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

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MASTER

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FOREWORD

by

NUCLEAR REGULATORY COMMISSION STAFF

The NRC staff is in the process of reappraising its regulatory position relative to the decommissioning of nuclear facilities.(1) As part of this activity, NRC has initiated or will initiate several studies through technical assistance contracts. These contracts are being undertaken to develop specific background information to support the preparation of new standards covering decommissioning.

These studies will describe decommissioning alternatives and will evaluate the safety and costs associated with them. The plan is to cover all major types of nuclear facilities in the work conducted over the next several years. Separate reports will be prepared as the studies of the various facilities are completed.

Current plans include studies of decommissioning of light water reactors (LWRs) and their associated fuel cycle facilities by Battelle, Pacific Northwest Laboratories. In general, facilities of current design on typical sites are selected for the studies.

The first report in this series covered a fuel reprocessing plant.(2) The following report is the second of the series and deals with a pressurized water reactor. Additional topics will be reported on the tentative schedule as follows:

GFY 1978	Small Mixed Oxide Fabrication Plant
GFY 1979	Boiling Water Reactor Low Level Waste Burial Ground Uranium Mill

(1) Plan for Reevaluation of NRC Policy on Decommissioning of Nuclear Facilities. NUREG-0436, Office of Standards Development, U.S. Nuclear Regulatory Commission, March 1978.

(2) Technology, Safety and Costs of Decommissioning a Reference Nuclear Fuel Reprocessing Plant. NUREG-0278, Battelle, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, October 1977.

GFY 1980 Uranium Fabrication Plant
Uranium Hexafluoride Conversion Plant

The information provided in this report on the pressurized water reactor, including any comments, will be included in the record for consideration by the Commission in establishing criteria and new standards for decommissioning. Persons wishing to comment on this report should mail their comments to:

Chief
Fuel Process Systems Standards Branch
Division of Engineering Standards
Office of Standards Development
Washington, DC 20555

ABSTRACT

Safety and cost information was developed for the conceptual decommissioning of a large [1175 MW(e)] pressurized water reactor (PWR) power station. Two approaches to decommissioning, Immediate Dismantlement and Safe Storage With Deferred Dismantlement, were studied to obtain comparisons between costs, occupational radiation doses, potential radiation dose to the public, and other safety impacts.

Immediate Dismantlement was estimated to require about six years to complete, including two years of planning and preparation prior to final reactor shutdown, at a cost of $42 million, and accumulated occupational radiation dose, excluding transport operations, of about 1200 man-rem.

Preparations for Safe Storage were estimated to require about three years to complete, including 1-1/2 years for planning and preparation prior to final reactor shutdown, at a cost of $13 million and an accumulated occupational radiation dose of about 420 man-rem. The cost of continuing care during the Safe Storage period was estimated to be about $80 thousand annually. Accumulated occupational radiation dose during the Safe Storage period was estimated to range from about 10 man-rem for the first 10 years to about 14 man-rem after 30 years or more.

The cost of decommissioning by Safe Storage with Deferred Dismantlement was estimated to be slightly higher than Immediate Dismantlement. Cost reductions resulting from reduced volumes of radioactive material for disposal, due to the decay of the radioactive containments during the deferment period, are offset by the accumulated costs of surveillance and maintenance during the Safe Storage period. All costs are given in terms of constant 1978 dollars.

The decommissioning by permanent entombment of a PWR that had been operated for 20 to 30 years or more was found to be unsatisfactory because: 1) the radiation dose rates from the long-lived radionuclides ^{59}Ni and ^{94}Nb in the activated reactor vessel internals remain well above unrestricted release levels for a period of time far exceeding the known lifetime of any man-made structure, and 2) permanent entombment results in the proliferation of sites permanently committed to the containment of radioactive materials.

The principal incentive for deferring dismantlement comes from the reduction of radiation exposure that can be achieved. Compared with Immediate Dismantlement, deferral for 10 years reduced the estimated total radiation dose by about 40%; for 30 years, by more than 60%. Deferral of dismantlement beyond 30 years does not produce a significant further reduction in total radiation dose since most of the exposure is accumulated during the preparation for Safe Storage, rather than during Deferred Dismantlement.

The safety impacts of the decommissioning operations on the public were found to be small, compared with those of the operating power station. The principal impact on the public is the radiation dose resulting from the transport of radioactive materials to a disposal site.

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1.0 INTRODUCTION

This report presents the results of a study sponsored by the U.S. Nuclear Regulatory Commission (NRC) to conceptually decommission a present-generation pressurized water reactor (PWR) power station. The primary purpose of the study is to provide information on the available technology, the safety considerations, and the probable costs for the decommissioning of a large PWR power station at the end of its operating life. This information is intended for use as background data and bases in the modification of existing regulations and in the development of new regulations pertaining to decommissioning activities. It is also intended for use by utilities in their planning for the eventual decommissioning of their own nuclear power stations.

Decommissioning is defined, for a nuclear facility, as the measures taken at the end of the facility's operating life to assure the continued protection of the public from any residual radioactivity or other potential hazards present in the facility. Two basic approaches to decommissioning are considered:

- Immediate Dismantlement - Radioactive materials are removed and the station is disassembled and decontaminated during the four-year period following final cessation of power production operations. Upon completion, the property is released for unrestricted use.
- Safe Storage with Deferred Dismantlement - Radioactive materials and contaminated areas are secured and structures and equipment are maintained as necessary to assure the protection of the public from the residual radioactivity. During the period of Safe Storage, the facility remains limited to nuclear uses. Dismantlement is deferred until the radioactivity within the station has decayed to lower levels. Upon completion of dismantlement, the property is released for unrestricted use.

Deferred Dismantlement, as used here, is a generic term that includes whatever actions are required at some future time to accomplish termination of the facility's nuclear license and the release of the property for unrestricted use. These actions can range from radiation surveys that show that the residual radioactivity has decayed to releasable levels, to disassembly and removal of radioactive material.

A broad span of methods is possible under Safe Storage. These methods range from minimal removal and fixation of residual radioactivity with maintenance and surveillance using either custodial (layaway) or passive (mothballing) protection systems, to extensive cleanup and decontamination with hardened (temporary entombment) passive protection of highly radioactive areas using limited surveillance and continuing maintenance programs. Each method encompassed within Safe Storage requires some level of continuing care during the holding period which may vary in length from a few years to about a hundred years. Each method ends with the deferred dismantlement of the facility and the termination of the license for radioactive materials, thus permitting the unrestricted use of the property.

The intent of the Nuclear Regulatory Commission to minimize the number of sites permanently committed to the containment of radioactive material is satisfied by dismantlement, whether immediate or deferred.

The Portland General Electric Company's TROJAN NUCLEAR PLANT at Rainier, Oregon, was selected as the reference PWR power station for this study. TROJAN is a 1175 MW(e) station that utilizes a four-loop pressurized water reactor manufactured by the Westinghouse Electric Corporation in the nuclear steam supply system. The single-reactor station was assumed to be on a generic site, typical of reactor locations in the midwestern or middle southeastern United States. The structures, systems, and components are typical of the current generation of large PWR power stations.

A set of work plans was developed for the conceptual decommissioning of the reference PWR power station via Immediate Dismantlement. Additional sets of work plans were developed for two methods under Safe Storage with Deferred Dismantlement. The principles that guided the development of these plans were:

- to assure the safety of the public and the decommissioning workers in a cost-effective manner,
- To utilize demonstrated methods for decontamination and dismantlement.

From these work plans, estimates were developed for the manpower requirements, the major resource and equipment needs, the volumes of contaminated material packaged for disposal, the costs of accomplishing the work, and the exposure of the decommissioning workers and the public to radiation as a result of the decommissioning efforts. Because wide variations are possible within the work plans and in the decommissioning techniques that can be utilized to achieve the desired decommissioned condition, the results of the study are dependent upon the detailed choices made. The choices of plans and techniques made in this study are believed to be realistic and representative of the operations that would be required to decommission the reference PWR power station, and to provide an appropriate level of safety at a reasonable cost.

A methodology was developed for the determination of the level of radioactive contamination that could remain on a site or in a facility and still allow unrestricted use of the property. This methodology utilizes the calculated maximum annual dose to the maximum exposed individual as the basis for setting these levels. The relationship between dose and contamination level is a complex one involving the spectrum of radionuclides present and the pathways of these nuclides to the maximum exposed individual.

The work plans and postulated scenarios for airborne release of radioactive materials were used in evaluations of the potential impacts of decommissioning operations in the safety of the decommissioning workers and the public. Estimates were made of radiation exposure, lost-time injuries, and fatalities for each decommissioning approach studied. The reduction in potential radiation exposure to decommissioning workers and the public as a function of the length of the Safe Storage period was also estimated.

The operating techniques, safety impacts, and estimated costs developed in this study are sensitive to the specifics of the reference PWR power station. Such specifics include the mixture of radionuclides in and the levels of residual radioactive contamination, the location of the station relative to waste disposal sites, past operating practices, and the structural details of the station. These specifics must be examined carefully before attempting to apply the results of this study to a different station.

The study results are presented in two volumes. Volume 1 (Main Report) contains the results in summary form. Volume 2 (Appendices) contains the detailed analyses and data needed to support the results given in Volume 1. The supporting data are presented in a manner that facilitates their use for examining decommissioning actions other than those included in this study.

2.0 SUMMARY

The results of a study sponsored by the U.S. Nuclear Regulatory Commission (NRC) to conceptually decommission a large pressurized water reactor (PWR) power station are summarized in this section. The purpose of the study is to provide information on the available technology, the safety considerations, and the probable costs for decommissioning a large PWR power station after a 40-year operating life.

Decommissioning is defined, for a nuclear facility, as the measures taken at the end of the facility's operating life to assure the continued protection of the public from any residual radioactivity or other potential hazards present in the facility. Two basic approaches to decommissioning are considered:

- Immediate Dismantlement - Radioactive materials are removed and the station is decontaminated and disassembled during the four-year period following final cessation of power production operations. Upon completion, the property is released for unrestricted use.
- Safe Storage with Deferred Dismantlement - Radioactive materials and contaminated areas are secured and structures and equipment are maintained as necessary to assure the protection of the public from the residual radioactivity. During the period of Safe Storage, the facility remains limited to nuclear uses. Dismantlement is deferred until the radioactivity within the station has decayed to lower levels. Upon completion of dismantlement, the property is released for unrestricted use.

Deferred dismantlement, as used here, is a generic term that includes whatever actions are required at some future time to accomplish termination of the facility's nuclear license and the release of the property for unrestricted use. These actions can range from radiation surveys that show that the residual radioactivity has decayed to releasable levels, to disassembly and removal of radioactive material.

Immediate dismantlement is estimated to require about six years to complete, including two years of planning and preparation prior to final reactor shutdown, at a cost of $42 million, and accumulated occupational radiation dose, excluding transport operations, of about 1200 man-rem.

Preparations for Safe Storage are estimated to require about three years to complete, including 1-1/2 years for planning and preparation to prior final reactor shutdown, at a cost of $13 million and an accumulated occupational radiation dose of about 420 man-rem. The cost of continuing care during the Safe Storage period was estimated to be about $80 thousand annually. Accumulated occupational radiation dose during the Safe Storage period was estimated to range from about 10 man-rem for the first 10 years to about 14 man-rem after 30 years or more. All costs are given in terms of constant 1978 dollars.

The cost of decommissioning by Safe Storage with Deferred Dismantlement is estimated to be slightly higher than Immediate Dismantlement. Cost reductions resulting from reduced volumes of radioactive material for disposal, due to the decay of the radioactive contaminants during the deferment period, are offset by the accumulated costs of surveillance and maintenance during the Safe Storage period.

The decommissioning by permanent entombment of a PWR that had been operated for 20 to 30 years or more was found to be unsatisfactory because: 1) the radiation dose rates from the long-lived radionuclides ^{59}Ni and ^{94}Nb in the activated reactor vessel internals remain well above unrestricted release levels for a period of time far exceeding the known lifetime of any man-made structure, and 2) permanent entombment results in the proliferation of sites permanently committed to the containment of radioactive materials.

The principal incentive for deferring dismantlement comes from the reduction in radiation exposure that can be achieved. Compared with immediate dismantlement, deferral for ten years reduces the estimated total radiation dose by about 40%; for 30 years, by more than 60%. Deferral of dismantlement beyond 30 years does not produce a significant further reduction in total radiation dose since most of the dose is accumulated during the preparation for Safe Storage, rather than during Deferred Dismantlement.

The safety impacts of the decommissioning operations on the public were found to be small, compared with those of the operating power station. The principal impact on the public is the radiation dose resulting from the transport of radioactive materials to a disposal site.

2.1 KEY BASES AND ASSUMPTIONS

One of the key bases for this study is that the methods used to accomplish decommissioning utilize presently available technology. While a number of devices and techniques that are presently under development are discussed in the study, the results do not depend upon any breakthroughs or advances in present-day technology. Such advances would likely serve to reduce costs and occupational radiation exposure when fully developed and utilized.

The decommissioning effort is assumed to be carried out within the framework of existing regulations. No assumptions are made regarding what future regulatory requirements might be. It is recognized that future regulations could have significant impacts on the methods and results of this study. Efforts were made to follow the principle of minimizing exposures to radiation in developing the work sequences and for methods. The radiation dose rates used in the analyses are at the upper bound values that might be encountered, based on conservative estimates of the effectiveness of the chemical decontamination of the plant systems.

For Immediate Dismantlement, the decommissioning staff is assumed to be drawn largely from the operating personnel of the station, and is very familiar with the facility and its systems. Also, all craft labor assignments during decommissioning, except for demolition, are handled by plant maintenance mechanics who are qualified in all basic craft skills. This category of skilled worker is fairly common at operating reactor stations and eliminates the problems of craft jurisdiction frequently encountered on construction-type jobs.

The rate at which radiation levels diminish with time during the decommissioning efforts is assumed to be controlled by the half-life of ^{60}Co. The estimated radiation dose rates throughout the station are based on data measured at operating reactor stations during the first month of refueling and maintenance outages. Therefore, the radiation dose rates present during decommissioning operations that take place later than the initial month after reactor shutdown are reduced in proportion to the decay of ^{60}Co over that time interval.

The planning and preparations for decommissioning take place during the final two years of reactor operation, making it possible for the necessary approvals to be in place by the time the reactor is defueled and disabled, so that decommissioning work can commence without delay.

Decontamination of the dismantled facilities and the site is assumed to be performed as required to achieve levels of residual radioactivity sufficiently low to permit unrestricted use of the property.

The methods and procedures for decommissioning are selected to provide the required degree of radiation safety for the decommissioning worker and for the public, and are performed in a safety-conscious and cost-effective manner.

All materials, except spent reactor fuel, that are radioactively contaminated or are neutron-activated to levels above those permitting unrestricted use are packaged and shipped to a licensed burial site for disposal. These materials are assumed to be principally activated metals, activated metal corrosion products, and small quantities of fission products, with no significant quantities of transuranic materials. The spent reactor fuels are postulated to be shipped to an unspecified fuel repository.

The power station is assumed to contain a single reactor plant with no other nuclear facilities on the site. Thus, no support from shared facilities is assumed.

The results obtained in this study are specific to these bases and assumptions, and to other assumptions specifically stated throughout the report. Application of these results to situations where the conditions are different from those assumed in this study could produce erroneous conclusions.

2.2 REVIEW OF DECOMMISSIONING EXPERIENCE

A review of the documented cases of decommissioning of nuclear facilities shows that, while the facilities decommissioned were generally small and had operated for relatively short periods of time, the problems encountered tended to be common to all decommissioning undertakings. The review also shows that a wealth of experience exists within the nuclear industry regarding methods and equipment for accomplishing decommissioning, and that there are no major technical impediments to the successful decommissioning of large commercial power reactors.

