400	321
after 30 years	Occupational	36	29	23
after 50 years	Occupational	3	2	2
after 100 years	Occupational	<1	<1	<1
Entombment Doses (man-rem) (b)				
without internals	Occupational	1 684 ^(h)	1 362	1 091
	Transportation	69 ^(d)	56	45
	Public	7 ^(d)	6	5
with internals	Occupational	1 573 ^(h)	1 273	1 019
	Transportation	51 ^(d)	41	33
	Public	5 ^(d)	4	3

(a) Overall Scaling Factor (Equation 0.6 evaluated at PPR_j).

(b) Rounded to the nearest whole man-rem; does not imply accuracy to that many significant figures.

(c) Based on information presented in Table I.4-1 in Appendix I.

(d) Based on information presented in Table N.5-2 in Appendix N.

(e) Based on information presented in Table J.6-1 in Appendix J.

(f) Based on information presented in Table J.6-2; no transportation or public doses expected.

(g) Based on information presented in Table J.7-4; transportation and public doses are not available.

(h) Based on information presented in Table K.4-1 in Appendix K.

restoration includes backfilling with concrete rubble as possible to 1 m below grade, leveling to grade with indigenous topsoil, and seeding with native ground cover. More details concerning demolition and site restoration of the reference BWR are contained in Appendix L.

Table L.3-1 in Appendix L lists four structures whose demolition costs contribute heavily to the total demolition cost: 1) the Reactor Building and Primary Containment, 2) the Turbine Generator Building, 3) the Radwaste and Control Building, and 4) the cooling towers. In the following subsections, individual scaling relationships to estimate the demolition costs for the first, second, and fourth of these structures are developed and combined to obtain an overall relationship for estimating the demolition cost of an entire BWR power plant.

0.1.3.1 Reactor Building and Primary Containment Demolition Cost

All BWR power plants have Reactor Buildings and Primary Containments. The Reactor Buildings for the selected BWR power plants are of similar design but the Primary Containments are of two different designs, designated Mark I and Mark II.

The Primary Containment of a BWR is made up of a drywell and a wetwell (pressure suppression chamber). The Mark I and Mark II designs, as well as the later Mark III design, are described in more detail later in Section 0.5. Both the Mark I and the Mark II designs consist of steel vessels surrounded by reinforced-concrete biological shields. Demolition requires the steel vessels to be sectioned and disposed of in addition to the concrete being broken up and disposed of. Thus, the demolition cost is proportional to the surface area of the Primary Containment. Comparison data for the Primary Containment structures of these two different designs are given in Table 0.1-27.

The steel plate used in the containment vessel is about the same thickness for both power plants. It is assumed that the concrete walls are also of comparable thickness and that their construction (reinforcing steel and type of concrete) is similar. Thus, the data in Table 0.1-27 indicate that it costs about 22% more to demolish a Mark I containment than to demolish a Mark II containment.

TABLE 0.1-27. Comparison Data for BWR Mark I and Mark II Containment Designs

<u>Power Plants</u>	<u>Peach Bottom 2 or 3</u>	<u>WNP-2</u>
<u>Plant Power Rating (MWt)</u>	3 293	3 320
<u>Containment Design</u>	Mark I	Mark II
<u>Drywell: Volume (m³)</u>	6 089	7 379
Surface Area (m ²)	1 890	2 299
<u>Wetwell: Volume (m³)</u>	7 453	9 615
Surface Area (m ²)	3 155	1 842
<u>Total Surface Area (m²)</u>	5 045	4 141

From the data given in Table L.3-2 in Appendix L, the cost of removing the Primary Containment Vessel is 5.4% of the total estimated cost of demolishing the Reactor Building and the Primary Containment Vessel. Therefore, the difference in demolition cost between a Mark I containment and a Mark II containment is about 1% of the total cost for demolition of the entire Reactor Building. This difference is small enough, especially in comparison with the total demolition cost for the entire plant, that it can be safely neglected.

The cost of demolishing a Reactor Building is proportional to the amount of material in the building. It is assumed that the walls and shielding structures are the same thickness for the different reactor buildings. The demolition cost is then roughly proportional to the wall area. The relationship (developed earlier in Section 0.1.1.4.3) for estimating the wall area of the Reactor Building of the jth plant relative to the wall area of the reference BWR Reactor Building is:

$$SF_{Cj} = 0.837 + (2.334 \times 10^{-5}) PPR_j$$

This relationship can also be used to estimate the cost of demolishing the Reactor Building and Primary Containment of the jth plant as a function of that cost for the reference power plant.

From Table L.3-2, the total cost of demolishing the Reactor Building and Primary Containment Vessel is \$5,169,000. When the above relationship is multiplied by this cost, a relationship is obtained for determining the demolition

cost for the Reactor Building and Primary Containment Vessel of any BWR power plant, DC_{Rj} , as a function of the plant power rating (PPR_j), as follows:

$$DC_{Rj} = \$4,326,000 + \$120.60 (PPR_j)$$

0.1.3.2 Turbine Generator Building Demolition Cost

The cost of demolishing the Turbine Generator Building is also proportional to the surface area of the building. Since the ratio of the surface areas of two rectangular prisms is proportional to the ratio of their volumes raised to the two-thirds power, the scaling factor for the Turbine Generator Building of the j th plant is the ratio of its volume and the volume of the WNP-2 Turbine Generator Building, the ratio raised to the two-thirds power $(V_{Tj}/V_{Tr})^{2/3}$.

The scaling factors for the Turbine Generator Buildings of selected BWR power plants are presented in Table 0.1-28, together with the plant power ratings and the building volumes.

TABLE 0.1-28. Scaling Factors for BWR Turbine Generator Buildings

<u>Power Plant</u>	<u>Plant Power Rating (MWt)</u>	<u>Building Volume (m³)</u>	<u>Scaling Factor(a)</u>
Vermont Yankee	1 593	109 020	0.746
Oyster Creek	1 600	121 760	0.657
Cooper	2 381	224 700	0.989
Peach Bottom 2 or 3	3 293	265 040	0.940
WNP-2	3 320	228 400	1.000

(a) Calculated as the ratio of the volume of the Turbine Generator Building from the j th plant and that volume from the reference BWR, WNP-2, the ratio raised to the two-thirds power (Equation 0.1 applicable).

The equation of the "best-fit" line through the scaling factors is:

$$SF_{Tj} = 0.491 + (1.540 \times 10^{-4}) PPR_j$$

where PPR_j is the plant power rating of the j th plant.