2.3 STATUS OF REGULATORY GUIDANCE FOR DECOMMISSIONING

A review of existing regulations and guidelines shows that, in general, regulations are in place to cover the subject of decommissioning of a nuclear reactor power station. In many cases, the existing regulations do not speak specifically to the question of decommissioning but the regulations can readily be interpreted as being applicable. In these cases, modifications to the regulations to clearly define their applicability to decommissioning would be desirable.

Areas where more specific guidance could be helpful:

- Financial qualifications and responsibility for decommissioning, to more clearly define the commitments of the facility owner for achieving the final status of unrestricted use of the property. Specific definitions need to be established by the utility industry and its regulating agencies as to what are acceptable methods for providing funds for decommissioning.
- The advisability of burying the highly radioactive components from the reactor vessel in shallow land burial sites as permitted by current regulations is under consideration by the NRC, in light of the long-lived radionuclides and high levels of radioactivity present in some of those materials. Regulations may be needed that will define more clearly which materials can be disposed of and where they can be placed.
- Some centralization or at least a central indexing of all regulations pertaining to decommissioning in the Code of Federal Regulations would be very helpful.
- Existing guidance on what levels of residual radioactivity are permitted on materials, structures, and sites that can be released for unrestricted use tends to be some what fragmentary and does not have a common identifiable basis. Methodology is developed in this study that could

form that basis, predicated on a decision by regulatory agencies as to what constitutes an acceptable annual radiation dose to the maximally-exposed individual from such residual radioactivity in unrestricted use.

2.4 APPROACHES TO FINANCING DECOMMISSIONING

A recent NRC survey of state public utility commissions found that the preferred approach to providing funds for decommissioning was to treat the anticipated decommissioning costs as a negative salvage value for purposes of calculating depreciation on the nuclear power station. Several approaches were suggested for handling the monies so collected. These ranged from the establishment of a separate sinking fund with annual payments made from revenues, with the fund independent from and unavailable for use by the utility, to allowing the utility to invest the money in its own new facilities. In this latter case, the utility could then issue securities against those unencumbered facilities as the need for decommissioning funds arose, thus minimizing the overall cost to the electricity consumer of providing funds for decommissioning.

2.5 SITE AND FACILITY DESCRIPTION

The site used in these analyses is a generic one typical of a midwestern or southeastern river site, developed for use in a family of studies devoted to the decommissioning of nuclear fuel cycle facilities being performed for the NRC by Battelle-Northwest. The reactor used as the reference facility in this study is the Portland General Electric Company's TROJAN Nuclear Plant, a 1175 MW(e) station. The nuclear steam supply system is a four-loop pressurized water reactor manufactured by the Westinghouse Electric Company, and is generally representative of the current generation of large PWRs. Sufficient descriptive information is presented for the facility to permit the development of the detailed work plans, costs estimates and radiation dose estimates that are the results of this study.

2.6 CHARACTERIZATION OF THE RADIONUCLIDE INVENTORY

Levels of radioactivity and dose rates from activated reactor components, from contamination deposited throughout the plant, and from the site soil surface are calculated and/or derived from existing data. The radionuclides that are the principal contributors to external occupational radiation exposure are: immediately after reactor shutdown, ^{58}Co and ^{60}Co; during the four year period of Immediate Dismantlement and during Safe Storage, ^{60}Co; after 100 years of more, ^{59}Ni and ^{94}Nb. The amount of radioactivity present in the activated reactor components at the time of reactor shutdown is calculated to be nearly 5 million curies. The calculated radiation dose rates of ^{60}Co from the activated reactor components ranged from a maximum at the core shroud of 300,000 to 500,000 R/hr to 2 to 5 R/hr at the reactor pressure vessel wall. The calculated radiation dose rates from ^{59}Ni and ^{94}Nb have maximum values in the core shroud of about 100 mR/hr and 2 R/hr, respectively. Dose rates at locations throughout the facility range from 100 to 200 R/hr on ion exchange resins and 30 to 50 R/hr on the steam generators to a few mR/hr in many areas, based on a composite of data from operating plants.

Annual atmospheric releases from operating PWRs vary widely, depending on such factors as the plant operating conditions, the design of the plant gaseous effluent clean-up systems, and the plant size. For this study, the ground contamination levels and mixtures of radionuclides on the site resulting from deposition of atmospheric releases from the plant during 40 years of normal operation are calculated and compared using two data bases (generic annual release information and measured annual release information). The variation in the calculated contamination levels on the site surface that results from using the different sets of radioactive gaseous release data is illustrated.

2.7 ACCEPTABLE CONTAMINATION LEVELS FOR UNRESTRICTED USE OF THE DECOMMISSIONED REFERENCE PWR

A methodology for determining acceptable residual radioactive contamination levels for unrestricted use of the decommissioned reference PWR facility and/or site is presented and example acceptable contamination levels are calculated in this study. The methodology is based on the concept that no member of the public will be allowed to receive an annual dose in excess of a limit yet to be established by U.S. regulatory agencies. These acceptable contamination levels, or disposition criteria, are based on an assumed range of 1 to 25 millirem per year. The effect of radioactive decay upon the acceptable levels of residual radionuclides both in the facility and on the site is demonstrated by calculating these criteria for the radionuclide mixture present at reactor shutdown and for radioactive decay times of 10, 30, 50, and 100 years.

For the facility, surface radioactive contamination measurements are used in determining whether unrestricted use can be permitted, thus units of surface contamination are presented for the acceptable release levels. Surface contamination values are converted into units of radioactivity per gram of soil sample by assuming mixing of the radiation source with dry soil to depths of 1 and 15 cm. After 40 years of normal PWR operation, the residual radioactive contamination is assumed to be mixed to a depth of 1 cm from natural processes. When the site is released, the residual radioactive contamination is assumed to be mixed to a depth of 15 cm as farming activities begin.

A summary of the calculated radioactive contamination levels that result in an annual dose of one millirem to any organ of any individual calculated in this study is given in Table 2.7-1. These levels are used in determining the extent of decontamination required to decommission the reference PWR by Immediate Dismantlement and by Safe Storage with Deferred Dismantlement.

TABLE 2.7-1. Summary of Example Disposition Criteria for the Reference PWR and the Reference Site

	Time After Shutdown (Years)	Acceptable Residual Contamination Levels for an Annual Dose Limit of 1 mrem per Year		
		Facility	Soil Contamination[b]	
		Surface Contamination ($\mu Ci/m^2$)	Mixed to 1 cm (pCi/g)	Mixed to 15 cm (pCi/g)
PWR Facility[a]	0	2.3×10^{-1}	--	--
	100	3.2×10^{-1}	--	--
Site (GESMO)	0	1.4×10^{-2}	9.4×10^{-1}	6.2×10^{-2}
	100	1.1×10^{-2}	7.4×10^{-1}	4.9×10^{-2}
Site (NUREG-0218)	0	1.1×10^{-2}	7.4×10^{-1}	4.9×10^{-2}
	100	6.6×10^{-3}	4.4×10^{-1}	2.9×10^{-2}

(a) In the facility, surface contamination levels are assumed to be used to determine the necessary decommissioning procedures. All wastes generated during the decommissioning procedures are assumed to go to a nuclear waste burial site for disposal.

(b) At plant shutdown, assuming no mechanical mixing in the soil, the radiation source is assumed to be in the top 1 cm of surface. After decommissioning, plowing for farming mixes the radiation source to a depth of 15 cm.

2.8 RADIATION EXPOSURE ESTIMATES

Estimates of accumulated occupational radiation dose range from over 1200 man-rem for Immediate Dismantlement to over 400 man-rem for placing the facility in Safe Storage, with an additional 10 to 14 man-rem for surveillance and maintenance during postulated periods of continuing care that range in length from 10 to 100 years. Radiation dose associated with Deferred Dismantlement depends upon when the dismantlement takes place.
Relatively little reduction in accumulated occupational radiation dose is estimated to result from deferment of the decommissioning sequence beyond 30 years, and virtually no reduction results from deferments beyond 50 years.

The individual estimates of occupational radiation dose for the various phases of decommissioning are summarized in Table 2.8-1.

TABLE 2.8-1. Summary of the Estimated External Occupational Radiation Doses for Decommissioning the Reference PWR

Decommissioning Mode	Time After Reactor Shutdown (Years)	Estimated Dose (Man-rem)[a]
Immediate Dismantlement	0	1200
Safe Storage [b]		
Preparations for Safe Storage	0	420
Continuing Care	10	10
	30	14
	50	14
	100	14
Deferred Dismantlement	10	330
	30	24
	50	2
	100	1
Total for Safe Storage[b] with Deferred Dismantlement in year	10	760
	30	460
	50	440
	100	430

[a] Estimates of man-rem of radiation dose have been rounded to two significant figures

[b] Safe Storage consists of three phases Preparations for Safe Storage, Continuing Care, and Deferred Dismantlement

Additional radiation dose is received by truck drivers, garagemen, trainmen, onlookers and the general public as a result of transporting the spent fuel and the radioactive materials to disposal sites. These radiation doses are summarized in Table 2.8-2.

TABLE 2.8-2. Radiation from Transport of Radioactive Materials from Decommissioning

	Radiation Doses from Transport (man-rem)[a] Immediate Dismantlement	Preparations for Safe Storage
Occupational:		
Truck Transport	99	10
Rail Transport	3.5	3.5
TOTAL	100	14
Public:		
Truck Transport	21	2.1
Rail Transport	1	1
TOTAL	22	3

(a) All values are rounded to 2 significant figures.

2.9 DECOMMISSIONING COSTS

All costs are given in terms of 1978 dollars and 25% contingency is included in the values presented.

Immediate Dismantlement is estimated to cost just over $42 million. The major contributors to the total are summarized in Table 2.9-1. The cost for shipment and disposal of radioactive materials, including transportation only for spent fuel, is about 33% of the total decommissioning cost. About 27% of the total decommissioning cost is due to staff labor, not including contractor and demolition labor. Demolition of the decontaminated structures is estimated to be about 19% of the total decommissioning costs. Since demolition of the decontaminated structures is not required by NRC regulations, the total decommissioning cost could be reduced by 19% by not demolishing the structures.

TABLE 2.9-1. Estimated Cost for Immediate Dismantlement

Cost Item	$ Million[a]	Percent of Total
Fuel Shipment	3 084	7 3
Equipment	1 028	2 4
Supplies	1 949	4 6
Power	4 375	10 4
Activated Materials	3 418	8 1
Contaminated Material	6 479	15 4
Radioactive Waste	0 866	2 0
Staff Labor	11 233	26 7
Contractor Services	0 680	1 6
Demolition Services	8 012	19 0
Nuclear Insurance	1 000	2 4
TOTAL (rounded)	42 1	

[a] Number of figures shown is for computational accuracy and does not imply precision to the nearest one thousand dollars

The preparations for Safe Storage are estimated to cost just under $13 million. The major contributors to the total are summarized in Table 2.9-2. Shipment and disposal of radioactive materials, including transportation only for spent fuel, account for about 30% of the total preparations cost. Staff labor contributes about 36% of the total cost, with contractor services making up another 3%.

TABLE 2.9-2. Estimated Cost of Preparations for Safe Storage

Cost Items	$ Million[a]	Percent of Total
Fuel Shipment	3.084	24.4
Equipment	0.094	0.7
Supplies	1.114	8.8
Power	2 331	18.5
Radioactive Waste	0.680	5.4
Staff Labor	4.564	36.2
Contractor Services	0.381	3.0
Nuclear Insurance	0.368	2.9
TOTAL (rounded)	12.6	

[a] Number of figures shown is for computational accuracy and does not imply precision to the nearest one thousand dollars.

The cost of continuing care during the period of Safe Storage is estimated to be $80,000 per year.

The cost of Deferred Dismantlement, starting after intervals of 10, 30, 50 and 100 years after final reactor shutdown has been estimated in constant 1978 dollars to be $37 million, $37 million, $31 million and $30 million, respectively. The lesser costs after the longer intervals are the result of having less contaminated material for packaging, shipment and burial due to decay of the radionuclides.

The total cost of Safe Storage with partial dismantlement and eventual Deferred Dismantlement is estimated to be essentially the same as for Safe Storage without partial dismantlement. The principal difference is in the time distribution of expenditures.

The total cost in constant dollars for each of the decommissioning choices is summarized in Table 2.9-3.

TABLE 2.9-3. Total Estimated Costs for Possible Decommissioning Choices

Decommissioning Mode	Decommissioning Costs ($ millions)[a][b] Number of Years After Reactor Shutdown Dismantlement is Deferred				
	0	10	30	50	100
Immediate Dismantlement	42.1	--	--	--	--
Preparations for Safe Storage	--	12.6	12.6	12.6	12.6
Continuing Care	--	0.6	2.2	3.7	7.8
Deferred Dismantlement	--	37.0	37.0	30.5[c]	30.4[c]
Total Decommissioning Cost	42.1	50.2	51.8	46.8	50.8

(a) Values include a 25% contingency.

(b) Values are in constant 1978 dollars.

(c) These reduced values result from lesser amounts of contaminated materials for burial in a licensed disposal site.

2.10 OCCUPATIONAL AND PUBLIC SAFETY

Radiological and nonradiological safety impacts from normal decommissioning operations and potential accidents are identified and evaluated for Immediate Dismantlement and Safe Storage decommissioning modes for the reference PWR. The safety evaluation includes consideration of radiation dose to the public from normal operations and postulated accidents, occupational radiation exposure, industrial-type accidents and potential chemical pollutants. The safety evaluation utilizes current data and methodology, along with engineering judgment when necessary, to estimate the required input information and the resulting safety impacts. The approach used to evaluate all the safety aspects of a particular decommissioning activity is believed to be conservative.

The results of the safety evaluation of normal decommissioning operations are summarized in Table 2.10-1. The principal radiation dose to the public

TABLE 2.10-1. Summary of Safety Analysis for Decommissioning the Reference PWR

Type of Safety Concern	Source of Safety Concern	Units	Immediate Dismantlement	Safe Storage with Deferred Dismantlement After 10 Years	30 Years	50 Years	100 Years
Public Safety[a]							
Radiation Exposure	Decommissioning Operations	man-rem	0 0001	<0 0001	<0 0001	<0 0001	<0 0001
	Transportation	man-rem	22	(c)	(c)	(c)	(c)
	Safe Storage	man-rem	--	neg[b]	neg[b]	neg[b]	neg[b]
Occupational Safety							
Serious Lost-time Injuries	Decommissioning Operations	total no	4 0	4 9	4 9	4 9	4 9
	Transportation	total no	1 1	1 2	1 2	1 2	1 2
	Safe Storage	total no	--	0 96	1 2	1 4	1 9
Fatalities	Decommissioning Operations	total no	0 029	0 029	0 029	0 029	0 029
	Transportation	total no	0 068	0 075	0 075	0 075	0 075
	Safe Storage	total no	--	0 00087	0 0026	0 0045	0 0087
Radiation Exposure	Decommissioning Operations	man-rem	1200	760	460	440	430
	Transportation	man-rem	100	(c)	(c)	(c)	(c)
	Safe Storage	man-rem	--	10	14	14	14

(a) Radiation doses from postulated accidents are not included

(b) neg = negligible Radiation doses to the public from normal continuing care activities were not analyzed in detail, but are expected to be significantly smaller than those from decommissioning operations

(c) Not estimated

materials from the reactor station to disposal facilities. The estimated dose to the public resulting from decommissioning operations and from Safe Storage is extremely small.

Less than 5 lost-time injuries from industrial-type accidents are predicted to occur during the decommissioning effort, with one additional injury predicted to result from transportation operations. Essentially no fatalities are predicted to occur as a result of decommissioning operations, including transportation.