The cost of demolishing the Turbine Generator Building of the reference power plant is given in Table L.3-1 as \$3,364,000. The above relationship when multiplied by this cost gives a relationship for estimating the demolition cost for any BWR Turbine Generator Building, DC_{Tj} , as a function of the power rating, as follows:

$$DC_{Tj} = \$1,652,000 + \$518.10 (PPR_j)$$

0.1.3.3 Cooling Tower Demolition Cost

The cooling capacity of a cooling tower can be assumed roughly proportional to its internal volume (cross-sectional area times distance of air flow). It can also be assumed that the height/diameter ratio is about the same and the wall thickness does not vary widely for different sizes of cooling towers in the range of thermal powers being considered. If the shape is simplified to that of a right-circular cylinder, the volume of concrete in the cooling tower body varies approximately as the ratio of the plant power rating of the j th plant to that of the reference plant, the ratio raised to the two-thirds power $[(PPR_j/3320)^{2/3}]$. Since the demolition cost of a cooling tower is roughly proportional to the volume of concrete in the shell, the relationship to estimate the demolition cost of the cooling towers at the j th power plant, DC_{Cj} , obtained by multiplying this ratio by the cost for demolition of the cooling towers at the reference plant (\$1,252,000 in Table L.3-1), is:

$$DC_{Cj} = \$5,624.20 (PPR_j)^{2/3}$$

0.1.3.4 Cost for Demolition of a Different Size BWR

Since the combined demolition costs for the three aforementioned structures do not equal the total cost for demolition of the entire reference plant, the relationship for estimating the total demolition cost of a different size BWR (TDC_j) is derived by summing the three individual demolition cost relationships for these structures and normalizing the resultant relationship to yield the total demolition cost for the reference BWR when evaluated at the reference plant power rating. The relationship is:

$$\begin{aligned} TDC_j &= (NF) (DC_{Rj} + DC_{Tj} + DC_{Cj}) \\ &= (NF) [\$5,978,000 + \$638.70 (PPR_j) + \$5624.20 (PPR_j)^{2/3}] \end{aligned}$$

where NF is the normalization factor and equals 1.77 when evaluated at $TDC_r = \$16,555,000$ (from Table L.3-1 in Appendix L) and $PPR_r = 3320$ MWt. Therefore, the relationship becomes:

$$TDC_j = \$10,584,030 + \$1131 (PPR_j) + \$9957 (PPR_j)^{2/3} \quad (0.7)$$

For illustration, the estimated demolition costs for two different size BWRs as well as for the reference BWR are listed in Table 0.1-29.

TABLE 0.1-29. Estimated Demolition Costs for Different Size BWR Power Plants

<u>Power Plant</u>	<u>WNP-2</u>	<u>Cooper</u>	<u>Vermont Yankee</u>
<u>Plant Power Rating, PPR_j (MWt)</u>	3 320	2 381	1 593
<u>Demolition Cost (\$ thousands)^(a)</u>	16 555 ^(b)	15 052	13 744

(a) Early 1978 dollars; rounded to the nearest \$1,000; does not imply accuracy to that many significant figures.

(b) Based on information presented in Figure L.3-1 in Appendix L.

0.2 SENSITIVITY OF OCCUPATIONAL RADIATION DOSES AND DECOMMISSIONING COSTS TO INCREASED RADIATION DOSE RATES

The radiation dose rate data presented in Appendix D are based on composites of measured data from seven BWRs that had operated for 3 to 8 effective full-power years. It is likely the depositions of radioactive contamination within the systems of those plants had probably not reached equilibrium levels by that time. Data on the trends in radiation dose rates from BWR coolant system piping suggest that equilibrium values may be two or three times greater than the values presented in Appendix D.⁽⁹⁾

In this section, the effects of a three-fold increase in the radiation dose rates on the occupational radiation doses and on the decommissioning costs are determined. It is recognized that radiation dose rates of that magnitude are unlikely to exist in operating plants since the doses to operations and maintenance personnel would be prohibitive. Chemical decontamination would likely be performed at intervals to keep the radiation dose rates at or below the present levels. Thus, the postulated three-fold increase probably represents an upper-limit situation.

0.2.1 Occupational Radiation Doses

For purposes of this analysis, the occupational radiation dose data given in Appendices I, J, and K for immediate dismantlement, preparations for passive safe storage, and entombment, respectively, are combined into general tasks that have common work conditions and requirements and that are used to determine the effects of a three-fold increase in radiation dose rates. It is assumed that the exposure times and the work conditions and requirements remain the same. Thus, an increase in radiation dose rate produces a corresponding increase in radiation dose.

The three-fold occupational radiation doses for immediate dismantlement, preparations for passive safe storage, and entombment are presented in Tables 0.2-1, 0.2-2, and 0.2-3, respectively. Those decommissioning tasks likely to be affected by the increased radiation dose rates are so designated. Some of the remaining tasks are performed using remote or semi-remote handling and shielding and are not significantly affected by the increased dose rates.

TABLE 0.2-1. Three-Fold Occupational Radiation Doses for Immediate Dismantlement

Building	Total Dose ^(a) (man-rem)	Three-Fold Dose ^(b) (man-rem)
<u>General Task</u>		
<u>Reactor Building/Primary Containment</u>		
* ^(c) Install Supplemental HEPA Filters	3.38	3.38
Initial Radiation Survey	0.50	1.50
Remove and Ship Reactor Vessel Internals	108.64	108.64
Remove and Ship Reactor Vessel	32.65	32.65
*Remove and Ship Containment Internals	301.74	905.23
*Decontaminate Systems	17.23	51.69
Drain and Clean Pools	3.91	3.91
*Remove Reactor Building Equipment	322.94	968.82
Remove Contaminated Concrete	19.27	19.27
Final Radiation Survey	0.07	0.07
Subtotals, Reactor Building ^(d)	810	2 095
<u>Turbine Generator Building</u>		
Install Supplemental HEPA Filters	9.29	9.29
*Initial Radiation Survey	0.20	0.60
*Preparations for Dismantlement	0.86	2.58
*Remove and Ship Turbine	8.43	25.29
*Decontaminate and Drain Systems	0.39	1.17
*Remove and Ship Condenser	34.72	104.16
*Remove Piping and Equipment	137.24	411.73
Remove Contaminated Concrete	1.94	1.94
Final Radiation Survey	0.04	0.04
Subtotals, Turbine Generator Building ^(d)	193	559
<u>Radwaste and Control Building</u>		
*Initial Radiation Survey	0.80	2.40
*Decontaminate Systems	18.47	55.41
*Remove Piping and Equipment	492.19	1 476.57
Remove Miscellaneous Equipment	7.77	7.77
Remove Contaminated Concrete	10.35	10.35
Final Radiation Survey	0.05	0.05
Subtotals, Radwaste and Control Building ^(d)	530	1 553
<u>Ancillaries</u>		
Discharge and Ship Fuel	81.04	81.04
*Operate Radwaste Systems	20.84	62.52
*Routine Radiation Surveys	6.35	19.05
Package Dry Solid Wastes	38.80	38.80
Miscellaneous	165	165
Subtotals, Ancillaries ^(d)	312	366
Totals ^(d)	1 845	4 573

(a)Based on data presented in Table I.4-1 in Appendix I.

(b)Calculated by multiplying the effected total dose by 3.

(c)Indicates those general tasks that are likely affected by an increase in radiation dose rates.

(d)Rounded to the nearest whole man-rem; does not imply accuracy to that many significant figures.