2.11 CONCLUSIONS AND RECOMMENDATIONS

Decommissioning of a large nuclear reactor power station is technically feasible with present-day technology. Further development of special equipment such as the plasma torch and the arc saw could lead to reductions in cost and occupational radiation exposure.

Existing regulations appear to cover decommissioning. However, some modifications and/or additions that speak specifically to the requirements for decommissioning would be helpful. Centralization or a central indexing of regulations that apply to decommissioning would also be helpful.

The estimated occupational radiation dose resulting from decommissioning is at most roughly equivalent to the dose resulting from about three typical refueling and maintenance outages, and thus does not appear to be prohibitively large. The impact of decommissioning on the safety of the public is vanishingly small, with no significant risk to the public identified.

In terms of constant dollars Immediate Dismantlement is the least expensive choice for decommissioning. While there is incentive to defer dismantlement due to the reduction in occupational radiation dose that can be achieved, the costs of surveillance and maintenance during Safe Storage increase the total cost linearly with time. On the other hand, a present value analysis of decommissioning costs indicates an incentive to defer dismantlement for as long as possible, providing the discount rate always exceeds the inflation rate. In practice, the choice will probably be made based on a detailed analysis of which approach is most financially advantageous to the station owner.

The acceptability of disposal of highly radioactive and/or long-lived materials by burial in a shallow land burial facility is under consideration by NRC and needs to be determined. If placement of these materials in a deep geologic disposal facility similar to that postulated for high-level radioactive wastes is required in the future, decommissioning costs will be increased by nearly $2 million.

If the bulk of the non-activated, contaminated stainless steel can be decontaminated to levels sufficiently low to permit unrestricted reuse of that material, a savings of about $1-1/4 million can be realized. However, the appropriate definitions of levels of radioactivity that would be permitted on such materials when released for unrestricted use are not presently available.

Certain types of data that are useful in decommissioning analyses are essentially non-existent at this time. Some measurements on activated stainless steel that has been irradiated for an extended period of time (>10 years) to determine the growth of such long-lived radionuclides as ^{59}Ni and ^{94}Nb would be valuable for confirmation of calculations. Similarly, measurements of the growth of radionuclides in the biological shield concrete would be helpful in evaluating the radiation dose rates that might be encountered from the activated shield. In particular, the levels of ^{152}Eu and ^{154}Eu resulting from trace amounts of europium present in the concrete are important contributions to the total radiation dose rate from the concrete. In addition, studies to determine the actual levels of radioactivity on the soil surfaces surrounding operating reactor facilities would help to characterize in a realistic manner the residual radioactivity that might be present after 40 years of operation, and would help to quantify the decontamination effort that might be required to release the site for unrestricted use.

Careful attention to simplifying the problems of remote maintenance and eventual dismantlement during the design phase of a reactor project would be effective in reducing costs and in reducing occupational radiation exposure during maintenance operations as well as during decommissioning.

3.0 REVIEW OF DECOMMISSIONING EXPERIENCE

This section contains a review of the available experience in the decommissioning of nuclear facilities. Because of differences in reactor size, type and design, operating time, licensing requirements, location, motive for installation of the facility, and conditions of concern (e.g., costs, shutdown radiation levels, amounts of radioactive waste) extrapolations from these experiences to large commercial reactors are considered to be generally unreasonable. Many of the power reactors that have been decommissioned were involved in the U.S. AEC power demonstration program and were operated only for limited periods of time. The primary value of past decommissioning experience is that it identifies the individual measures required for decommissioning. The summation of the experience provides a reasonable assessment of available technology, safety considerations, and probable costs associated with decommissioning a present-generation PWR power station.

For example, during the total dismantlement of the Elk River Reactor[1] in Minnesota, a remotely manipulated plasma arc torch was utilized for the underwater segmenting of the 80 mm thick walls of the carbon steel reactor pressure vessel. The cost of the design, development, and testing required to produce this useable, prototypical cutting unit was over one-half million dollars. A review of this technology reveals that further plasma arc development will be needed before this technique can be used on the thicker ($\sim$ 230 mm) carbon steel walls of a present-day, large PWR. This is important information for the PWR power station owner who is planning, as part of his decommissioning program, to segment the 230 mm walls of his reactor pressure vessel by remote means using a plasma arc torch.

Power demonstration reactors, military reactors, and experimental/research reactors have been decommissioned safely using a variety of decommissioning approaches, without undue risk to personnel or to the environment. It is our conclusion that similar techniques can be safely and successfully applied to a large [1175 MW(e)] commercial PWR as described in this report.

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[1] Final Elk River Reactor Program Report. COO-651-93 Revised, p. C-7 United Power Association, Elk River, MT, November 1974.

3.1 NUCLEAR REACTOR DECOMMISSIONING HISTORY

The decommissioning of nuclear reactor facilities is a relatively well-developed technology. In the United States, the term "decommissioning" means to retire from active service, without the option of resuming the original operations. Historically, decommissioning of nuclear facilities did not necessarily result in a terminal condition. In fact, the Safe Storage (mothballing, layaway, and entombment) approaches that have been used should be recognized as only one stage in the decommissioning process. Current NRC philosophy indicates a decommissioning approach that ends in the termination of the facility's nuclear license and the release of the property for unrestricted use within a finite period of time.

Past decommissionings of licensed reactor facilities have been accomplished by Safe Storage (mothballing or entombment), and by dismantlement or by a combination of these alternatives. To date, alternative selection has been based primarily on cost. Whichever approach to decommissioning was selected, it provided for the protection of the safety of the public and for minimal adverse impacts on the environment. Small research reactors have been decommissioned primarily by dismantlement. Licensed nuclear power plants were usually placed in Safe Storage (mothballed) in which they continue to retain a reactor "possession-only" [a] license and the associated licensee responsibilities.

Lear and Erickson[2] report that between 1960 and mid-1976, a total of 65 licensed nuclear reactors had been or were in the process of being

[a] Title 10 CFR Part 50 § 50.59, "Authorization of Changes, Tests and Experiments" and § 50.90, "Application for Amendment of License or Construction Permit" provides the rules by which a licensee may amend his license to attain a possession-only status. Once this possession only license is issued, reactor operation is not permitted.

[2] G. Lear and P. B. Erickson, "Decommissioning and Decontamination of Licensed Reactor Facilities and Demonstration Nuclear Power Plants." Proceedings of the First Conference on Decontamination and Decommissioning (D&D) of ERDA Facilities, CONF-750827 pp. 31-45, Idaho Falls, ID, August 1975.

decommissioned. Of these, 5 were nuclear power plants, 4 were demonstration nuclear power plants, 6 were licensed test reactors, 28 were research reactors, and 22 were critical facilities. Of the fifty licensed research reactors and/or critical facilities decommissioned or scheduled to be decommissioned, all but four had been or will be totally dismantled with the licenses terminated. The remaining four will retain a possession-only type license for an indefinite period in Safe Storage.

Three of four government-owned nuclear power demonstration plants (BONUS, HALLAM, and PIQUA) have been entombed as a decommissioning alternative.[3,4,5] The fourth demonstration plant, Elk River Reactor, remains the largest project to date that has been completely dismantled and removed from its site.[6]

Two useful references concerning past decommissioning activities are: The Proceedings of the First Conference on the Decontamination and Decommissioning of ERDA Facilities, August 1975 (Reference 7), and A Preliminary Study of the Decommissioning of Nuclear Reactor Installations, July 1977 (Reference 8). A tabulation of nuclear reactor decommissionings is presented in Table 3-1. Descriptions of some of the more significant reactor decommissionings that follow are excerpted, in part, from Reference 8.

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(3) Boiling Nuclear Superheater Power Station Decommissioning Final Report. WRA-B-70-500, Docket-1154-2, Puerto Rico Water Resources Authority and United Nuclear Corporation, September 1, 1970.

(4) Retirement of Hallam Nuclear Power Facility. AI-AEC-12709, Atomics International, Canoga Park, CA, May 1970.

(5) C. W. Wheelock, Retirement of the Piqua Nuclear Power Facility. AI-AEC-12832, Atomics International, Canoga Park, CA, April 1970.

(6) Final Elk River Reactor Program Report. COO-651-93, Revised, United Power Association, Elk River, MT, November 1974.

(7) Proceedings of the First Conference On Decontamination and Decommissioning (D&D) of ERDA Facilities, CONF-750827, Idaho Falls, ID, August 19-21, 1975.

(8) A. Martin, et al., A Preliminary Study of the Decommissioning of Nuclear Reactor Installations, ANS Report No. 155, July 1977. Associated Nuclear Services, 123 High Street, Epsom, Surrey, KT19 8EB.

TABLE 3-1. A Digest of Nuclear Reactor Decommissionings[a]

Facility Name and Location	Reactor Type	Category[b]	Power Rating[c]	Type of Decommissioning	License Status	Monitoring System	Safe Storage Measures	Year Decommissioned	Decommissioning Cost	Miscellaneous Information
HRE-1 (Homogeneous Reactor Experiment) HNL	Fluid-fuel	TR	1 MW	Dismantled	—	—	—	1954	—	—
HRE-2 (Homogeneous Reactor Experiment) HNL	Fluid-fuel	TR	<1 MW	Dismantled	—	—	—	1954	—	—
ARE (Aircraft Reactor Experiment) HNL	Fluid-fuel	TR	1 MW	Dismantled	—	—	—	1955	—	—
PM-2A (Portable Medium Power Plant) Greenland	Pressurized water type light water moderated and cooled	M	10 MW(t)	Dismantled	—	—	—	1964	—	—
HPR's (Hanford Production Reactors, 8 total) Richland, WA	Graphite moderated, water cooled	PR	—	Safe Storage 4-Standby 4-Retired (Layaway)	—	Continuous surveillance	Continuous maintenance	1965 1971	—	—
CVTR Carolina-Virginia Tube Reactor) Parr, SC	Pressure tube heavy water (D_2O) cooled and moderated	LPR	65 MW(t)	Safe Storage (mothballed)	Byproduct per 10 CFR 30	Periodic surveillance	Welded closure locked doors security fence	1968[e]	—	—
Hallam Hallam Neb	Graphite moderated sodium cooled	DPR	256 MW(t)	Entombed	Operating authorization terminated	Not required	Welded closure concrete cover weatherproofed	1969	—	Decommissioning took 3 years
PNPF (Piqua Nuclear Power Facility) Piqua, Ohio	Organic cooled and moderated	DPR	45 MW(t)	Entombed	Operating authorization terminated	Not required	Welded closure concrete cover waterproofed	1969	—	Decommissioning took 3 years
BONUS (Boiling Nuclear Superheater Power Station) Ricon, Puerto Rico	BWR with nuclear superheating	DPR	50 MW(t)	Entombed	Operating authorization terminated	Not required	Welded closure concrete cover locked doors security fence	1970	—	—
Walter Reed Research Reactor Washington DC	Atomics International Model L-54, homogeneous-fueled	MTR	50 kW	Dismantled	—			1971	—	—
Pathfinder Sioux Falls SD	BWR nuclear super heat	LPR	190 MW(t)	Safe Storage (mothballed) with steam plant conversion	Two part license 10 CFR 50 possession only and byproduct 10 CFR 30	Continuous security force[f]	Welded closure security fence	1972	\$3.7M	—
EBR-1 (Experimental Fast Breeder Reactor) NRTS - Idaho	Liquid metal cooled	TR	—	Deactivated decontaminated converted for public access	—	Public access via National Park Service	—	1973	\$775,000 (cost to convert for public access only)	Dedicated a National Monument in 1966
Saxton Reactor Facility Saxton, PA	PWR	LTR	23.5 MW(t)	Safe Storage (mothballed)	Possession only	Intrusion alarms	Welded closure locked doors security fence	1973	\$2.5M	Owned operated and decommissioned by SNEC (Saxton Nuclear Experiment Corporation)
SEFOR (Southwest Experimental Fast Oxide Reactor) Strickler, AR	Sodium cooled fast	LTR	20 MW(t)	Safe Storage (mothballed)	Byproduct to state	Intrusion alarms	Welded closure locked doors security fence	1973		—
Elk River Reactor, Elk River, Minn	BWR, fossil fuel superheating	DPR	58 MW(t)	Dismantled and partial conversion	Operating authorization terminated	Not required	Not required	1974	\$6.15M	Decommissioning took 3 years
ASTR (Aerospace Test Reactor) U.S. Air Force NARF Ft Worth TX	—	M	10 MW	Dismantled	—	—		1974		—
GTR (Ground Test Reactor, U.S. Air Force - NARF Ft Worth, TX	—	M	10 MW	Dismantled	—	—	—	1974	—	—
RTA Reactivity Test Assembly) U.S. Air Force NARF Ft Worth, TX	—	M	1 MW	Dismantled			-	1974		—

For explanation of notes see the bottom of the next page

TABLE 3.1. (Continued)

Facility Name and Location	Reactor Type	Category(b)	Power Rating(c)	Type of Decommissioning	Licence Status	Monitoring System	Safe Storage Measures	Year Decommissioned	Decommissioning Cost	Miscellaneous Information
FERMI 1 Monroe Co Mich	Sodium cooled fast	LPR	200 $MW_{(t)}$	Safe Storage (mothballed)	Possession only(g)	Continuous security force(f)	Locked doors security fence	1975	$6 95M	—
PM-3A (Portable Medium Power Plant) McMurdo Station Antarctica	Pressurized Water Type light water moderated and cooled	M	9 4 $MW_{(t)}$	Dismantled	—	—	—	1975	—	—
HTR (Hanford Test Reactor) Richland, WA	Graphite moderated	TR	Zero Power	Dismantled	—	—	—	1977	$0 18M	—
B & W Lynchburg, VA	Pool	LTR	6 $MW_{(t)}$	Partially Dismantled	Byproduct per 10 CFR 30	Not required	Not required	—	—	—
GE EVESR Alameda Co, CA	BWR with nuclear superheat	LTR	17 $MW_{(t)}$	Safe Storage (mothballed)	Possession only	Continuous security force	Locked doors security fence	—	—	—
NASA Plumbrook Sandusky, Ohio	Light water	LTR	0 1 $MW_{(t)}$	Safe Storage (mothballed)	Possession only	Continuous security force(f)	Locked doors security fence	—	—	—
Peach Bottom 1 York Co , Penn	Gas cooled graphite moderated	LPR	115 $MW_{(t)}$	Safe Storage (mothballed)	Possession only	Continuous security force(f)	Not yet established	—	—	—
VBWR (Vallecitos Boiling Water Reactor Alameda Co CA	BWR	LPR	50 $MW_{(t)}$	Safe Storage (mothballed) with steam plant conversion	Possession only	Continuous security force(f)	Locked doors security fence	—		—
Westinghouse Test Reactor Waltz Mill PA	Tank	LTR	60 $MW_{(t)}$	Safe Storage (mothballed)	Possession only	Continuous security force(f)	Locked doors security fence	—	—	—
SRE (Sodium Reactor Experiment), AI Santa Susana CA	Graphite moderated sodium cooled	PP	20 $MW_{(t)}$ modified 1964 to 30 $MW_{(t)}$	Safe Storage (mothballed - 1967) dismantling started (1974)	—	—	—	In progress(h)	—	—
IRL (Industrial Reactor Laboratories Inc , Research Reactor) Plainsboro NJ	Pool	TR	5 $MW_{(t)}$	Partially dismantled	—	—	Unrestricted use	1977	Less than $1M	Decommissioned by NL Industries Incorporated Decommissioning took ~ two years(i)

NOTES:

(a) Blank spaces in the tabular columns indicate author was unable to locate the information from the literature studied

(b) Categories of Reactors

- PP = Power Production
- PR = Production Reactor (AEC)
- LPR = Licensed Power Reactor
- LTR = Licensed Test Reactor
- DPP = Demonstration Power Plant
- M = Military
- TR = Test Reactor (Experimental/Research)

(c) Power ratings are given in electrical (e) or thermal (t) megawatts (MW) or kilowatts (kW)

(d) Byproduct Licenses may be either "Byproduct NRC issued in accordance with 10 CFR Part 30 or Byproduct License issued by an agreement state in accordance with authority granted by 10 CFR Part 150

(e) First to be placed in safe storage (mothballed) provided significant experience in developing criteria and methods

(f) Not required for NRC due to other onsite security force availability unrelated to the decommissioning activities If the security force had not been available, the NRC may have required other control measures (e g , manned security or access control)

(g) Title 10 CFR Part 50, § 50 59, ' Authorization of Changes, Tests and Experiments and § 50 90 'Application for Amendment of License or Construction Permit provides the rules by which a licensee may amend his license to obtain a possession-only status Once this possession-only license is issued, reactor operation is not permitted

(h) The SRE facility is only 1 of 8 complete decontamination and dismantling programs at the AI Santa Susana, CA , DOE-owned site as part of the AI Decontamination and Disposition of Facilities Program

(i) The site is the first for a decommissioned commercial reactor to be approved by the government for unrestricted use

ACRONYMS:

AI	Atomics International
BWR	Boiling Water Reactor
DOE	Department of Energy formerly Energy Research and Development (ERDA)
HNL	Holifield National Laboratory formerly Oak Ridge National Laboratory (ORNL)
HTGR	High Temperature Gas Cooled Reactor
LASL	Los Alamos Scientific Laboratory
NARF	Nuclear Aerospace Research Facility
NASA	National Aeronautics and Space Administration
NRTS	National Reactor Testing Station
SRL	Savannah River Laboratory

3.1.1 Carolina Virginia Tube Reactor (CVTR), Parr, South Carolina

The CVTR was a 65 MWt heavy water cooled and moderated pressure tube reactor. The decision to decommission the plant was taken in 1967 after 4 years' experimental operation. The plan adopted was to deactivate the reactor, by the Safe Storage mode, surrender the facility license, and to use the Containment Building and Reactor Building for long term storage of remaining radioactive materials under a by-product license.

All fuel and heavy water were shipped off site. The facility license was amended from operation to possession-only status and an authorization obtained from the AEC to dismantle the facility. The facility license was replaced by the by-product license on completion of the decommissioning and final AEC inspection. Remaining radioactive materials were stored, where possible, in their normal operating position. The control rod drive system was deactivated. Voids containing radioactive materials were sealed, valves were secured and access hatches to the Containment Building were bolted shut so that special equipment was required to open them. A double security barrier was placed around all areas containing radioactive material.

The decision to decommission the reactor with minimum dismantling and removal of radioactive materials meant substantial cost-savings and minimum radiation exposure to plant personnel during the operation. Decommissioning of the CVTR is described in Reference 9.

3.1.2 Hallam Nuclear Power Facility, Hallam, Nebraska

The Hallam Facility was located at the Sheldon Station of the Consumers' Public Power District, first became operational in 1963, and was shut down following moderator element problems. Owing to the cost of repair, doubtful value of continued operations and a shift of emphasis, the AEC ordered retirement of the reactor in 1966. The reactor was designed to produce 256 MWt and was sodium cooled with a graphite moderator. Physical activities involved in carrying out the reactor retirement by entombment were completed in September 1969.

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(9) W. Willoughby and H. T. Babb, "CVTR," Nuclear News. 13: 48, June 1970.

All fuel and bulk sodium were removed from the site. Residual sodium was rendered inert and all radioactive residues were disposed of by removal to a federal repository. Radioactive components and materials remaining on site were sealed in the underground vaults of the remaining structure. Two 0.5 in thick steel plates were welded over the reactor vessel and reactor area. All penetrations were weld sealed and the whole covered with layers of tar, earth, and plastic film. Heat exchangers were dismantled and removed. System components were removed and the reactor and vaults were sealed by a construction contractor under the direction of the utility's Health and Safety Administration.

No special techniques or equipment were reported to have been developed for this operation. Residual sodium was rendered passive by purging with a gaseous nitrogen-steam mixture. Normal operational procedures were used for the removal of all radioactive materials.

A total of 300,000 Ci of radioactivity, mainly associated with the reactor vessels and internals, was sealed in the reactor and underground vaults. The bulk sodium removed from the primary circuit was slightly radioactive (7 Ci in the 250,000 kg shipment). A special sodium cleaning facility was erected for the decontamination of system components. The site is periodically inspected by the State of Nebraska authorities. In addition to being archived, drawings, reports, analyses and photographs relating to the buried structures were encapsulated and placed within the structure in two locations.

Details of the retirement of the Hallam facility are reported in References 10,11, and 12.

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(10) A. Giambusso, "Four Decommissioning Case Histories," Nuclear News, 13:40, June 1970.

(11) W. F. Heine and B. F. Ureda, Decontamination and Disposition of Hallam and Piqua Reactors. Reference 7, p. 139.

(12) HALLAM Nuclear Power Facility, Entombment. Report on Retirement of HALLAM Nuclear Power Facility, AI-AEC-12709, May 1970.

3.1.3 Piqua Nuclear Power Facility, Piqua, Ohio

The Piqua Facility, an organically cooled 45 MWt power reactor went critical in 1963, was shutdown in 1966 because of coolant problems and was decommissioned by entombment in 1969.(13) The Piqua site was purchased by the U.S. Government and leased to the City of Piqua. The decommissioning activities were undertaken by City of Piqua personnel with engineering and safety support from Atomics International. Consulting was provided by the Battelle Memorial Institute.

A reactor retirement plan, work specifications and detailed procedures were prepared. A safety analysis and study evaluation of residual radionuclides was conducted and reported.

Reactor core components, fuel and other radioactive materials were shipped to a federal repository under normal operating procedures. Organic coolant was disposed of by burning. Contaminated piping and equipment inside the Reactor Building was removed or decontaminated and the Reactor Building converted to a warehouse. The reactor vessel, thermal shield, grid plates and support barrels remained in place; the vessel was filled with sand, weld sealed and all penetrations into the reactor complex were plugged. The complex was then sealed with a waterproof barrier and concrete cover. The development of special equipment or techniques was not required.

The total radioactivity of significant radionuclides sealed in the facility was 260,000 Ci. In addition to being archived, detailed records of all operations were duplicated and placed in sealed metal boxes at the site.

Cost estimates or actual cost totals were not available from the literature studied. Decommissioning lasted from July 1966 to August 1969.

3.1.4 Boiling Nuclear Superheater (BONUS) Power Station, Ricon, Puerto Rico

This was a 50 MWt boiling water reactor with nuclear superheat which was entombed. The reactor ceased operation in 1967 and the operating contract was terminated a year later. Increasingly stringent AEC design criteria involving expensive retro-fitting, poor economics, low availability and change in

(13) Atomics International. Retirement of the Piqua Nuclear Power Facility. AI-AEC-12709, 1969.

emphasis away from the superheat program led to the decision to decommission. The utility, the Puerto Rico Water Resources Authority, was to reduce the decommissioned plant to the status of a static exhibition open to the public for a maximum of 5 years. The utility was responsible for implementing decommissioning, preparing documents, scheduling and carrying out the operations. Control of the program was achieved in accordance with activity specifications and detailed procedures. All activities except entombment construction were allowed to be implemented before issue of the AEC dismantling order.

The work was divided into four phases:

Phase 1 Initial radiation survey; sampling of selected plant equipment and piping; shipping of spare unused fuel assemblies; removal of spent fuel from pressure vessel; permanent disabling of control rod drive mechanism.

Phase 2 Shipping of spent fuel, radioactive sources and wastes; decontamination; preparation for entombment.

Phase 3 Construction of entombment structure.

Phase 4 Preparation of documentation for removal of license requirements; handing over facility for exhibition purposes.

A radiological safety analysis was conducted to assist design of the entombment structure. The initial entombed radioactivity total was approximately 50,000 Ci comprising 71% ^{55}Fe, 29% ^{60}Co, and <1% ^{63}Ni. The dose rate at the entombment surface was not to exceed 0.4 mR/hr at 1 cm except for permissible hot spots up to 1 mR/hr as long as an average surface radiation level of 0.2 mR/hr was not exceeded.

A hazard assessment was made of the decommissioned plant for a postulated design basis accident (severe earthquake followed by tidal wave flood). Even on the basis of the most pessimistic assumptions it was calculated that such an accident would not result in unacceptable radiation doses.

The decommissioning aspects of the BONUS facility are described in more detail in Reference 14.

3.1.5 Walter Reed Research Reactor, Washington DC

The Walter Reed Research Reactor, built in 1961, was dismantled 10 years later, in 1971. The facility was an Atomics International Model L-54 homogeneous fueled reactor having a maximum operating power of 50 kW. The reactor was surrounded by a four-story research institute and was housed 20 ft below ground with only limited access via elevators. Heavy duty cranes and equipment could not be used.

The fuel, 26.5 liters of aqueous UO_2 SO_4 and 1215 g of ^{235}U enriched uranium, was removed in specially shaped containers. Recombiner unit water and decontamination solutions were solidifed in vermiculite and shipped in shielded stainless steel drums.

A hydraulic device called a 'Darda' rock splitter was used to demolish the heavy, dense concrete biological shield.(15) Very high lateral pressures were generated by this tool which was inserted in drilled holes. Fracture planes were established by this method and conventional road surface breakers were used to separate the concrete. Normal research institute operations continued almost uninterrupted during dismantlement and decontamination. Radioactive materials were removed at nights and on weekends.

No information is available on costs or on radiological experience. A brief review of the reactor dismantlement is given in Reference 16.

3.1.6 Pathfinder, Sioux Falls, South Dakota.

Pathfinder was a 66 MWe boiling water reactor with integral nuclear superheater that was placed in Safe Storage After discovering component

(14) Boiling Nuclear Superheater Power Station Decommissioning Final Report, WRA-B-70-500, Prepared by Puerto Rico Water Resources Authority (San Juan, Puerto Rico) and United Nuclear Corporation (Elmsford, New York), September 1970.

(15) "A World's First: Darda Splitter Used for Dismantling a Reactor," Construction. February 21, 1972.

(16) B. G. Bass and E. C. Holman, "The Walter Reed Research Reactor Dismantling Project." Trans. Am. Nucl. Soc. 15:897, November 1972.

failures following a forced shutdown in 1967, the decision was taken to convert the plant to conventional operation replacing the reactor with three fossil-fueled boilers. The operating license was eventually replaced by a two-part license, a Part 50 possession-only license for the isolated reactor and fuel storage and a Part 30 by-product license. Major activity reported concerned the conversion of the turbine cycle equipment.[17] Piping and turbine components were decontaminated during this operation. Decontamination fluids were barrelled, solidified and shipped for burial. Over three hundred 208-liter (55-gal) barrels of solidifed waste were removed from the site. Total cost was estimated to be $3.7 million.

3.1.7 Saxton Nuclear Experimental Facility, Saxton, Pennsylvania

The Saxton plant was a 23.5 MWt prototype pressurized water reactor supplying steam to an existing 10 MW turbo-generator. The reactor was located in the Saxton Steam Generating Station of the Pennsylvania Electric Company and operated by the Saxton Nuclear Experimental Corporation (SNEC) under an AEC Title 10 part 50 License DPR-4. Decommissioning was accomplished by placing the facility in Safe Storage. SNEC was responsible for all decommissioning activities including those of contractors. These activities were carried out in accordance with written procedures approved by SNEC. Decommissioning was completed during 1973.

Prior to decommissioning, an extensive planning program was carried out which included:

- an assessment of the optimum way of decommissioning the plant;
- preparing the decommissioning plan;
- Licensing the plan with AEC.

A decision was reached to proceed with a minimum decommissioning[b] and included the option of removing all radioactive material after a delay period (of, say, 50 years). This procedure would involve three levels of licensing:

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[17] M. N. Bjeldanes, "PATHFINDER" Nuclear News, 13 p: 56, June 1970.

[b] As used herein, "a minimum decommissioning" necessitates the use of continued surveillance; this corresponds to the Regulatory Guide 1.86 definition of "mothballing."

- Reduction in status from 10CFR Part 50, Utilization, to Part 50 Possession Only.
- Transition from Part 50, Possession Only, to Part 30, Status.
- Reduction from Part 50, Possession Only, to Part 30, Status.

Details of the planning and licensing for the Saxton facility are given in References 18 and 19.

3.1.8 Experimental Breeder Reactor, EBR-1, Idaho Nuclear Engineering Laboratory, Scottsville, Idaho

EBR-1 was the world's first source of nuclear electricity, first demonstrated in 1951. This fast breeder reactor had a eutectic alloy sodium-potassium coolant. EBR-1 suffered a core melt-down accident in 1955. It was eventually decided to make EBR-1 the site of a National Historic Monument and ceremonies took place in 1966. Public access could not be permitted due to radioactive contamination and hazardous accumulations of NaK. Steps to correct this situation were taken in 1973 when a decontamination and decommissioning program was initiated. The program plan was performed and completed by Aerojet Nuclear Company, assisted by Allied Chemical Corporation and Argonne National Laboratory. Information regarding the deactivation steps taken are given in Reference 20.

3.1.9 Elk River Reactor, Minnesota

The Elk River Reactor (ERR) was a 58 MWt indirect cycle, natural circulation boiling water reactor, built under a USAEC contract and operated by the United Power Association (UPA). Commercial operation lasted four years from 1964 until final operation in 1968. UPA waived its option to purchase the plant and agreement was eventually reached between the AEC and UPA to dismantle the plant and restore the site to as near as possible its original condition.

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(18) B. J. Reckman and C. R. Montgomery, Planning and Licensing for the Decommissioning of the Saxton Nuclear Experimental Facility. ASME Paper 73-WA/NE-8, 1973.

(19) Saxton Decommissioning Plan and Safety Analysis Report. Docket 50-146, Saxton Nuclear Experimental Corporation, General Public Utilities System, Reading, PA. April 1972.

(20) J. D. Cerchione, et al, "EBR-1 and Borax-V Reactor Deactivation," Trans. Am. Nucl. Soc. 8:114, June 1965.

Prior to decommissioning, a program was prepared that was to be carried out in three overlapping phases: planning, dismantling and final site closure.

Dismantling was carried out in three overlapping stages:

- removal of the most highly radioactive components, e.g. reactor internals and pressure vessel
- removal of systems and equipment outside the biological shield which contained low level contamination
- removal of noncontaminated structures

It was decided to use plasma arc cutting under water and oxy-acetylene cutting in air to dismantle the inner and outer thermal shields and the pressure vessel. Plasma arc cutting was not used on the outer thermal shield because of the high temperatures that would vaporize the lead liner. A full test and development program was carried out on the cutting processes to be used. These included bench scale tests and full scale mockup tests on the plasma arc cutting procedures and equipment and separate tests on the in-air cutting procedures. A manipulator system for remote handling of the cutting torches was developed at Oak Ridge National Laboratory (ORNL) for this operation.

For the removal of concrete, conventional drilling methods were feasible up to a depth of 0.6 m but were uneconomical due to the time element. Controlled use of explosives (0.7 kg maximum dynamite charges) was successful in removing the ERR biological shield with adequate safety and without release of radioactive contamination. An important consideration was that the Reactor Building was located only some 67 m from an operating electrical generating facility.

The total project cost including technical support services was $6.15 million, the highest constituents of costs were material disposal ($1.25 million), removal and disposal of the bio-shield ($1.23 million) and removal and disposal of the pressure vessel ($1.06 million) in shallow land burial grounds.