TABLE 0.2-2. Three-Fold Occupational Radiation Doses for Preparations for Passive Safe Storage

Building	Total Dose ^(a) (man-rem)	Three-Fold Dose ^(b) (man-rem)
<u>General Task</u>		
<u>Reactor Building/Primary Containment</u>		
* ^(c) Install Supplemental HEPA Filters	3.38	3.38
*Initial Radiation Survey	0.50	1.50
Drain and Wash Pools	3.60	3.60
*Decontaminate Systems	16.78	50.35
*Clean Up, Stage, and Shield Hot Spots	5.71	17.13
Cover Pools; Seal Access to Containment	6.15	6.15
*Isolate Equipment and Rooms	25.61	76.83
*Decontaminate Accessible Areas	6.35	19.05
*Deactivate Utilities; Install	5.75	17.24
Additional Security Equipment		
*Final Radiation Survey	0.30	0.90
<u>Subtotals, Reactor Building^(d)</u>	<u>74</u>	<u>196</u>
<u>Turbine Generator Building</u>		
Install Supplemental HEPA Filters	9.30	9.30
*Initial Radiation Survey	0.18	0.54
*Drain and Clean Tanks and Condenser	0.56	1.68
Decontaminate Drain System	0.31	0.31
*Isolate Equipment and Rooms	3.30	9.90
*Decontaminate Accessible Areas	3.29	9.87
*Deactivate Utilities; Install	1.07	3.21
Additional Security Equipment		
*Final Radiation Survey	0.15	0.45
<u>Subtotals, Turbine Generator Building^(d)</u>	<u>18</u>	<u>35</u>
<u>Radwaste and Control Building</u>		
*Initial Radiation Survey	0.07	0.21
*Chemically Decontaminate Equipment	16.05	48.15
Chemically Decontaminate Drains	0.43	0.43
*Isolate Equipment and Rooms	45.24	135.72
*Decontaminate Accessible Areas	35.44	106.32
*Deactivate Utilities; Install	1.34	4.02
Additional Security and Other Equipment		
*Final Radiation Survey	0.13	0.39
<u>Subtotals, Radwaste and Control Building^(d)</u>	<u>99</u>	<u>295</u>
<u>Ancillaries</u>		
Discharge and Ship Fuel	81.04	81.04
*Operate Radwaste Systems	20.63	61.89
*Routine Radiation Surveys	3.51	10.53
Package Dry Solid Wastes	13.29	13.29
Miscellaneous	66	66
<u>Subtotals, Ancillaries^(d)</u>	<u>184</u>	<u>233</u>
<u>Totals^(d)</u>	<u>375</u>	<u>759</u>

(a)Based on data presented in Table J.6-1 in Appendix J.

(b)Calculated by multiplying the effected total dose by 3.

(c)Indicates those general tasks that are likely affected by an increase in radiation dose rates.

(d)Rounded to the nearest whole man-rem; does not imply accuracy to that many significant figures.

**TABLE 0.2-3. Three-Fold Occupational Radiation Doses
for Entombment**

Building	General Task	Total Doses		Three-Fold Doses ^(a)	
		Without Internals ^(b) (man-rem)	With Internals ^(c) (man-rem)	Without Internals (man-rem)	With Internals (man-rem)
Reactor Building/Primary Containment					
*(d)	Install Supplemental HEPA Filters	3.38	3.38	3.38	3.38
	Initial Radiation Survey	0.50	0.50	1.50	1.50
	Remove and Ship Reactor Vessel Internals	106.60	--(e)	106.60	--
	*Decontaminate Systems	16.95	16.95	50.85	50.85
	Drain and Clean Pools	3.74	3.74	3.74	3.74
	*Clean Up, Stage, and Shield Hot Spots	8.15	8.15	24.45	24.45
	*Remove and Relocate Piping and Equipment	353.89	353.89	1 061.67	1 061.67
	*Prepare Containment Structure	139.14	139.14	417.38	417.38
	Remove Contaminated Concrete	16.67	16.67	16.67	16.67
	Remove HVAC and Disable Crane	7.39	7.39	7.39	7.39
	Final Radiation Survey	0.06	0.06	0.06	0.06
	Install Monitoring Equipment; Deactivate Utilities	0.73	0.73	0.73	0.73
	Subtotals, Reactor Building ^(f)	657	551	1 694	1 588
Turbine Generator Building					
	Install Supplemental HEPA Filters	9.30	9.30	9.30	9.30
	*Initial Radiation Survey	0.20	0.20	0.60	0.60
	*Clean Up and Stage	0.82	0.82	2.46	2.46
	*Decontaminate and Drain Systems	0.39	0.39	1.17	1.17
	*Remove and Ship Turbine	7.89	7.89	23.67	23.67
	*Remove Piping and Equipment	139.89	139.89	417.67	417.67
	*Remove and Ship Condenser	33.96	33.96	101.88	101.88
	Remove Contaminated Concrete	2.03	2.03	2.03	2.03
	Remove HVAC and Electrical Systems	0.45	0.45	0.45	0.05
	Final Radiation Survey	0.04	0.04	0.04	0.04
	Subtotals, Turbine Generator Building ^(f)	195	195	559	559
Radwaste and Control Building					
	*Initial Radiation Survey	0.79	0.79	2.37	2.37
	*Clean Up	2.36	2.36	7.08	7.08
	*Decontaminate Systems	18.47	18.47	55.41	55.41
	*Remove Piping and Equipment	484.31	484.31	1 452.93	1 452.93
	Remove Contaminated Concrete	7.79	7.79	7.79	7.79
	Remove Miscellaneous Equipment	7.57	7.57	7.57	7.57
	Final Radiation Survey	0.05	0.05	0.05	0.05
	Subtotals, Radwaste and Control Building ^(f)	521	521	1 533	1 533
Ancillaries					
	Discharge and Ship Fuel	81.04	81.04	81.04	81.04
	*Operate Radwaste Systems	21.20	21.20	63.60	63.60
	*Routine Radiation Surveys	7.31	7.31	21.93	21.93
	Package Dry Solid Wastes	38.07	38.07	38.07	38.07
	Miscellaneous	163	158	163	158
	Subtotals, Ancillaries ^(f)	311	306	368	363
	Totals ^(f)	1 684	1 573	4 154	4 043

(a) Calculated by multiplying the respective effected total dose by 3.

(b) Based on data presented in Table K.4-1 in Appendix K.

(c) Based on information presented in Section K.4 of Appendix K for Scenario-2 Entombment.

(d) Indicates those general tasks that are likely affected by an increase in radiation dose rates.

(e) This task not included in this entombment scenario.

(f) Rounded to the nearest whole man-rem; does not imply accuracy to that many significant figures.

The other remaining events are performed after removal of the more radioactive components and are also not affected.

For the postulated three-fold increase in radiation dose rate, 4573 man-rem of occupational radiation dose are estimated (versus 1845 man-rem in Appendix I) for immediate dismantlement, 759 man-rem are estimated (versus 375 man-rem in Appendix J) for preparations for passive safe storage, and 4154 man-rem are estimated (versus 1684 man-rem in Appendix K) for entombment (without internals, entombment scenario 1 in Appendix K).

The impacts of these increased radiation doses on decommissioning costs are discussed in the following section.

0.2.2 Decommissioning Costs

It is obvious that at least twice as many workers are possibly required for each decommissioning mode to keep the individual radiation doses within limits. One alternative to increased manpower is to install temporary shielding in many areas to reduce the radiation dose rates to the workers during active decommissioning. A second alternative is to increase the effectiveness and scope of the chemical decontamination. A third alternative is to take a longer time to complete the active decommissioning.

The effects of the increased radiation dose rates on the following decommissioning costs, which are the only costs assumed affected, are discussed in the following subsections, with conclusions in the last subsection.

- staff labor costs
- chemical decontamination costs
- disposal costs.

0.2.2.1 Staff Labor Costs

The impact of increased radiation dose rates on staff labor costs will vary, depending on the decommissioning mode selected. For both immediate dismantlement and entombment, because of the large amount of direct contact work on the piping systems and associated equipment, the impact can be significant. For preparations for passive safe storage, the fraction of direct contact work is much less, resulting in a much smaller impact. These impacts are examined in the following subsections.

Immediate Dismantlement. One approach used in the nuclear industry when faced with jobs involving high radiation dose rates is to employ temporary workers, work them until they have received their allowable radiation dose, and then replace them with new temporary workers, and so on until the job is finished. Because of the time required to train temporary workers and because of their lack of familiarity with the plant, it is estimated that these workers are only about 60% as efficient as the workers who come from the plant's operating staff. Thus, the staff labor costs for the decommissioning workers increase by 60 to 70%. This results in an increase in staff labor costs of about \$6 million for immediate dismantlement, not including a contingency. This conclusion is based on the manpower costs given in Table I.3-6 in Appendix I and includes the following staff categories: 1) utility operators, 2) laborers, 3) craftsmen, and 4) health physics technicians.