The decommissioning activities are described in References 21, 22, and 23.

3.1.10 Peach Bottom 1, York County, Pennsylvania

Peach Bottom 1, a 40 MWe prototype high temperature gas-cooled reactor, began commercial operation in 1967. The plant was scheduled to be shut down in 1974 after seven years' commercial operation. The decision to decommission was taken because of the high cost of retrofitting modifications required to meet more stringent safety criteria.

A full evaluation of the implications of decommissioning as regards the schedule, safety costs and licensing was carried out by the utility, Philadelphia Electric Company, and the SUNTAC Nuclear Corporation. Several decommissioning alternatives were considered in light of the following criteria:

- Current state and federal licensing problems and possible changes in regulations that could affect the cost or viability of the options.
- Diminishing licensing obligations throughout the life of each option.
- Cost of disposal of radioactive materials.
- Cost of preparing the detailed decommissioning plan, technical specifications and safety analysis reports for each option considered together with an environmental impact statement for entombment and removal options.
- Decontamination requirements for each option.
- Schedule considerations influencing the availability of operating staff.

(21) R. Blumberg, "Technology for Dismantling Large Radioactive Structural Components," Proceedings of the First Conference of ERDA Facilities. CONF-750827, p. 71, Idaho Falls, ID, August 1975.

(22) B. J. Davis, Elk River Reactor Dismantling. ibid, p. 83.

(23) D. McConnon, "Operational Health Physics during Dismantling of the Elk River Reactor." Proceedings of the 3rd International Congress of IRPA, Washington, DC, 1973.

The decommissioning plan included a schedule of operations to take place over a period of 24 months and contained manpower details, schedule of activities, safety analyses, proposed surveillance program and projected final facility status.

The option chosen for Peach Bottom 1 was that of minimum decommissioning, reducing the controlled access area to include only the Reactor Containment Vessel and maintaining the facility under a Part 50 possession only license. After a period of 50 years, major radioactive materials would be removed to a licensed burial ground. No significant dismantling of the facility took place during initial decommissioning. Fuel handling equipment was disabled and kept in place after external decontamination. All penetrations into the containment were cut and capped outside the containment vessel wall. A 12 mm pipe and filter vent was provided to prevent any pressure build-up in the vessel.

No special techniques or equipment were reported to be required for decommissioning. Normal operational procedures applied to the removal of fuel and radioactive materials.

Although final costs for decommissioning Peach Bottom are not available, the estimated cost obtained from the pre-decommissioning evaluation was just over $2 million at 1974 prices for the minimum decommissioning option including initial cost, 50 year maintenance and disposal of radioactive materials. It was estimated that the initial cost of minimum decommissioning (approximately $500,000) would be about equal to the cost of minimum decommissioning of a similar sized light-water reactor since there was a high ratio of engineering and licensing cost to field cost.[24]

It was also reported that any significant modification to the original Peach Bottom 1 design would have been unlikely to make decommissioning significantly easier, and it was also felt that this was probably true of large power reactors being constructed at the time. The costs of any radical changes would be significant, and in any case, existing design features were inherently beneficial when the minimum decommissioning option was selected.

[24] R. J. Stouky and E. J. Kohler, Planned Decommissioning of the Peach Bottom Unit No. 1 High Temperature Gas-cooled Reactor. ASME Paper 73-WA/NE-7, 1973.

3.1.11 Sodium Reactor Equipment (SRE), Santa Susana, California

This plant and its associated auxiliary facilities have been maintained in storage since 1968 awaiting funds for dismantling.

A remote manipulator system was to be developed for use in cutting up system components. The design developed by Oak Ridge National Laboratory for dismantling Elk River Reactor was adopted by Atomics International for use on the Sodium Reactor Experiment. A vessel mock-up was constructed to check manipulator operation. Cutting would be carried out under water. Explosive demolition techniques similar to those used for the Elk River Reactor were used for the removal of concrete structures at this and other Department of Energy (DOE), formerly ERDA, facilities at Santa Susana. Remote techniques for piping and component removal such as explosive pipe cutting used in offshore drilling operations were to be used wherever possible.

A Dismantling Plan was prepared together with Activity Requirements and Detailed Working Procedures. During decommissioning, all quality assurance and health, safety and radiation services records were maintained for permanent reference. The final report for the facility will record the effort, schedule and costs expended compared to original estimates. In addition, any special problems and their solution will be noted together with any tooling or process developments that would be useful for future programs. Approval of remaining radioactivity levels by DOE in agreement with Atomics International and the California Bureau of Radiological Health will also be recorded in the final report.

Decommissioning was commenced with the preparation of the SRE Dismantling Plan in 1975. Eight other DOE facilities at the AI Santa Susana site have also been included in an overall program of decontamination and disposition, some of which have been completely dismantled. Removal of the 20 MW SRE facility is described in Reference 25.

(25) W. F. Heine and A. W. Graves, Preparation for Decontamination and Disposition of the Sodium Reactor Experiment and other ERDA Facilities at AI. Ref. 7, p. 417.

3.1.12 Other Nuclear Reactor Decommissioning Experience[a]

Three Oak Ridge National Laboratory (ORNL) reactors were dismantled in 1954 and 1955.[26] These were the Homogeneous Reactor Experiments (HRE-1 and HRE-2 reactors) and the Aircraft Reactor Experiment (ARE reactor).

The Hanford Production Reactors have been retired.[27] The reactor structures were considered to be adequate to safely contain the radioactive material inventory. Fuel was removed, cavities were dried, the tubes were capped, and the control rods were disconnected. Routine surveillance has been provided.

Two nuclear power plants unique to military utilization were the U.S. Army's PM-2A and the U.S. Navy's PM-3A. The PM-2A was a 1.5 MWe power reactor system installed in the cut-and-cover snow trenches of Camp Century, the "City Under the Ice", in Northern Greenland. It was completely dismantled and removed from its site in 1964.[28] The complete removal of PM-3A, a 9.4 MWt unit, formerly located at the McMurdo Station, Antarctica, took about two years and was completed in early 1975.

The SL-1 Reactor at Idaho Falls was completely dismantled following an accident in 1961. High radiation fields and the presence of wide-spread contamination complicated the operation. The reactor and the building were completely demolished and the radioactive wastes were transferred to a local burial ground.[29]

(a) Costs and occupational radiation exposures for these decommissioning projects were unavailable from the literature referenced.

(26) S. E. Beall, "Experiences in the Decommissioning and Dismantling of Three ORNL Experimental Reactors," Trans. Am. Nucl. Soc. 8:113, June 1965.

(27) S. L. Nelson, "Operational Procedures in Deactivation of the Hanford Production Reactors." Trans. Am. Nucl. Soc. 8:116-117, June 1965.

(28) J. P. Franklin, "Removal of the PM-2A Nuclear Power Plant from Camp Century (U.S. Army)," Trans. Am. Nucl. Soc. 8:117, June 1965.

(29) Final Report of SL-1 Recovery Operation, May 1961 through July 1962, IDO-19311 (1962).

3.2 DECOMMISSIONING HISTORY OF NONREACTOR FUEL CYCLE FACILITIES

Many other fuel cycle facilities, ranging in size from prototype fuel reprocessing plants to one-room experimental laboratories, have been safely decommissioned. A partial listing of some of these facilities is given in Table 3-2. In many cases, the precautions and controls necessary for dealing with plutonium, polonium, and radium had to be considered. It should be noted that this is an additional consideration not normally present when decommissioning nuclear reactors. From the variety of facilities shown in Table 3-2, it is evident that the technology and expertise to decommission any type of fuel cycle facility has been effectively and safely demonstrated.

TABLE 3-2. Nonreactor Nuclear Facilities That Have Been Decommissioned

Facility	Location	Year Decommissioned	Type of Decommissioning	Reference
Polonium-210 Facilities (Units III & IV)	Mound Laboratory; Miamisburg, Ohio	1950	Partial Dismantlement; decontaminated to release levels	30
Cave Facility (Radium-226 and Actinium-227 Processing Facility)	Mound Laboratory; Miamisburg, Ohio	1957	Partial Entombment, remainder decontaminated to release levels	30
Prototype Reprocessing Plant	Fontenay-aux-Roses Nuclear Research Center, France	1963	Dismantled	31
SM Facility (Space Programs, Plutonium-238 Facility)	Mound Laboratory; Miamisburg, Ohio	1972	Decontaminated and placed in Safe Storage (Mothballed); awaiting Final disposition by DOE (formerly ERDA)	30
Plutonium Filter Facility (Building 12)	Los Alamos, (LASL) NM	1973	Dismantled	32
Laboratory For Plutonium Criticality Studies (P-11)	Hanford Operations (ERDA) Richland, WA	1974	Dismantled	33
Plutonium Physics Study Building No. 21	Area 33, LASL	1975	Dismantled	34
Eurochemic Reprocessing Plant (Demonstration)	Mol, Belgium	1976	Dismantled	35

(30) J. M. Garner, W. P. Davis, "A Summary Review of Mound Laboratory's Experience in D&D of Radioactive Facilities (1949-1973)," in Proceedings of the First Conference on Decontamination and Decommissioning (D&D) of ERDA Facilities, CONF-750827, pp. 225-231, Idaho Falls, ID, August 19-21, 1975.

(31) P. Cerre, "D'emantelement de l'usine pilote d'extraction du plutonium de Fontenay-aux-Roses," Bull. Inf. Sci. Tech., pp. 55-74, Paris 70, 1963.

(32) E. L. Christensen, et al., "Demolition of Building 12, An Old Plutonium Filter Facility," in Proceedings of the First Conference on Decontamination and Decommissioning (D&D) of Erda Facilities, CONF-750827, pp. 303-324, Idaho Falls, ID, August 19-21, 1975.

(33) M. N. Raile, "Demolition and Removal of Plutonium-Contaminated Facilities at Hanford," Trans. Am. Nucl. Soc. 22:758, San Francisco, CA, November 1975.

(34) E. J. Cox, et al., "Disposition of TA-33-21, A Plutonium Contaminated Experimental Facility," in Proceedings of the First Conference on Decontamination and Decommissioning (D&D) of ERDA Facilities, CONF-750827, pp. 165-206, Idaho Falls, ID, August 19-21, 1975.

(35) E. J. Detilleux, "Status of the Decommissioning Program of the Eurochemic Reprocessing Plant," Proc. of the International Symposium on the Management of Wastes from the LWR Fuel Cycle, CONF-76-0701, Denver, CO, July 11-16, 1976

3.3 LESSONS FROM PAST DECOMMISSIONINGS

We have learned from past decommissionings some of the aspects of the practicality and acceptability of the various decommissioning approaches. The necessary technology not only exists, but has been safely and successfully applied numerous times to a wide variety of nuclear installations. Because of the unique sizes, locations, and conditions under which past decommissionings took place, it can be seen that no two had identical problems or conditions. However, the basic approach to any mode of decommissioning remains virtually unchanged; i.e., the gathering of staff manpower, a period of planning and preparation, followed by the desired decommissioning operations. This fundamental course of events varies only in the numerous plant-specific refinements applied to the various stages of decontamination and decommissioning. The area of greatest challenge lies in improving job-specific technology such as remote cutting equipment and improved decontamination techniques.

Past decommissionings have led us to more carefully consider the socio-economic impacts on the local communities, impacts on the environment, and the wisdom of considering design features that would facilitate the ultimate decommissioning of the nuclear facility.

Improvements in decommissioning techniques can be expected to occur and will occur. Witness the development and practical use of plasma arc cutting techniques and improvements in explosive techniques employed during the dismantlement of the Elk River Reactor and the Sodium Reactor Experiment. These techniques and others can be expected to be further improved. Improvements, in turn, will directly impact future decommissioning costs. Such changes are being followed with great interest by the power reactor owner-operator and by the NRC.

3.4 ONGOING EXPERIENCE

Radiation field buildup and subsequent operations personnel exposure is a recognized problem area that can impede operational maintenance and inspection and can impact the end-of-plant-life decommissioning operations. Efforts currently underway to reduce radiation levels buildup include methods for reduction of corrosion product formation in the reactor primary system, cost-effective primary system decontamination methods, more effective filter and purification systems, and modification to operational techniques that appear to have a direct influence on radiation fields. Industry involvement includes the gathering of available data to assess the overall extent and seriousness of the problem across the industry.[36]

Ongoing industrial activities impacting on exposure control and activated corrosion product reduction programs include:

- Decontamination operations at Dresden 1 (BWR, with steam generator)
- Surry and Turkey Point (PWR's) steam generators replacement programs
- Indian Point 1 (PWR) secondary side steam generator chemical decontamination

The latter two programs, when completed, will yield information that can have a significant bearing on decommissioning; e.g., effectiveness of chemical decontamination methods, steam generator removal technology, and exposure reduction techniques associated with these operations.

During reactor operations the radiation levels in many areas are controlled by radiation from the Reactor Coolant System and minimal efforts, if any, are made to keep surface contamination cleaned up. After 30 or 40 years of operation, these areas may have fairly high radiation levels. At Dresden 1, for example, it was learned that although chemical decontamination of the test loop was effective, considerable radiation levels were still present as surface contamination on floors and surrounding structures. This surface

[36] S. G. Sawochka, N. P. Jacob, and W. L. Pearl, Primary System Shutdown Radiation Levels at Nuclear Power Generating Stations. EPRI 404-2, December 1975.

was quite high (1 R/hr radiation readings), but prior to decontamination it was not controlling. This phenomenon may well be encountered in PWR decommissioning and may have an effect on occupational exposures and on the volumes of waste for disposal.

A DOE government program is establishing methods, costs, and priorities for the decommissioning of retired, contaminated DOE facilities at Hanford and at other DOE facilities.[37,38] Other Hanford companies have active programs to demonstrate the techniques for dismantling and consolidating contaminated equipment and facilities.[39]

Based on technical and economic evaluations of several decommissioning options, Safe Storage (mothballing) of the Peach Bottom Unit 1 under a Part 50 possession-only license was selected. The decommissioning of this reactor is nearing an end. However, in March 1975, implementation of the Peach Bottom End-of-Life Program, cosponsored by the U.S. DOE, formerly ERDA, and EPRI, was initiated. The prime objective of the program is to validate specific reactor design codes and predictions by comparison of actual predicted physics, thermal, fission product, and material behavior in Peach Bottom. It is concluded that such programs of end-of-life research, when appropriately coordinated with decommissioning activities, can significantly advance nuclear plant and fuel development technology.[40]

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(37) K. W. Harmon, "PNL Studies of D&D at Hanford," Proceedings of the First Conference on Decontamination and Decommissioning (D&D) of ERDA Facilities, CONF-750827, pp. 345-365, Idaho Falls, ID, August 19-21, 1975.

(38) J. W. Litchfield and J. C. King, Planning For Decommissioning and Decontamination of Hanford Nuclear Facilities. BNWL-SA-6450, Battelle, Pacific Northwest Laboratories, Richland, WA, September 1977.

(39) C. W. Manry, et al., Hanford Production and Waste Management Master Plan, ARH-2956, July 1974.

(40) E. J. Kohler, K. P. Steward and J. V. Iacono, "Peach Bottom Decommissioning and Component Removal," Conference on Reactor Operating Experience, Trans. Am. Nucl. Soc.. Suppl. 1 p. 51, August 1977.

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10. A. Giambusso, "Four Decommissioning Case Histories," Nuclear News. 13:40, June 1970.

11. W. F. Heine and B. F. Ureda, Decontamination and Disposition of Hallam and Piqua Reactors. Reference 7, p. 139.

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28. J. P. Franklin, "Removal of the PM-2A Nuclear Power Plant from Camp Centry (U.S. Army)," Trans. Am. Nucl. Soc. 8:117, June 1965.

29. Final Report of Sl-1 Recovery Operation, May 1961 through July 1962, IDO-19311 (1962).

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32. E. L. Christensen, et al., "Demolition of Building 12, An Old Plutonium Filter Facility," in Proceedings of the First Conference on Decontamination and Decommissioning (D&D) of ERDA Facilities, CONF-750827, pp. 303-324, Idaho Falls, ID, August 19-21, 1975.

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36. S. G. Sawochka, N. P. Jacob, and W. L. Pearl, Primary System Shutdown Radiation Levels at Nuclear Power Generating Stations, EPRI 404-2, December 1975.

37. K. W. Harmon, "PNL Studies of D&D at Hanford," Proceedings of the First Conference on Decontamination and Decommissioning (D&D) of ERDA Facilities CONF-750827, pp. 345-365, Idaho Falls, ID , August 19-21, 1975.

38. J. W. Litchfield and J. C. King, Planning For Decommissioning and Decontaminat of Hanford Nuclear Facilities. BNWL-SA-6450, Battelle, Pacific Northwest Laboratories, Richland, WA, September 1977.

39. C. W. Manry, et al., Hanford Production and Waste Management Master Plan, ARH-2956, July 1974.

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4.0 DECOMMISSIONING ALTERNATIVES AND STUDY APPROACH

Once a nuclear reactor has reached the end of its useful life, it must be decommissioned, i.e., placed in a condition such that future risk from the facility to public safety is within acceptable bounds. A number of decommissioned conditions (modes) are possible that will satisfy the requirements for decommissioning, ranging from a minimal cleanup effort with physical security (Safe Storage) to complete cleanup and removal (Dismantlement).

4.1 DECOMMISSIONING ALTERNATIVES

Three basic modes of decommissioning have been utilized in the past for the retirement of light water reactor facilities, safe storage, permanent entombment, and dismantlement. A summary of the basic conditions for Dismantlement and Safe Storage is presented in Table 4.1.1. Each of these terms as applied to the reference PWR, is defined and discussed in subsequent sections.

4.1.1 Definition Of Rational For Safe Storage

Safe Storage is defined as those activities required to place and maintain a nuclear facility in such condition that future risk from the facility to public safety is within acceptable bounds and that the facility can be safely stored for as long a time as desired.

Several subcategories of safe storage are possible. These are:

- Custodial Safe Storage [layaway[a]] - a minimum cleanup and decontamination effort is made initially, followed by a period of continuing care with the active protection systems (principally the ventilation system) kept in service throughout the storage period. Full-time onsite surveillance by security forces is required to prevent accidental or deliberate intrusion into the facility and the subsequent exposure to radiation or the dispersal of radioactivity beyond the confines of the facility.

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[a] This nomenclature was used in NUREG-0278 (Reference 1).

[1] K. J. Schneider and C. E. Jenkins, Study Coordinators, Technology, Safety and Cost of Decommissioning a Reference Nuclear Fuel Reprocessing Plant, Report of U.S. Nuclear Regulatory Commission by Battelle, Pacific Northwest Laboratory, NUREG-0278, October 1977.

TABLE 4.1-1. SUMMARY OF DECOMMISSIONING MODE CHARACTERISTICS

Mode	Facility Status	Plant/Site Use
Dismantlement	Plant Equipment - removed Continuing Care Staff - none Security - none Environmental Monitoring - none Radioactivity - removed Surveillance - none Structures - removal optional	Plant - Unrestricted Site - Unrestricted
Safe Storage		
Hardened	Plant Equipment - none operating Continuing Care Staff - none on site Security - hardened barriers, fencing and posting Environmental Monitoring - infrequent Radioactivity - hardened sealing Surveillance - infrequent Structures - partial removal optional	Plant - Conditional Non-nuclear Site - Conditional Non-nuclear
Passive	Plant Equipment - none operating Continuing Care Staff - optional (onsite) - routine inspections Security - remote alarms Environmental Monitoring - routine periodic Radioactivity - immobilized/ sometimes sealed Surveillance - periodic Structures - intact	Plant - Nuclear Only Site - Conditional Non-nuclear
Custodial	Plant Equipment - some operating Continuing Care Staff - some required Security - continuous Environmental Monitoring - continuous Radioactivity - confined Surveillance - continuous Structures - intact	Plant - Nuclear Only Site - Nuclear Only

- Passive Safe Storage [mothball[b]] - a more comprehensive cleanup and decontamination effort is performed initially, sufficient to permit deactivation of the active protective (ventilation) systems during the continuing care period. The structures are strongly secured and electronic surveillance is provided to detect accidental or deliberate intrusion. Maintenance of the integrity of the structures is required.
- Hardened Safe Storage (temporary entombment) - The comprehensive cleanup and decontamination is coupled with the construction of barriers around areas containing significant quantities of radioactivity. These barriers are of sufficient strength to make accidental intrusion impossible and deliberate intrusion extremely difficult. Surveillance requirements are limited to detection of attack upon the barriers and maintenance of the integrity of the structures.

All categories of Safe Storage are open-ended and some positive action is required at the conclusion of the period of continuing care to release the property for unrestricted use and terminate the license for radioactive materials. Depending upon the nature of the nuclear facility and its operating history, the necessary action can range from a radiation survey that shows the property to be releasable, to dismantlement and removal of residual radioactive materials. These latter actions, whatever their scale, are generically identified as deferred dismantlement.

Safe Storage is used as a means to satisfy the requirements for protection of the public while minimizing the initial commitments of time, money, occupational radiation exposure and waste repository space. Modifications to the facilities are limited to those which ensure the security of the buildings against intruders, and those required to assure containment of radioactive or toxic material. It is not intended that the facility would ever be reactivated. The principal activation product in a light water reactor

[b]This is the nomenclature used in Regulatory Guide 1.86 (Reference 2).

(2)U.S. Atomic Energy Commission Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, June 1974.

facility which has the potential for causing large occupational radiation exposures also has a relatively short half-life (^{60}Co, $T_{1/2} \sim 5.27$ years). Therefore, large reductions in personnel exposure and significant simplifications in the complexity of operations can be achieved by deferring any dismantlement efforts for up to fifty years. Additionally, many of the contamination and activation products present in the facility will have decayed to background levels after a lengthy storage period, thus permitting recycle of the valuable material back into commercial channels and greatly reducing the volume of material that must be packaged for disposal.

The reduced initial effort (and cost) of Safe Storage is tempered somewhat by the need for continuing surveillance and physical security to assure the protection of the public. Electronic surveillance devices are in service full-time, with off-shift readouts in a local law enforcement office or private security agency. These devices, which monitor for intruders, increases in radiation levels and detection of fires, will require periodic checks and maintenance.

Maintenance of the facility outer confinement surfaces and an on-going program of environmental surveillance is also necessary. The duration of the storage and surveillance period can vary from a few years to more than 100 years. The actual choice will be made by the facility owner, primarily on the basis of economic trade-offs. For example, if the value of the property for unrestricted use is large and the cost of continuing care is also large, there is an incentive to dismantle the plant earlier than would otherwise be dictated by the decay of radioactivity within the plant, and some penalty will be paid in terms of occupational exposure to obtain the unrestricted use of the site. Similarly, the decision on whether or not to chemically decontaminate the piping systems during the preparations for safe storage depends upon the cost of continuing care and the planned length of the storage and surveillance period. Assuming ^{60}Co is the controlling source of occupational exposure, a chemical decontamination campaign achieving a conservative decontamination factor (DF) of 10 (residual radioactivity reduced to 1/10) is approximately equivalent to a decay period of 17-1/2 years.

At the end of the period of Safe Storage, several things will remain to be done before the possession-only license can be terminated. In all probability, the ^{59}Ni and ^{94}Nb radioactivity in the activated reactor vessel internals will be significantly above unrestricted release levels, necessitating the removal, packaging and disposal of the internals at a regulated disposal site. The reactor pressure vessel may also have ^{59}Ni and ^{94}Nb radioactivity levels above unrestricted release levels, thus requiring removal and disposal of the vessel. If the Safe Storage period is sufficiently long ($\geq$110 years), it is expected that most dose-producing radioactive materials in the facility other than ^{59}Ni, ^{94}Nb and ^{137}Cs will have decayed to levels indistinguishable from normal background. The facility may be releasable without major additional decontamination, once the materials containing ^{59}Ni, ^{94}Nb, ^{14}C and ^{137}Cs have been removed and transported to an appropriate disposal site. The possession-only license can then be terminated.

Safe Storage consists of a short period of preparation (1 to 2 years), a variable period of continuing care consisting of security, surveillances and maintenance (up to about 100 years), and a short period of dismantlement, either partial (pressure vessel and internals) or total, depending on the plans of the owner.

4.1.2 Rationale For Rejection Of Permanent Entombment As A Viable Decommissioning Mode

Based on the guidance put forth in Regulatory Guide 1.86, entombment of a reactor facility requires the encasement of the radioactive materials in concrete or other structural material sufficiently strong and structurally long-lived to assure retention of the radioactivity until it has decayed to levels which permit unconditional release of the site. In previous reactor decommissionings, it was assumed possible to entomb the reactor pressure vessel and its internal structures within the biological shield since the principal source of radiological dose was ^{60}Co, which decays with a relatively short half-life (5.27 years). Thus, within about 100 years, the residual radioactivity will have decayed to levels indistinguishable from normal background, well within the safe structural lifetime of the entombment structure. The presence of any ^{94}Nb was ignored. The amount of ^{59}Ni formed in the relatively brief operating life of these early plants was sufficiently small as to present no significant hazard. However, in large power reactors that have operated for 30-40 years, the induced ^{94}Nb and ^{59}Ni activities in the reactor vessel and its internal structures are well above unconditional release levels and, since ^{59}Ni has an 80,000 year half-life and ^{94}Nb has a 20,000 year half-life, the radioactivity will not decay to unconditional release levels within the foreseeable lifetime of any man-made surface structure. Thus, the basic requirement for entombment cannot be met for the reactor vessel and internal structures. These activated components comprise over 90% of the total radioactivity present in the facility at reactor shutdown (not including fuel). Once these highly activated components have been removed, there is little incentive to entomb the small amount of radioactive material remaining at the site. Therefore, permanent entombment is not a viable decommissioning option for light water power reactors that have operated for 30 to 40 years.

4.1.3 Definition Of And Rationale For Dismantlement

Immediate Dismantlement provides a way to meet the requirements for termination of a possession-only license in the near term, thus eliminating long-term security, maintenance, and surveillance needs, and making the site available for unrestricted use within 4 to 5 years following reactor shutdown. To accomplish dismantlement requires that all potentially contaminated systems be disassembled and removed, and that all activated and/or contaminated material be removed from the facility and be transported to a regulated disposal site, thus occupying large volumes at the disposal site. Since this work is performed within a few years following reactor shutdown, the relatively short-lived radioactive corrosion products have not decayed significantly and large amounts of personnel radiation exposure are accumulated. The facility structures are decontaminated to unrestricted use levels and either put to some beneficial use or demolished, at the owner's option. In Immediate Dismantlement, large commitments of money, personnel radiation exposure, and disposal site space are made in exchange for prompt availability of the reactor site for other purposes and for the elimination of continuing security, maintenance, and surveillance needs.

An additional advantage of Immediate Dismantlement is the availability of a work force highly knowledgable about the facility (the operations staff).

Deferred Dismantlement, as would occur at the end of a period of Safe Storage, is a relatively straight-forward disassembly job, complicated by radioactivity primarily in the reactor vessel and its internals. Removal and transport of the materials containing ^{94}Nb, ^{59}Ni and ^{14}C to a regulated disposal site is the principal task that must be completed to qualify for termination of the possession-only license. All other radioactive dose-producing species will either have decayed to levels indistinguishable from background or will be removed from the facility. Further action, such as disassembly of the various systems and use or demolition of the buildings would be at the owner's discretion.

A disadvantage of Deferred Dismantlement is the lack of personnel familiar with the facility. More time for training and orientation would be needed. One potential solution to this problem would be the establishment of companies specializing in the decommissioning of nuclear reactor power stations and other nuclear facilities.

4.2 TECHNICAL APPROACH

The initial effort was to develop a plan to accomplish the objectives of this study, which are discussed in Section 1. The plan was developed by a team of key personnel with expertise in the primary areas of interest in the study. The areas of expertise included nuclear reactor stations and their operation, decommissioning techniques, chemical decontamination, chemical and radiological toxicant regulations, safety analyses (including pathways of toxic materials in the environment), operational health physics, and cost and benefit estimating and analyses. The resultant approach is shown in simplified form as Figure 4.2.1. The study was then carried out by the same staff or by staff with similar backgrounds.

The first step in conducting the analysis was to select the reference facility and to characterize it in sufficient depth to perform an engineering and safety analysis of decommissioning the facility. A contemporary specific existing plant was selected for this analysis. The total facility was placed

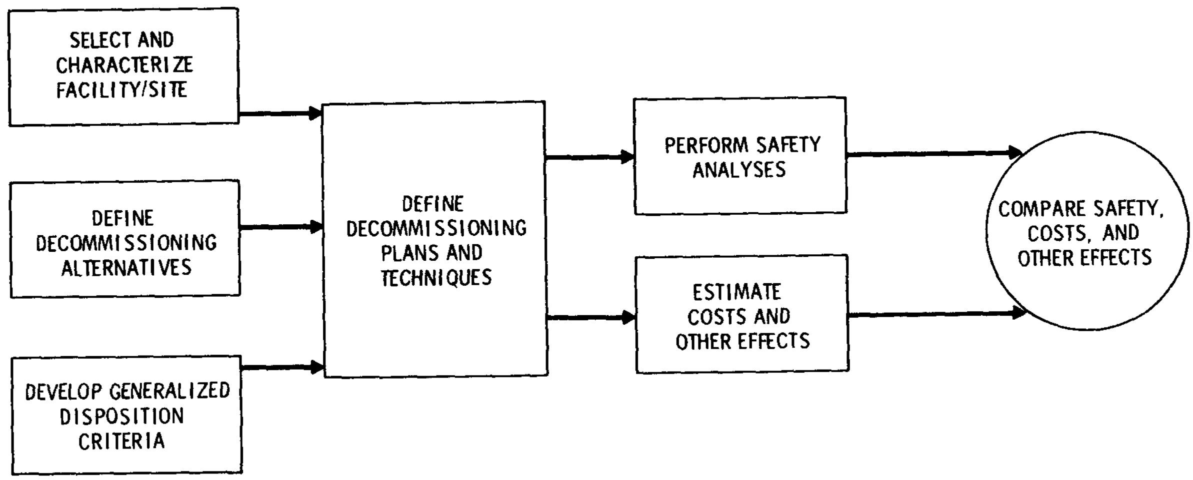

FIGURE 4.2.1. Approach for Decommissioning Study

on a conceptual generic site that is also being used in similar and related studies of other fuel cycle facilities. A detailed description of the facility was compiled, including development of information such as plant equipment and material sizes, volumes surface areas, and weights. Pre-decommissioning conditions for the plant and site were defined, including residual radioactivity levels.

A range of viable decommissioning mode characteristics (i.e., various forms of safe storage, and dismantlement) and site use limitation for decommissioned facilities (i.e., nuclear use only, conditional non-nuclear use, and unrestricted use) were selected. Related regulatory guidance was reviewed, summarized, and used as an aid and basis in the study.

Methodology was developed for defining residual radioactivity levels in facilities and sites that are acceptable for unrestricted use of the decommissioned facilities, based on the radiation dose to the maximum exposed member of the public from the variety of potential pathways through which radionuclides at the facility could reach man. This methodology was applied to develop example criteria for allowable amounts of residual radionuclides, based upon the assumed spectra for radionuclide mixtures at the plant/site, for unrestricted use of the decommissioned facilites. These criteria were then used in the analysis to define the extent of decontamination necessary to achieve the planned end use objectives.