Another approach is to employ temporary laborers to install temporary shielding wherever necessary to reduce the radiation dose rates to the levels presented in Appendix D. This approach offers only partial success in reducing the radiation dose to the decommissioning workers, since the piping and equipment have to be handled during removal, and shielding the workers during those operations would be difficult (if not impossible) in many cases. There is no clear-cut way to evaluate the cost for this approach, but it seems likely that the cost differential for staff labor is in the same range as the \$6 million estimated previously for the first approach. The cost of the additional shielding is estimated to be on the order of \$100,000. It is also likely that the radiation dose received during the placement of the shielding would cause the total occupational radiation dose for this alternative to be as high as if it were not used.

Preparations for Passive Safe Storage. The occupational radiation dose to workers during preparations for passive safe storage is increased by about 120% by the postulated three-fold increase in radiation dose rates throughout the facility. This increased dose could be received by the work force given in Appendix J without exceeding existing limits. Therefore, the impact of increased radiation dose rates on staff labor costs is minimal for this mode.

Entombment. Since entombment requires dismantlement of all areas outside Primary Containment, the impact of increased radiation dose rates on staff labor costs for entombment is similar to that for immediate dismantlement. Making the same assumptions about the use of temporary laborers as for immediate dismantlement, the staff labor costs for entombment increase by about \$6 million, not including a contingency.

0.2.2.2 Chemical Decontamination Costs

The chemical decontamination method chosen in Section H.5 of Appendix H uses the complexing agent EDTA combined with citric acid and oxalic acid in a one-step process. An average decontamination factor (DF) of 10 is assumed. Laboratory and in-reactor loop decontamination studies have shown that the dissolution of deposited radionuclide film proceeds in a decreasing exponential relative to the length of time of circulation.⁽¹⁰⁾ The data from these studies suggest that to reduce the radiation dose rate from internally deposited radionuclides in the piping systems, longer recirculation of the decontamination solutions is required. In addition, the thermolytic and radiolytic decomposition of EDTA during the decontamination process requires monitoring and intermittent addition of the decontamination chemicals throughout the process to maintain optimum conditions for removal of the radionuclides and their suspension in the circulating fluid. Since the reduction in radiation dose rates during decontamination tends to follow an exponential decay curve, the time required to obtain a DF of 3, in addition to the 10 assumed, increases by about 50% over that required for a DF of 10 only.

The principal cost impacts from an extended chemical decontamination effort are the cost for electricity to operate the recirculation pumps and the cost for additional staff labor. The cost for chemicals given in Appendix H (based on the 5% solution) is probably an upper-limit estimate, even when considering the cost for the chemicals added during the recirculation period; thus, no incremental expense is expected for the additional chemicals. Similarly, the volume of fluid to be processed is essentially the same as given in Appendix H (except when the scope of the chemical decontamination is increased to include additional BWR systems that, with the postulated three-fold radionuclide deposition, now meet the selection criterion on dose rate). The staff labor involved is

roughly proportional to the recirculation time, and the additional staff labor cost is small compared to the total decommissioning cost. Therefore, the incremental cost of achieving a DF of 30 compared to a DF of 10, is postulated to be small, probably less than \$100,000.

From a practical point of view, it is unlikely that dose rates in operating plants will be allowed to reach anything approaching three times present levels, because of the considerable difficulty such increases would cause for normal operation and maintenance. Studies are presently in progress within the nuclear industry to develop decontamination solutions that can be injected into the reactor coolant stream immediately following reactor shutdown and circulated through the primary coolant system with the fuel in place in the core. The decontamination chemicals and the suspended and dissolved radio-nuclides are removed from the coolant water by side-stream cleanup, resulting in reactor quality water by the end of the outage. It is anticipated that these decontaminations will produce a DF of about 2, which should be sufficient to keep the radiation dose rates at a manageable level for the life of the plant.

0.2.2.3 Disposal Costs

If the chemical decontamination is not increased, the residual radioactivity deposition in the piping and equipment would be about three times greater than is assumed in Appendix E. The effect of these higher residual radioactivity levels is seen principally in the area of container shielding, although additional containers may also be required.

Of the over 5000 containers used for disposal of the contaminated material during immediate dismantlement (see Section I.3 of Appendix I), it is estimated that less than 1000 require additional shielding. An allowance of \$100,000 is suggested as reasonable for the cost of this additional shielding. The higher residual contamination level has no significant effect on the disposal costs for the neutron-activated materials, since the radiation dose rate from the activated constituents is much larger than the radiation dose rate from the deposited radioactivity.

0.2.2.4 Conclusions

For the situation where the radiation dose rates from radionuclides deposited in the BWR systems are postulated to be three times the levels presented in Appendix D:

- If no corrective actions are taken to reduce the radiation dose rates, the accumulated radiation dose to the decommissioning workers increases about 2728 man-rem, and the total decommissioning cost increases about \$6 million for additional staff labor and about \$100,000 for additional container shielding (a total increase of about \$6.1 million) for immediate dismantlement. For preparations for passive safe storage, the radiation dose increases about 384 man-rem and there is essentially no change in the cost. For entombment, the radiation dose increases about 2470 man-rem and the cost increases about \$6 million.
- If corrective action is taken, such as an extended chemical decontamination following final shutdown, the total additional cost could be as much as \$100,000.

On the basis of these estimates, it appears that additional chemical decontamination is the most cost-effective approach to handling the higher initial radiation dose rates postulated. This approach is also more consistent with the principles of ALARA, since the occupational radiation dose associated with a chemical decontamination is relatively small, compared with the radiation dose associated with installing temporary shielding or with attempting to perform the dismantlement without additional shielding. In addition, it is likely that annual decontaminations of the reactor coolant system and related systems will be standard operating practice when decontamination solutions become available, thus preventing large buildups of radionuclides in the systems.

0.3 SENSITIVITY OF IMMEDIATE DISMANTLEMENT COSTS TO DIFFERENT CONTRACTUAL ARRANGEMENTS

Two basic premises used in Appendix I concerning the decommissioning work force for immediate dismantlement of the reference BWR are: 1) the majority of the staff consists of personnel who were members of the facility's operating

organization prior to shutdown of the facility, and 2) the operating utility functions as the general contractor for decommissioning.

Presently, numerous organizations provide specialized contract services to nuclear power plant owners during plant outages. It is likely that when a significant number of nuclear power plants are approaching retirement, organizations will form specifically to decommission such facilities. This section examines the postulated organization for and the impact on immediate dismantlement costs by a contract-type approach to decommissioning.

0.3.1 Postulated Organization for Decommissioning by Contract

The plant owner, as the licensee, is responsible for ensuring that the decommissioning work is performed in accordance with applicable regulations. Therefore, even in a contract situation, the plant owner is the prime contractor and provides the plant operations, industrial safety, and contracts and accounting as well as the quality assurance functions in addition to the overall management of the job. Specialty contractors provide the working-level supervision and the decommissioning workers. In this instance, it is assumed that separate contractors are employed to provide for health physics, security, transportation, and dismantlement. The other specialty contractors identified in Section H.2.6 of Appendix H (explosives, radwaste treatment, and environmental monitoring) become subcontractors to the dismantlement contractor. The functional organization presented in Figure H.1-1 in Appendix H is assumed to be modified to that shown in Figure 0.3-1.

0.3.2 Impact on Immediate Dismantlement Costs

The principal cost impacts associated with utilizing contractors to accomplish decommissioning of the reference BWR result from: contractual considerations, including the type of contract, the fees, and the incentives; contractor overhead rates; and contractor mobilization and demobilization. A discussion of these factors and of their impact on immediate dismantlement costs is presented in the following subsections.

NOTE: HEALTH PHYSICS TECHNICIANS AND CRAFTSMEN ARE ASSIGNED TO CREWS AS WORK SITUATIONS DEMAND

FIGURE 0.3-1. Postulated Organization for Decommissioning by Contract

0.3.2.1 Contractual Considerations

Clearly, the degree of work let out on contract differs depending on the work that the plant owner can perform. The type of contract is primarily conditioned by financial risk or uncertainty of performance. Where specific work packages are prepared and are suitable for definitive pricing, a firm fixed-price contract best serves the interests of both contracting parties. A definitive work scope is generally necessary for bid advertisement and enhances the prospect for both competition and cost savings. In the situation where the owner maintains day-to-day surveillance and supervision of decommissioning tasks, a less definitive work statement is called for. In this situation, a cost-reimbursement contract or one that specifies a fixed price per hour of labor is most suitable.

In the safe storage mode, continuing care services can probably be fixed-priced. These services (such as radiation monitoring, building maintenance, and security) are readily quantifiable. It is easily understood that the least definitive decommissioning work involves dismantlement. Work involving radioactivity or high-risk skills (e.g., explosives work) has built-in inefficiencies that result in significantly higher costs. Under these circumstances, very conservative estimating is to be anticipated from the bidders. Therefore, it is prudent for utilities to include an "inefficiency" factor in budgeting for such decommissioning tasks.

Budgeting is also impacted by the long-term planning required for decommissioning. Estimates older than 1 year degrade rapidly in accuracy, if they are available at all. General economic conditions, inflation rates, regulatory impacts, energy costs, and labor costs are all interdependent elements that require careful evaluation from either a budgeting or a cost proposal viewpoint.

Several factors need consideration when negotiating a contract. Selected contract terms and conditions that should be considered in a contract for BWR decommissioning are:

- indemnity and insurance
- third-party liability (public liability)

- work delays
- default
- work scope changes
- licenses, permits, and approvals
- safety
- quality assurance
- labor disputes
- overtime premiums
- hazardous material handling
- accident reporting
- salvage rights.

While provisions for all of these conditions are not required in a given contract situation, all deserve consideration. Before a meaningful estimate or proposal is possible, each contract situation must be evaluated for work scope, performance period, cost risk, and terms and conditions. There are no model contracts or auditable trails to study for specific experience in contracting for nuclear power plant decommissioning. This situation is not unique but it requires proper emphasis on the part of the utility management designating a contract procurement staff.

Incentives may be considered for early completion, for reduced cost, or for some other specific performance criterion. Such incentives must be carefully determined and, as a general rule, selected on the basis of easy measurement. However, a less-objective incentive in the form of an award fee may be negotiated. The award fee, determined on the basis of subjectively measured criteria after performance, is thus a bonus, either in whole or in part, and is in addition to the base fee.

Indirect expenses vary significantly from one contractor to another within the same industry. Overheads for staff labor range from 50 to 150% of the base salary costs. Contractor profit or fee is in the range of 10 to 15% of total contractor costs, depending on such things as the type of contract and the risk of performance.

To estimate the impacts of these contractual considerations on decommissioning costs, it is necessary to make several assumptions on how selected

costs are paid. For example, it is assumed that the utility pays the radioactive material transportation and disposal costs directly, rather than channeling these costs through the dismantlement contractor. The utility is also assumed to provide the energy needed for the job. On the other hand, the costs for special equipment and supplies, as well as those for specialty subcontractors, are assumed to flow through the dismantlement contractor and are subject to the contractor fee. Thus, of the items listed in Table I.3-1 in Appendix I, only staff labor, special equipment, miscellaneous supplies, and specialty contractors are subject to the fee. Another significant cost, not separately identified in Table I.3-1 but included in the disposal costs for radioactive materials, is for shipping containers, which are also assumed to be subject to the contractor fee.

0.3.2.2 Contractor Overhead Rates

Overhead rates applied to direct staff labor are significantly higher for the contractors than for the utility, because of the usually larger ratio of supervisory and support personnel to direct labor in contractor organizations. In addition, travel and living expenses increase the overhead costs. In view of these factors, an overhead rate of 110% is assumed applicable to all contractor direct staff labor costs.

The staff labor costs for the utility-only approach and for the combined utility-contractors approach are shown in Table 0.3-1. In the table, the projected staff labor requirements and costs, presented respectively in Tables I.2-2 and I.3-6 in Appendix I, are regrouped to reflect the organizational structure shown previously in Figure 0.3-1, and, for the utility-contractors approach, the contractors' staff labor costs are adjusted for the higher overhead rates. The combined utility-contractors operation results in a staff labor cost increase of \$5.0 million.

0.3.2.3 Mobilization and Demobilization

There are significant costs associated with establishing a contractor at the work site. These costs include providing temporary office facilities, obtaining the required special equipment, and assembling the work force. Similarly, there are costs associated with closing down the work site. These

TABLE 0.3-1. Staff Labor Requirements and Costs for Immediate Dismantlement by Contract

Position	Total Staff Labor Required (man-years)(a)	Total Staff Labor Costs (\$ thousands)(b)	
		Utility-Only(c)	Utility-Contractors
<u>Utility Staff</u>			
Decommissioning Superintendent	5.1	380.0	380.0
Secretary	5.1	103.5	103.5
Control Room Operator	17.1	495.9	495.9
Industrial Safety Specialist	4.8	210.2	210.2
Contracts and Accounting Specialist	5.1	195.9	195.9
Clerk	4.5	91.4	91.4
Quality Assurance Supervisor	5.1	223.4	223.4
Quality Assurance Engineer	6.0	236.4	236.4
Quality Assurance Technician	14.5	336.4	336.4
Consultant (Safety Review)	2.8	187.6	187.6
Subtotals, Utility Staff	71.0	2 460.7	2 460.7
<u>Contractors' Staffs (d)</u>			
Health Physics Supervisor	5.8	292.4	361.2
Secretary	4.0	81.2	113.4
Health Physicist	4.0	157.6	194.7
Senior Health Physics Technician	8.8	289.6	357.7
Health Physics Technician	50.0	1 255.1	1 757.1
Protective Equipment Attendant	7.0	162.4	227.4
Security Supervisor	3.5	115.2	142.3
Security Shift Supervisor	17.5	537.3	322.4
Security Patrolman	86.5	1 842.5	2 579.5
Dismantlement Supervisor	5.8	368.3	455.0
Clerk	3.0	60.9	85.3
Secretary	4.5	91.4	127.6
Shift Engineer	10.0	438.0	541.1
Craft Supervisor	7.4	292.8	361.7
Craftsman	93.2	2 525.9	3 536.3
Tool Crib Attendant	6.8	157.8	220.9
Crew Leader	31.5	1 152.7	1 423.9
Utility Operator	123.5	3 347.1	4 674.6
Laborer	54.3	1 406.6	1 969.5
Assistant Dismantlement Engineer	5.4	236.6	292.3
Radioactive Shipment Specialist	4.4	144.8	178.9
Procurement Specialist	4.7	144.3	178.3
Subtotals, Contractors' Staffs	540.7	15 100.5	20 101.1
Totals	611.7	17 561.2	22 561.8

(a)Based on information presented in Table I.2-3 in Appendix I.