Techniques for decontamination of facilities were reviewed. A work and time schedule was developed to conceptually decommission the reference facility for each of the two modes. The techniques utilized were selected on the basis of engineering judgment while maintaining a balance of safety and cost.

Safety analyses were performed for each of the decommissioning modes studied. These analyses included radiological and chemical exposures to the public and to workers for normal decommissioning operations and from potential accidents. Nonradiological industrial accidents to workers were also estimated. The safety analysis utilized established data and methodology to estimate the various factors required, such as release mechanisms, dispersion, pathways and exposure modes of the released materials.

Direct costs of decommissioning were estimated, including labor, materials, equipment, packaging, transportation, disposal, and surveillance costs where applicable. Costs were projected into the future to provide a reference base for estimating future financial requirements. Alternatives for financing decommissioning were examined and compared using example costs from this study. Cost ranges were defined to estimate the sensitivity of the total cost to variations in selected key cost element.

4.3 KEY STUDY BASES

From the outset a number of important ground rules were established to guide the emphases of the study. These bases were derived from the primary objective of the study--to provide an analysis of safety, costs, and other factors involved in decommissioning a commercial pressurized water reactor power station. The study is intended to provide background information useful to regulators, plant designers and operators of such facilities. From these objectives the key bases were established for all aspects of the study to assure that the overall study objectives were achieved. These key bases, listed below, can have major impact on the issues of safety, cost, and time for decommissioning. As stated earlier, many aspects of decommissioning will change with facility locations, specific facility shutdown conditions and residual contamination levels in the plant. The bases and assumptions used in this study must therefore be carefully examined before the results can be applied to a different facility and site.

1. The study is to yield realistic and up-to-date results. This primary basis is a requisite to meeting the objectives of the study, and provides the foundation for most of the other study bases.

2. The study is to evaluate, insofar as possible, a real and contemporary facility. This basis is an obvious necessity to meet the study objectives and the primary basis above. The facility selected as the reference for study, the Trojan Nuclear Plant, is felt to satisfy this condition since it is typical of the current generation of large PWR stations. The facility was assumed to be a single reactor station, rather than a multiple reactor complex.

3. The study is to include an analysis of a spectrum of decommissioning modes. This was done by investigating the modes of dismantlement, and safe storage followed by deferred dismantlement.

4. Only facilities planned to contain radioactive material and contiguous areas are dealt with in depth in the study. Decommissioning of separate nonradioactive subfacilities is to be accomplished by conventional demolition/salvage techniques.

5. Current and proven decommissioning technology and techniques are used. Where developmental techniques are applied, they are in an advanced state of development and believed to be ready for the application in this study.

6. A single decommissioning plan is evaluated for each mode analyzed. Where different techniques or assumptions have significant impact on the study results, the effects of alternatives are discussed at least qualitatively.

7. The decommissioning plans were selected to provide for adequate public/occupational safety in a cost-effective manner.

8. The performance of decommissioning is assumed to be relatively trouble-free; that is, no scheduling or cost allowances were made for unforeseen events that might impede the conduct of the work. This assumption may

lead to somewhat optimistic results, but is believed to be achievable with good planning and preparations.

9. It is assumed that the accessable plant process areas have been kept relatively clean during the operating period to allow for easier operational maintenance. As a result, expected contamination levels from deposited radioactivity are generally modest, but should be reasonably consistent with the quality of operation expected in modern commercial facilities.

   Accidents that may have occurred during plant operaton are assumed to be relatively minor with respect to contamination of normally clean surfaces (e.g., the outsides of process vessels, the soil within the site, etc.). Any major contamination episodes are assumed to have been reasonable well cleaned up immediately following the event.

   Inaccessable areas are assumed to have surface contaminations that were built up over the forty years of station operation. Specifically, contamination inventories are assumed to accumulate at the rate of 1/40 per year of the total accumulation, for the assumed 40 years of plant operation.

10. Decommissioning and radiation protection philosophies and techniques applied conform to the principle of keeping occupational radiation doses as low as is reasonably achievable (ALARA).

11. Wastes resulting from decommissioning are assumed to be sent to regulated burial grounds. Incremental costs for disposal of highly activated material at a Federal Deep Geological Disposal facility are estimated.

From these major study bases, more specific bases and assumptions were derived for specific study areas. These latter bases and assumptions are presented in the respective report sections where they are used.

REFERENCES

1. K. J. Schenider and C. E. Jenkins, Study Coordinators, Technology, Safety and Costs of Decommissioning a Reference Nuclear Fuel Reprocessing Plant, Report of U.S. Nuclear Regulatory Commission by Battelle, Pacific Northwest Laboratory, NUREG-0278, October 1977.

2. U.S. Atomic Energy Commission Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, June 1974.

5.0 REGULATORY CONSIDERATIONS FOR DECOMMISSIONING

In planning for the decommissioning of the reference PWR, the facility licensee must be cognizant of applicable regulatory requirements. This section identifies and highlights existing regulations, standards, and guides that apply to decommissioning the reference PWR. Presentation and discussion of these regulations is grouped in accordance with the phase of decommissioning activity to which it applies, i.e., planning and preparation, dismantlement or preparations for safe storage, and continuing care.

Users of information from this section should recognize that regulations and guidelines in this area are dynamic. National policy relating to the LWR nuclear fuel cycles is changing and federal reorganizations in the energy area are forthcoming. For example, the NRC has just announced[1] that it is considering the development of a more explicit overall policy for decommissioning nuclear facilities, and has issued a plan[2] for assuring that (a) a general decommissioning policy is developed, (b) the appropriate changes in regulations are developed, (c) the detailed information needed for use in decommissioning licensing decisions is developed, and (d) guidance is established for the facilitation of decommissioning. The information found in this section reflects the current status, regulations and federal guidelines, and can be used as a departure point for future application.

5.1 REGULATIONS PERTAINING TO THE PLANNING AND PREPARATION PHASE

Prior to terminating the operation of a PWR, the licensee will decide on the final disposition of the facility (with approval of the NRC) and plan how to accomplish that end point. A key consideration upon plant shutdown is the termination of the operating license regulated by 10 CFR 50 Licensing of Production and Utilization Facilities. Section 50.82, "Application for

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(1) Federal Register, Vol. 43, No. 49, p. 10371, March 13, 1978.

(2) Plan For Reevaluation of NRC Policy on Decommissioning of Nuclear Facilities, NUREG-0436, Office of Standards Development, U.S. Nuclear Regulatory Commission, March 1978.

Termination of Licenses" specifies the requirements that must be satisfied to terminate an operating license. Regulatory Guide 1.86 describes methods acceptable to the NRC for satisfying the requirements of Section 50.82.

A licensee will request amendment of his operating license to allow him to possess radioactive and/or special nuclear materials but not operate the facility in a production mode. Because of the nature of some of the decommissioning activities anticipated at the site, the NRC may elect to issue either a possession-only license with administrative controls and facility requirements appropriate for the decommissioning option selected or a modified operating license. The rationale behind this logic is that although the plant operating functions are changed significantly during decommissioning, many unit operations may be similar (i.e., chemical decontamination, waste treatment and solidification) Active operations will be conducted in the plant involving radioactive material and utilization of existing systems and components that will result in release of effluents to the environment. Additionally, unplanned releases of radioactive material are possible from accidents during decommissioning. Title 10 CFR Part 50 Section 50.59, "Authorization of Changes, Tests and Experiments" and Section 50.90, "Application for Amendment of License or Construction Permit" provides the rules by which a licensee may amend his license. The amended facility license results from NRC approval to amend requirements in the technical specifications that are applicable to normal facility operations. It appears that the necessary requirements to assure public safety during decommissioning can be covered whether or not the license is a modified-operation or a possession-only license.

As part of the amended license, the licensee must have authorization for special nuclear material (10 CFR Part 70, Special Nuclear Materials), byproduct material (10 CFR Part 30, Rules of General Applicability to Licensing of Byproduct Material) and source material (10 CFR Part 40, Licensing of Source Material) until the radioactive material and any source and special nuclear material are removed from the facility.

Consistent with the intent of 10 CFR Part 51, Licensing and Regulatory Policy and Procedures for Environmental Protection, before decommissioning begins, an environmental impact statement or environmental impact appraisal will have to be prepared describing the probable effects of the proposed decommissioning actions. These requirements are defined in Section 51, Subpart A. Section 51.5.b(7) states that license amendments or other orders authorizing decommissioning of a PWR may or may not require an impact statement of such planned actions. If judged that an impact statement is not required, a negative declaration[a] and an environmental impact appraisal must be prepared in accordance with Section 51.7 and 51.50(d). Guidance is provided to the NRC on the need for an impact statement by the Council on Environmental Quality Guidelines, 40 CFR 1500.6.

In addition to Regulatory Guides, the NRC has internal guidance for their staff on how safety analysis reports and environmental impact statements should be evaluated. These guides are found in NUREG-75/087[3] and NUREG-0158.[4] Decommissioning is also addressed in NUREG-0158 (in preparation).

The financial qualification of the licensee is an important area considered by the NRC during a review of a license application. Regulations covering this area are found in Section 50.33(f) and Part 50, Appendix F.5. The latter regulation is an elaboration of the former, specifying that the license application shall include information showing that the applicant is financially qualified to provide for the removal and disposal of radioactive waste during operation and upon decommissioning of the facility. Section 50.33(f) addresses the necessity of sufficient funds to operate the facility for the period of the license or 5 years, whichever is greater, plus the estimated cost of permanently shutting the facility down and maintaining it in a safe condition. The latter regulation does not totally address decommissioning of the facility.

(a) A negative declaration is a document prepared by the NRC that states that the NRC has decided not to prepare an environmental impact statement for a particular action, and that an environmental impact appraisal setting forth the basis for that determination is available for public record.

(3) Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-75/087 (NTISUB/B201), Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, September 1975.

(4) Environmental Standard Review Plans for the Environmental Review of Construction Permit Application for Nuclear Power Plants (Draft), Nuclear Regulatory Commission, January 1977.

Currently, no regulation specifically requires a detailed decommissioning plan. Regulatory Guide 1.86 may be interpreted to imply that one is needed; it states that the NRC will impose requirements depending on the decommissioning option selected. It is the authors' feeling that such a plan, namely, a Master Decommissioning Plan (MDP) should be required and included as part of the amended license. The MDP should include the decommissioning objectives for the facility/site, safety analysis and procedures, safeguard plans, contingency plans for unplanned events postulated to occur, and a time schedule.

As part of this plan, quality assurance (QA) of the decommissioning should be addressed" ... to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public" (Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"). The requirements in Appendix B pertain to topics such as design, purchasing, and fabrication, and do not specifically address decommissioning. Guidance is also found for nuclear facilities in the NRC's Standards Review Plans (SRP) 17.1, "Quality Assurance During the Operating Phase." The principles and objectives of such guidance should be applied to all activities of decommissioning. Therefore, applicable portions of Appendix B, SRP 17.1, and 17.2, should be used to develop a QA plan to inclusion in the license amendment and the MDP.

Other considerations of significant concern, mainly to the licensee, are the amount of the annual license fee and the facility insurance premimums required to satisfy regulations during the active decommissioning and continuing care periods. These costs are dictated by the type and quantity of radioactive and/or special nuclear materials, the type of activities being conducted, and correspondingly the type of license regulating the activities. Licensing fees are addressed in 10 CFR, Part 170; the schedule of fees for production and utilization facilities (Part 50 license) is in Section 170.21. The requirements for financial protection and indemnity agreements are provided in 10 CFR, Part 140. The levels of protection required for a PWR during decommissioning are not specifically defined.

5.2 REGULATIONS PERTAINING TO THE ACTIVE DECOMMISSIONING PHASE

Once a decommissioning mode has been selected and the Part 50 license has been modified, the actual decommissioning activities can be inititated. Section 50.82 and Regulatory Guide 1.86 identify decommissioning options considered acceptable to NRC for nuclear reactors. Although the interpretations of some of the definitions are expanded and the term "mothballing" is not used in this study, these same alternative modes have been considered in this study as viable options (with exception of entombment) for decommissioning a PWR. The facility will be placed in the planned disposition mode according to the Master Decommissioning Plan.

During the period from plant shutdown and until quantities of radioactive materials and/or special nuclear materials that require safeguards and other regulatory control are removed from the facilities, safeguards and security precautions must be continued. Regulations defining required precautions are found in 10 CFR Part 70 Special Nuclear Materials and 10 CFR Part 73 Physical Protection of Plant and Materials. The highly radioactive nature of the remaining special nuclear material (i.e., irradiated fuel rods) makes it very unlikely that any of the material would be stolen. The principal concern is to protect against acts of industrial sabotage that could endanger the safety of the work force and the public.

As the final step in disposing of the fuel, a final cumulative Material Unaccounted For (MUF) value must be established. This is generally not too difficult since it is based on piece count of the fuel rods. Likely source of MUF at a PWR are misplaced fuel rods and pellets lost from severely damaged fuel rods, all of which will most probably be found as the fuel pool is emptied.

During the actual decommissioning of the PWR, regardless of the mode selected, radioactive waste will be accumulated, treated, packaged, stored, and transported to one or more disposal sites. This includes the solidification of radioactive liquid waste from decontamination flushing solutions. Regulations defining the requirements to assure safety of the public and occupational workers from such waste-related activities are found in 10 CFR Part 50, Licensing of Production and Utilization Facilities, 10 CFR Part 20, Standards for Protection Against Radiation, and 10 CFR Part 71, Packaging of Radioactive Materials for Transport and Transportation of Radioactive Material Under Certain Conditions. Means for compliance with these regulations, including those for safeguards and security precautions will be defined in the specifications and plans of the amended license at the start of decommissioning. These are the same requirements, although perhaps to a lesser degree in some areas, that the licensee would have to address in his application to construct and operate a PWR.

The decommissioning of a PWR will entail the disposal of residual radioactive materials, and components contaminated with transuranic elements, fission products, and activation products. Detailed procedures for the disposition of these wastes must be clearly defined in the application for license amendment.

Little guidance currently exists on the final disposition of some of the types of waste anticipated from decommissioning a PWR, such as the highly radioactive reactor vessel components and contaminated pieces of equipment. Shallow land burial of these wastes is currently being reviewed. A decision to require deep geologic disposal of these wastes could have a sizeable cost effect on the licensee because of the major difference in cost between shallow land burial and deep geologic disposal in a Federal repository. A review of the Federal regulations pertaining to the licensing and operation of commercial and DOE-owned waste management facilities has been recently completed.[5]

The radioactive effluents from waste processing operations or other activities during decommissioning must comply with Environmental Protection Agency regulations as well as 10 CFR Part 20. Currently, no specific EPA regulations exist for decommissioning. The EPA's 25 mrem/yr limit of exposure to the maximum exposed member of the public from operating facilities of the nuclear fuel cycle, defined in 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operations, excludes waste management

[5] J. J. Cohen, et al., Determination of Performance Criteria for High-Level Solidified Nuclear Waste, NUREG-0279, Lawrence Laboratory for USNRC, July 1977.

activities but such limits are now being developed. It is anticipated that a radiation dose limit from waste management operations similar to the 25/mrem/yr fuel cycle limit will be developed by the EPA. This new limit may well include the impact of decommissioning.

The NRC is now in the process of developing comprehensive waste management regulations that will include wastes from decommissioning. The NRC is currently considering recommending that Low Level Burial Grounds be placed under Federal operation. Regulatory authority for decommissioned facilities in Agreement States[6] is relinquished to the States. Since Section 274(b) of the Atomic Energy Act of 1954, as amended, requires Agreement State programs to be compatible with NRC regulations, the NRC will require that Agreement State programs reflect the NRC's lead in the area of decommissioning.