(b)Accuracy to the nearest \$1000 is not implied.

(c)Based on information presented in Table I.3-6.

(d)Costs for utility-contractors operation are based on the base salaries shown in Table M.1-1 in Appendix M and include 110% overhead.

costs, called mobilization and demobilization costs, vary with the size and complexity of the job. For the immediate dismantlement of a large BWR, these costs are estimated to be about \$1.25 million.⁽¹¹⁾

0.3.2.4 Impact on Immediate Dismantlement Costs

The various cost items for the utility-contractors operation are separated into utility costs and contractor costs in Table 0.3-2 for comparison with the costs for the utility-only approach assumed for immediate dismantlement in Appendix I, which are also shown. A 15% fee is applied to all contractor costs.

TABLE 0.3-2. Comparison of Costs Estimated for Dismantlement Using the Utility-Only Approach or the Utility-Contractors Approach

Cost Category	Cost (\$ millions) ^(a)	
	Utility-Only ^(b)	Utility-Contractors
Radioactive Material Disposal	6.278	6.278 --- ^(c)
Shipping Containers	2.400	--- 2.400 ^(d)
Staff Labor	17.561	2.461 ^(d) 20.101 ^(d)
Energy	3.519	3.519 ---
Special Equipment	2.016	--- 2.016
Miscellaneous Supplies	1.859	--- 1.859
Specialty Contractors	0.356	--- 0.356
Nuclear Insurance	0.800	0.800 ---
Licensing Fees	0.051	0.051 ---
Mobilization and Demobilization	---	--- 1.250
Subtotals, Utility-Contractors Costs	---	13.109 27.982
Total Contractors' Fee (15%)	---	--- 4.197
Totals, Contractors' Costs	---	--- 32.179
Subtotals	34.840	45.288
Contingency (25%)	8,710	11,322
Totals, Immediate Dismantlement Costs	43.550	56.610
<u>Other Possible Costs^(e)</u>		
Spent Fuel Shipment	3.788	3.788
Demolition and Restoration	13.244	13.244
Deep Geologic Disposal	0.848	0.848
Fuel Channel Disposal	0.617	0.617
Subtotals	18,497	18,497
Contingency (25%)	4,624	4,624
Totals, Other Possible Costs	23.121	23.121

(a)Early 1978 dollars.

(b)Data from Table I.3-1 in Appendix I; shipping container costs abstracted from Section I.3.1.

(c)Implies "not applicable."

(d)Based on data presented in Table 0.3-1.

(e)See Table I.3-1 for further details.

The utility-contractors approach for immediate dismantlement of the reference BWR increases the total decommissioning cost by about 30% to about \$57 million. This total does not include any incentive or bonus payments that might be negotiated for an undertaking of this magnitude.

0.4 SENSITIVITY OF IMMEDIATE DISMANTLEMENT COSTS TO WASTE DISPOSAL CHARGES

The costs for disposal of the radioactive materials removed from the reference BWR during immediate dismantlement, as developed in Section I.3.1. of Appendix I, are based on the assumption that all of these materials are placed in a shallow-land burial facility. These costs are developed using the fee schedule for a commercially licensed radioactive waste burial ground, published in September 1977 (see Section M.5 of Appendix M). The alternative cost of placing selected portions of the reactor vessel internals in a deep geologic disposal facility is also estimated in Section I.3. The impact of increased disposal charges on the total cost for immediate dismantlement is examined in this section for both the shallow-land burial concept and the deep geologic disposal alternative. The costs referred to in this section do not include the 25% contingency.

0.4.1 Impact of Increases in Shallow-land Burial Charges

The disposal costs for immediate dismantlement of the reference BWR are summarized in Table 0.4-1. The total estimated cost for immediate dismantlement is \$34.840 million, as given in Table I.3-1 in Appendix I. From that table, it can be seen that the total disposal cost constitutes about 25% of the total immediate dismantlement cost. As seen in Table 0.4-1 under "burial," the actual costs at the disposal facility constitute only 35% of the total disposal cost. Thus, less than 9% of the dismantlement cost is sensitive to the fee schedule at the disposal facility. For example, a doubling of the burial cost from \$3.036 million to \$6.072 million causes an 8.7% increase in the total immediate dismantlement cost.

The burial ground charges shown in Section M.5 of Appendix M were increased on June 1, 1978. The fees for burial (in dollars per cubic foot) increased 79% for containers with low surface radiation dose rates and 30% for containers having

TABLE 0.4-1. Summary of Estimated Costs for Disposal of Radioactive Materials During Immediate Dismantlement

Radioactive Material Category	Disposal Cost (\$ millions) ^(a)			Total
	Containers	Transportation	Burial ^(b)	
Activated Materials ^(c)	0.393	0.816	1.091	2.300
Contaminated Materials ^(d)				
Reactor Building	0.639	0.609	0.622	1.870
Turbine Generator Building	0.765	0.814	0.739	2.417
Radwaste and Control Building	0.265	0.131	0.225	0.622
Dry Solid Wastes ^(e)	0.064	0.237	0.095	0.396
Wet Solid Wastes ^(f)	0.217	0.692	0.164	1.073
Totals	2.343	3.299	3.036	8.678
Percent of Total	27	38	35	100

(a) Early 1978 dollars.

(b) Includes handling, burial, liner, and curie costs as applicable.

(c) Data summarized from Table I.3-3 in Appendix I.

(d) Data summarized from Table I.3-4.

(e) Data summarized from Table I.3-5.

(f) Data summarized from Table H.5-9 in Appendix H.

high surface radiation dose rates, with a sliding scale increase in between. The basic curie surcharge per container was increased by 25%. The surcharge for personnel radiation exposure was increased by 117%. The new fee schedule is estimated to cause an increase in the total cost for immediate dismantlement from \$3.036 million to \$4.569 million, a 4.4% increase in the total immediate dismantlement cost.

0.4.2 Impact of Higher Deep Geologic Disposal Costs

The analysis presented in Section I.3.9.3 of Appendix I is based on placing the highly activated portions of the reactor vessel internals in intermediate-level storage at a deep geologic disposal facility, as an alternative to using shallow-land disposal. In that analysis, the unit cost for emplacement is assumed to be \$7,100/m³, based on information in Reference 12. Other estimates of the cost for deep geologic emplacement range from about \$9,000/m³ to about \$19,000/m³, depending on whether the geologic medium is salt or various types of crystalline rock.⁽¹³⁾

Table 0.4-2 presents the effects of higher deep geologic emplacement costs by comparing the costs of nearly doubling and tripling the \$7,100/m³ emplacement

TABLE 0.4-2. Estimated Costs for Deep Geologic Disposal of the Highly Activated Materials

		Emplacement Cost of:		
		<u>\$7 100/m³(a)</u>	<u>\$14 000/m³</u>	<u>\$21 000/m³</u>
Disposal Costs		0.631	1.245	1.867
Container Cost	\$0.320			
Cask Rental Cost	\$1.386	2.258	2.258	2.258
Transportation Cost	\$0.552			
Total Costs		2.889	3.503	4.125
Funds Not Used for Shallow-land Burial		(2.041)	(2.041)	(2.041)
Incremental Costs for Deep Geologic Disposal		0.848	1.442	2.084
Total Disposal Cost from Table 0.4-1		8.678	8.678	8.678
Total Disposal Costs		9.526	10.140	10.762
Percent Increase in Total Immediate Dismantlement Cost		2.4	4.1	6.0

(a)Based on information presented in Section I.3.9.3 of Appendix I.

cost assumed in Appendix I. It can be seen that doubling the emplacement cost per unit volume of waste increases the total immediate dismantlement cost by over 4%. By approximately tripling the emplacement cost per unit volume, the total cost is increased by 6%.

If all the activated materials, including the activated portion of the sacrificial shield, are emplaced in a deep geologic disposal facility, an estimated volume of about 467 m³ of material (based on the same assumptions used in Section I.3.9.3) requires emplacement. The total cost for such action, comprising about \$3.3 million for the emplacement (at \$7,100/m³), about \$1.7 million for the required 336 canisters, and about \$2.0 million for transportation, is about \$7.0 million. The cost for shallow-land burial of the highly activated materials is about \$2.3 million. Thus, the total immediate dismantlement cost in this instance increases by \$4.7 million (about 13.5%) over the cost associated with disposal of all wastes in a shallow-land burial facility.

From the discussion above, it can be seen that if only the highly activated materials are placed in a deep geologic disposal facility, the total immediate

dismantlement cost is not very sensitive to the emplacement cost (i.e., increasing about 4% by doubling the emplacement cost). On the other hand, if all of the activated materials are placed in a deep geologic disposal facility, the total cost increases more than three times this much with no change in the emplacement cost.

In the unlikely event that all radioactive materials from immediate dismantlement are placed in a deep geologic disposal facility, the cost increase becomes quite significant. If it is again assumed that the wastes are packaged with a 15% packing density (as in Section I.3.9.3 of Appendix I) so that the disposal volume is constant, 18,876 m³ of waste must be emplaced, at an emplacement cost (at \$7,100/m³) of about \$134 million. The required 15,730 canisters cost over \$78 million. Thus, the cost for waste disposal in this instance, even without cask and transportation costs, becomes more than \$212 million, for an over 700% increase in the total immediate dismantlement cost when shallow-land burial of all radioactive materials is assumed.

Obviously, there are strong incentives to minimize the volume of material placed in deep geologic disposal.

0.5 SENSITIVITY OF IMMEDIATE DISMANTLEMENT COSTS TO DIFFERENT BWR CONTAINMENT DESIGNS

The design of Primary Containment for BWR power plants has changed considerably since the first plant was constructed. The purpose of this section is to describe the various BWR containment designs and estimate their effect on the cost for immediate dismantlement.

0.5.1 BWR Primary Containment Design Description

The design of the reactor containment in the first few BWR power plants was not standardized. The first plants were all small (less than a few hundred MWt), and the containment designs varied. With the advent of plants larger than 1000 MWt, the containment design stabilized in the Mark I configuration. This was later superseded by the Mark II design which, in turn, has been superseded by the Mark III design in the most recent BWR power plants.

The differences in these three types of containment designs are discussed in the following subsections. The pertinent characteristics of these three different BWR containment designs, from three BWR power plants (including the reference plant) all of approximately the same thermal power rating, are listed in Table 0.5-1.

TABLE 0.5-1. Pertinent Characteristics of the Three Different BWR Containment Designs^(a)

<u>Power Plant</u>	<u>Peach Bottom 2 or 3</u>	<u>WNP-2</u>	<u>Perry 1 or 2</u>
<u>Containment Design</u>	Mark I	Mark II	Mark III
<u>Power Rating (MWt)</u>	3 293	3 320	3 579
<u>Suppression Pool</u>			
Pool Volume (m ³)	3 851	3 160	4 100
Specific Volume (m ³ /MWt)	1.17	0.95	1.15
<u>Wetwell</u>			
Volume (m ³)	7 453	9 614	13 500 ^(b)
Specific Volume (m ³ /MWt)	2.26	2.90	3.77
Surface Area (m ²)	3 155	3 570 ^(c)	4 809
Specific Area (m ² /MWt)	0.96	1.08	1.34
Type of Surface	All Metal	Metal and Concrete	Metal and Concrete

(a)The number of figures shown is for computational accuracy and does not imply accuracy to that many significant figures. (14)

(b)Estimates from dimensions given in the PSAR. (14)

(c)Includes area of exposed steel vessel (1,111 m²) and area of exposed concrete surface (chamber bottom, inside and outside of reactor pedestal, and 17 support columns, 2,459 m²).

0.5.1.1 Mark I Containment Design

The shape of the drywell of a Mark I containment is similar to that of a inverted light bulb (see Figure 0.5-1). For the Peach Bottom 2 or 3 power plant, which is selected for this analysis, the cylindrical portion of the drywell is 11.7 m in diameter by 16.0 m high and the spherical bottom is 20.4 m in diameter.

FIGURE 0.5-1. BWR Mark I Containment

The pressure-suppression chamber (wetwell) of a Mark I containment is toroidal in shape. For the Peach Bottom 2 or 3 power plant, the torus has a cross-sectional diameter of 9.5 m and a major diameter of 33.8 m. The wetwell is located just below and surrounds the drywell, as shown in Figure 0.5-1. The two chambers are connected by a series of downcomer vents, which extend radially from the lower portion of the drywell like the spokes of a wheel. The bottom portion of the wetwell is filled with water to form the suppression pool.

The reactor vessel, the reactor pedestal, and the sacrificial shield are located completely within the drywell, which together with the wetwell forms the Primary Containment. The Primary Containment is completely surrounded by the Reactor Building, which provides secondary containment.

Some 26 different BWR power plants in the United States are constructed with Mark I containment designs.

0.5.1.2 Mark II Containment Design

The Mark II containment followed the Mark I design and has been used in some 10 different BWR power plants in this country (including the reference plant).

The drywell for the Mark II design is in the shape of a truncated right circular cone with the larger diameter at the bottom (see Figure 0.5-2). It is made of steel plate, has a concrete floor, and contains the upper portion of the reactor pedestal, the reactor vessel, and the sacrificial shield.

FIGURE 0.5-2. BWR Mark II Containment

The suppression chamber is in the shape of a coffee mug with an ellipsoidal bottom and a right-circular cylindrical body. The body has the same diameter as and connects directly to the bottom end of the drywell. The two chambers are connected by downcomer vents. The wetwell is made of steel plate with the lower portion filled with a layer of concrete, which forms a structural base for and supports the reactor pedestal. The bottom of the wetwell is filled with water to form the pressure-suppression pool. In the case of the WNP-2 power plant, the wetwell is approximately 26 m in diameter and 20 m high. The drywell extends upward another 26 m.

Again, the Reactor Building, as secondary containment, completely surrounds the Primary Containment comprising the drywell and the wetwell.

0.5.1.3 Mark III Containment Design

The latest BWR containment design is termed the Mark III design (see Figure 0.5-3). This type of containment is incorporated in some 21 different BWR power plants either in operation or presently under construction or design in this country. The BWR power plant selected for this analysis is the Perry Nuclear Power Plant, ⁽¹⁴⁾ unit 1 or 2, which are rated at 3579 MWt each.

The Mark III containment design is quite different from the previous two designs in that no other building structure surrounds the containment. The steel containment vessel is cylindrical with an ellipsoidal top, as is the concrete shield which completely surrounds the vessel. The drywell and the wetwell form concentric rings surrounding the reactor vessel and bounded by the containment vessel. The lower portion of the drywell is lined with steel plate, whereas all other internal surfaces except those of the sacrificial shield are concrete.

The outer boundary of the lower portion of the drywell is the weir wall, which forms the inside boundary of the suppression pool. The outer boundary of the suppression pool is the inside face of the containment vessel, a 36.6-m-diameter steel cylinder of steel plate. The containment vessel is 46.4 m high and is capped by an ellipsoidal steel dome.

FIGURE 0.5-3. BWR Mark III Containment

The drywell is connected to the pressure-suppression pool by means of steel tubes, which pass through the wall that forms the inner boundary of the suppression pool. The connecting tubes are located below the level of the pool, as shown in Figure 0.5-3.

0.5.2 BWR Containment Dismantlement Cost Comparisons

The principal costs associated with the dismantlement of a BWR containment, excepting the removal of the reactor vessel and the sacrificial shield, are: 1) treatment and disposal of the contaminated water, 2) decontamination of wetwell steel surfaces, 3) removal and disposal of contaminated concrete surfaces, and 4) removal and disposal of contaminated piping. These are discussed and compared for the three BWR containment designs in the following subsections. A concluding subsection summarizes the results of the individual comparisons.

0.5.2.1 Water Treatment and Disposal

Treatment and disposal of the water from a pressure-suppression pool is proportional to the volume of that water. As shown previously in Table 0.5-1, the specific volumes of the water contained in the suppression pools of the three selected BWR power plants are all similar in magnitude. For the purposes of estimating dismantling costs, it can be assumed that the volumes of the pools, and thus the costs of treatment and disposal of the water, are roughly proportional to the power ratings of the different BWR power plants. There is, therefore, no marked difference in the costs for treatment and disposal of the water from the suppression pools in the different wetwell designs (Mark I, Mark II, or Mark III) for BWR power plants of the same power rating.

0.5.2.2 Decontamination of Wetwell Steel Surfaces

The Mark I wetwell consists of a thick-walled steel vessel. The upper portion of a Mark II wetwell is a thick-walled steel cylinder. The outer boundary of the Mark III wetwell selected for this analysis is also a thick-walled steel cylinder. All of these steel wall surfaces may be decontaminated by cleaning them with a high-pressure water jet. It is estimated that this type of cleaning can be accomplished at the rate of $46 \text{ m}^2/\text{hr}$. The water used for cleaning will become contaminated, but the volume of water required ($0.0305 \text{ m}^3/\text{m}^2$) is small, and it is mixed with the water from the suppression pool for treatment. In short, the cost of decontaminating steel surfaces is small, and in Appendix H the surface-cleaning cost is lumped with the cost of treating the water from the suppression pool.

The total cost of cleaning the steel walls is small. The differences between the costs of cleaning the steel walls of the wetwells of the Mark I, Mark II, and Mark III designs are even smaller. Thus, the effect of the wetwell steel wall area on the total dismantlement cost due to the differences in the containment designs is negligible and can safely be ignored without affecting the accuracy of the overall dismantlement cost estimate.

0.5.2.3 Removal and Disposal of Contaminated Concrete Surfaces

The WNP-2 Mark II wetwell is a steel vessel, which is lined in the bottom with reinforced concrete. The concrete liner is the in-vessel foundational

support for the reinforced concrete reactor pedestal and support columns, which, in turn, support the reinforced concrete drywell floor. It is assumed necessary to remove a 50-mm-thick layer from all concrete surfaces inside Primary Containment to effect satisfactory surface decontamination. The estimated total cost for removing and disposing the contaminated concrete surfaces inside Primary Containment (see Tables I.3-4 and I.3-7 in Appendix I) is \$157,720. This amount is about 0.5% of the total immediate dismantlement cost for WNP-2.

The Mark I design contains considerably less (~10%) concrete surface area than does WNP-2. Therefore, the cost for removal and disposal can be neglected in estimating the cost for dismantling a Mark I power plant. Moreover, since the reference plant is a Mark II design, the error, small though it is, is conservative.

As previously indicated, the Mark III design is quite different from the two earlier designs. The outer boundary of the suppression pool and wetwell is the steel containment vessel. The inner boundary is a circular reinforced concrete bulkhead (see Figure 0.5-3), which is probably coated with an epoxy paint rather than being lined with steel. In this case, the area of concrete surface (wetwell inner boundary, bulkhead inner surface, and weir wall outer surface) in the Mark III containment is the same as, or a little greater than, that for the Mark II containment. There is, therefore, essentially no difference in the costs for removal and disposal of contaminated concrete surfaces in the selected Mark II and Mark III designs.

It should be noted, however, that some other Mark III wetwells and suppression pools do not have steel-lined surfaces. In these cases, the amount of concrete surface area to be removed is greater for a Mark III design than for a Mark II design. In no case, however, is the additional estimated cost for a Mark III design as much as 1% of the total immediate dismantlement cost.

It is concluded in this instance that the differences in containment design have a negligible effect on the total dismantlement cost.

0.5.2.4 Removal and Disposal of Contaminated Piping

In this discussion, the only piping considered is the drywell-wetwell vent piping, since the process piping is assumed to be the same for all three

containment designs. The Mark I containment design incorporates a series of large-bore vent pipes connecting the drywell to the wetwell, as previously discussed and as shown in Figure 0.5-1. In addition, there is a header inside the wetwell as well as a series of downcomers that drop from the header to below the surface of the suppression pool. The connecting pipes, the header, and the downcomers must be removed and disposed of during dismantlement.

In the Mark II containment design, there are a number of downcomer vent pipes that drop from the drywell floor down to a point below the surface of the suppression pool. All of these pipes must be removed and disposed of during the dismantlement operation.

The Mark III design contains a number of horizontal vent pipes that connect the suppression pool to the annular space surrounding the weir wall. These vent pipes must also be removed and disposed of during the dismantlement operation.

From the above, it can be seen that a Mark I containment contains more piping than a Mark II, and a Mark II contains more piping than a Mark III, relatively speaking. These relationships are inversely proportional to the relationships for contaminated concrete surfaces. Thus, the costs for removing and disposing of the piping tend to counteract any variations in dismantlement cost resulting from the removal and disposal of the contaminated concrete surfaces.

0.5.2.5 Overall Effect of Containment Design on Dismantlement Cost

Comparison of the four dismantlement cost factors discussed previously shows that, for BWR plants of equivalent power rating, differences in containment design have very little effect on the total cost for dismantlement of a BWR (whether Mark I, Mark II, or Mark III).

The cost for suppression pool water treatment and disposal is essentially the same for the different containment designs.

The dismantlement cost differences attributable to cleaning the wetwell steel surfaces are so small as to be inconsequential when compared with the total dismantlement cost.

The differences in the costs of removing and disposing the contaminated concrete surfaces are small in relation to the overall dismantlement cost, as are the different costs of removing and disposing the contaminated vent piping in the containment system. In addition, the cost differentials for these two items tend to counterbalance one another, with the result that the combined effect on the total dismantlement cost is so small as to be negligible.

Considering all of the above factors, it is concluded that the effect of containment design (Mark I, Mark II, or Mark III) on the total cost of dismantling a BWR power plant is so small that it will not measurably affect the estimated cost of dismantling a BWR power plant having a containment design different from WNP-2.

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Technology, safety and cost information is given for the conceptual decommissioning of a large (1100MWe) boiling water reactor (BWR) power station. Three approaches to decommissioning, immediate dismantlement, safe storage with deferred dismantlement and entombment, were studied to obtain comparisons between costs, occupational radiation doses, potential dose to the public and other safety impacts. It also shows the sensitivity of decommissioning safety and costs to the power rating of a BWR in the range of 200 to 1100 MWe.

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