Requirements for the packaging of the radioactive material are defined by transportation regulations. Packaging of the decommissioning wastes will also be dictated by their storage and/or ultimate disposal mode. Regulations governing the transport of radioactive materials have been established to prevent the loss or dispersal of material during shipment and to assure the safety of the public and the transportation workers. Some overlapping responsibility currently exists for regulating the safe transport of radioactive materials. Primary responsibility at the Federal level lies with the Department of Transportation (DOT) Material Transportation Bureau, and secondarily with the Nuclear Regulatory Commission (NRC).

A "Memorandum of Understanding" between these two agencies was signed in 1966 and revised in 1973.[7] This memorandum calls for cooperation between the DOT and the NRC and delineates the responsibilities of each agency.

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(6) 10 CFR 150, Exemptions and Continued Regulatory Authority in Agreement States Under Section 274.

(7) C. K. Beck, "Intergovernmental Relationships in the Transport of Radioactive Materials," in Proceedings of the Second Annual Legislative Workshop, CONF-730588, Oak Ridge, TN, May 1973.

The DOT is responsible for promulgating and enforcing safety standards governing packaging and shipping containers and for the labeling, classification, and marking of all packages. The DOT also implements safety standards for the mechanical condition of carrier equipment and qualifications of carrier personnel. The NRC develops performance standards for package designs and reviews package designs for Type B, fissile and large quantity packages. The DOT requires NRC approval to use these packages.[8] The Federal Aviation Administration (FAA), the Interstate Commerce Commission (ICC), and the U.S. Coast Guard also exercise some regulatory authority over the shipment of radioactive materials.

The transportation or packaging for transport of radioactive material is subject to issuance of the appropriate licenses. Applicants for a license to package or to transport radioactive material must show by a combination of analysis and experiments that the proposed package or transport vehicle satisifes all the requirements set forth in the Code of Federal Regulations. The application must describe proposed controls or precautions to be used in the loading, unloading, handling and transport of radioactive material, and the procedures to be followed in the event of an accident or delay in shipment. Inspection and accountability procedures must also be described.

The following Federal Regulations are applicable to the transport of radioactive materials:

- Title 49 Code of Federal Regulations Parts 170-199 (49 CFR 170-199) - Department of Transportation regulations governing the transport of hazardous materials.
- 10 CFR 71 - Nuclear Regulatory Commission regulations governing the packaging and shipment of radioactive materials.
- 14 CFR 103 - Federal Aviation Administration regulations for shipment of radioactive materials by air.
- 47 CFR 146 and 149 - U.S. Coast Guard regulations governing the shipment of radioactive materials by water.

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[8] W. M. Rogers, Jr., "State and Federal Roles in Regulating the Transportation of Radioactive Materials," in Proceedings of the 4th International Symposium on Packaging and Transportation of Radioactive Material, CONF-740901. Miami Beach, FL, September 1974.

- 10 CFR 73 - Nuclear Regulatory Commission regulations for the protection of special nuclear material in transit.

The DOT and NRC regulations are the most important for shipments made during the decommissioning of nuclear facilities.

Although Federal agencies dominate the regulatory process for the transport of radioactive materials, state governments also exercise some control over these shipments. State highway departments regulate gross vehicle weights, vehicular dimensions and other parameters for radioactive shipments just as they do for other kinds of shipments. Currently, about half of the states have adopted the U.S. DOT Hazardous Materials Regulations to cover intrastate shipments. Several states have adopted or proposed additional regulations concerning radioactive materials.[7,9] These include:

- special routing or radioactive shipments
- advance notification for shipments of large quantities of materials
- state inspections of some types of radioactive shipments
- prohibition of certain types of shipments within the states
- prior approval for radioactive shipments
- requirements of exclusive vehicle use for radioactive shipments
- use of pilot vehicles
- speed restrictions for radioactive shipments
- specific hours of movement
- accompaniment of all shipments by radiation monitoring personnel

The variation of regulations between adjacent states can often require special considerations for interstate shipments.

There is a potential conflict between some of the proposed state laws and the provisions of the National Transportation Act of 1974 (Public Law

(7) C. K. Beck, "Intergovernmental Relationships in the Transport of Radioactive Materials," in Proceedings of the Second Annual Legislative Workshop, CONF-730588, Oak Ridge, TN, May 1973.

(9) W. A. Brobst, "The State of State Regulations," in Proceedings of the 4th International Symposium on Packaging and Transportation of Radioactive Material, CONF-840901, Miami Beach, FL, September 1974.

93-633 signed in 1975). This law prohibits the states from adopting laws or regulations more stringent than Federal regulations unless the state regulations improve transportation safety. Even in this case, such rules can be adopted only if they do not unreasonably burden commerce.

A more detailed review of the regulations pertaining to the transport of radioactive material can be found in ERDA-76-43.[10]

Regulations were discussed previously that address the control of effluents from decommissioning activities. Because of the anticipated high radiation sources and contaminated work locations, occupational safety is also of major importance during decommissioning. Radiation protection to workers is regulated by 10 CFR Part 20. Section 20.101 defines the exposure limits. These limits have recently been changed to reflect the operating philosophy of ALARA (As Low As is Reasonably Achievable). This operating philosophy is described in Regulatory Guide 8.8 "Information Relevant to Ensuring That Occupational Radiation Exposure At Nuclear Power Stations Will Be As Low As Reasonably Achievable", and in Regulatory Guide 8.10 "Operating Philosophy For Maintaining Occupational Radiation Exposure As Low As Is Reasonably Achievable." Although not specifically cited for application to decommissioning activities, the guides definitely apply.

Additional information can be found on how to comply with the ALARA concept in the NRC Standard Review Plan, Section 12.1 "Assuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable." Some of the more relevant regulations and guidance cited in this document are given below:

- 10 CFR Part 19, Notices, Instructions and Reports to Workers; Inspections
- 10 CFR Part 20, Standards for Protection Against Radiation
- Regulatory Guide 1.8, Personnel Selection and Training
- Regulatory Guide 1.16, Reporting of Operating Information
- Regulatory Guide 1.39, Housekeeping Requirements for Water Cooled Nuclear Power Plants

(10) U.S. Energy Research and Development Administration, Alternatives for Managing Waste from Reactors and Post-Fission Operations in The LWR Fuel Cycle, ERDA-76-43, Vol. 5, Appendix E, May 1976.

- Regulatory Guide 8.2, Guide for Administrative Practices in Radiation Monitoring
- Regulatory Guide 8.3, Film Badge Performance Criteria
- Regulatory Guide 8.6, Standard Test Procedures for G-M Counters
- Regulatory Guide 8.7, Direct Reading and Indirect Reading Pocket Dosimeters
- Regulatory Guide 8.8, Information Relevant To Ensuring That Occupational Radiation Exposures At Nuclear Power Stations Will Be As Low As Is Reasonably Achievable
- Regulatory Guide 8.9, Acceptable Concepts, Models, Equation and Assumptions for a Bioassay Program
- Regulatory Guide 8.XX, Control of Radioactive Surface Contamination of Material, Equipment and Facilities to be Released for Uncontrolled Use (in preparation)
- ANSI N18.9-1972, Administrative Controls for Nuclear Power Plants, American National Standards Institute (1972)
- ANSI Z88.201969, Procedures for Respiratory Protection, American National Standards Institute (1969)
- USBM-23, Respiratory Protective Services for Use in Atmospheres Containing Radioactive Materials, U.S. Bureau of Mines (1973)

One of the goals of decommissioning a nuclear facility is to make the land available for other uses if desired. In order to release the facility and/or site for unrestricted use, the residual radioactive contamination must be at a level acceptable for public protection. Several attempts have been made to define the permissible levels of residual radioactivity. In Section 8 of this report, a methodology for determining the criterion, based on dose, is applied to the reference PWR. Other major guidance is found in Regulatory Guide 1.86 and the proposed ANSI Standard N328 Control of Radioactive Surface Contamination on Materials, Equipment and Facilities to be Released for

Uncontrolled Use.[a] Previously mentioned guidance that the NRC uses for terminations of byproduct, source, and special nuclear material licenses (similar to Regulatory Guide 1.86) contains a table of "Acceptable Surface Contamination Levels" identical to that in Regulatory Guide 1.86.

Additional guidance can be inferred from information developed for plutonium in soils.[11,12] The EPA is also in the process of finalizing their guidance[13] for the environmental limits of plutonium contamination in soils for unrestricted use.

During decommissioning activities at a PWR, normal industrial (nonradiation related) safety regulations governing occupational work conditions are provided by Title 29 Code of Federal Regulations, Parts 1900 to end (Occupational Safety and Health Administration, Department of Labor).

5.3 THE CONTINUING CARE PHASE

This phase primarily deals with surveillance and maintenance of the facility after it is in a Safe Storage mode. Primary concerns during this period are to assure public safety and safety of the staff maintaining the facility.

During this period, the license may need to be amended consistent with the level of concern for public safety that the facility represents. A possession-only license is likely in the case of a PWR.

If dismantlement follows the continuing care period, the requirements discussed in the active decommissioning phase would apply. Following dismantlement, termination of the license could then occur consistent with the guidance offered by Regulatory Guide 1.86.

(a) The NRC supports the provisions of this standard.

(11) J. W. Healy, A Proposed Interim Standard for Plutonium in Soils, LA-5483-MS, Los Alamos Scientific Laboratory, Los Alamos, NM, January 1974.

(12) A. J. Hazle and Bert L. Crist, Colorado's Plutonium-in-Soil Standard, Colorado Department of Health, Occupational and Radiological Health Division Denver, CO, 1975.

(13) Proposed Federal Radiation Protection Guidance, Federal Register, Vol. 42., No. 230, p. 60956, November 30, 1977.

REFERENCES

1. Federal Register, Vol. 43, No. 49, p. 10371, March 13, 1978.

2. Plan for Reevaluation of NRC Policy on Decommissioning of Nuclear Facilities, NUREG-0436, Office of Standards Development, U.S. Nuclear Regulatory Commission, March 1978.

3. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-75/087 (NTISUB/B/201), Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, September 1975.

4. Environmental Standard Review Plans for the Environmental Review of Construction Permit Application for Nuclear Power Plants (Draft), NUREG-0158 Parts I & II, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, January 1977.

5. J. J. Cohen, et al., Determination of Performance Criteria for High-Level Solidified Nuclear Waste, NUREG-0279, Lawrence Laboratory for USNRC, July 1977.

6. 10 CFR 150, Exemptions and Continued Regulatory Authority in Agreement States Under Section 274.

7. C. K. Beck, "Intergovernmental Relationships in the Transport of Radioactive Materials," in Proceedings of the Second Annual Legislative Workshop, CONF-730588, Oak Ridge, TN, May 1973.

8. W. M. Rogers, Jr., "State and Federal Roles in Regulating the Transportation of Radioactive Materials," in Proceedings of the 4th International Symposium on Packaging and Transportation of Radioactive Material, CONF-740901, Miami Beach, FL, September 1974.

9. W. A. Brobst, "The State of State Regulations," in Proceedings of the 4th International Symposium on Packaging and Transportation of Radioactive Material, CONF-740901, Miami Beach, FL, September 1974.

10. U.S. Energy Research and Development Administration, Alternatives for Managing Wastes from Reactor and Post-Fission Operations in the LWR Fuel Cycle, ERDA 76-43, Vol. 5, Appendix 3, May 1976.

11. J. W. Healy, A Proposed Interim Standard for Plutonium in Soils, LA-5483-MS, Los Alamos Scientific Laboratory, Los Alamos, NM, January 1974.

12. A. J. Hazle and Bert L. Crist, Colorado's Plutonium-in-Soil Standard, Colorado Department of Health, Occupational and Radiological Health Division, Denver, CO, 1975.

13. Proposed Federal Radiation Protection Guidance, Federal Register, Vol. 42, No. 230, p. 60956, November 30, 1977.

6.0 APPROACHES TO FINANCING OF DECOMMISSIONING

The question of how to assure the availability of funds for decommissioning efforts after the facility has ceased to produce any revenue is one that should be addressed at the time of station startup. The cost of the eventual decommissioning of a nuclear power station is as much a part of the cost of the electricity generated at the station as is the cost of fuel, and therefore, should be borne by the consumers of that electricity in an equitable manner. Since the money to pay for decommissioning has to be collected through charges for electricity, and these charges are controlled largely by the public utility commission in each state, it is important to have a reasonable picture of how the utility commissions view the situation.

A recent NRC survey[1] of state public utility commissions found that the preferred approach was to treat the anticipated decommissioning costs as a negative salvage value for purposes of calculating depreciation on the nuclear power station. Several approaches were suggested for handling the monies so collected. These ranged from the establishment of a separate sinking fund with annual payments made from revenues, with the fund independent from and unavailable for use by the utility, to allowing the utility to invest the money in its own new facilities. In this latter case, the utility could then issue securities against those unencumbered facilities as the need for decommissioning funds arose. A strong case was made for this latter approach[2] in terms of minimizing the overall cost to the electricity consumer of providing funds for decommissioning.

In contrast to the concerns voiced by some environmental and consumer advocate organizations[3] there was little concern on the part of the public

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

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

[3] Docket PRM 50-22, Petition for Rulemaking on Decommissioning of Nuclear Power Plants, Federal Register, Vol 42, pp. 40063, August 8, 1977.

utility commissions that the utilities might be unable to pay for decommissioning. As regulated monopolies, the utilities are monitored by the utility commissions to assure that their costs are realistic and legitimate and that their revenues are adequate to provide a reasonable return on their investment to the stockholders (in the case of investor-owned utilities). Thus, it appears there is no likelihood that a utility would go bankrupt, either in the normal course of business, or in the event of an accident that shutdown the reactor station, and that a utility would have no difficulty in paying for decommissioning when needed, regardless of how the monies collected during the operating life of the station are handled..

6.1 METHODS OF PROVIDING FUNDS FOR DECOMMISSIONING

Three basic approaches to providing the funds needed for decommissioning have been identified:

- pay the costs when they are incurred;
- establish a prepaid sinking fund at the time of the reactor commissioning; or
- collect the money as a part of operating revenue and invest these funds in suitable utility-related investments until needed.

The first case (pay when incurred) has the virtue of being the least complicated to administer. However, it is not very equitable in that the utility customers who pay for the decommissioning are probably not the same ones who used the major portion of the electricity generated by the station during its operating lifetime.

The second case (prepaid sinking fund) probably requires the utility to issue revenue bonds against the station to obtain the necessary money, and then to manage the money appropriately to assure adequate funds when needed. A good estimate of decommissioning costs and good projections of inflation rates over the subsequent years of operation are needed to avoid being under- or over-funded when decommissioning begins. This approach is more equitable than the first case since the customers of the utility will be paying for the revenue bonds during the life of the plant. The incremental costs associated with servicing the revenue bonds will be passed on to the consumer.

In the third case, the utility treats the projected decommissioning costs as a negative salvage value for computing depreciation on the station. These monies are collected as a part of operating revenue from the station customers during its operating lifetime. The funds are invested in new capital facilities in the utility system until needed for decommissioning. Then the utility issues securities against these unencumbered facilities to pay for decommissioning. The total cost to the consumer is reduced since the utility does not have to pay servicing costs on borrowed money to build the new facilities. The effective rate of return on the money is larger than could be attained from any investment of similar stability. The projected decommissioning cost (negative salvage value) portion of the depreciation schedule can be adjusted at frequent intervals to compensate for changes in inflation rates, and other variables.

A more detailed discussion of these basic approaches is presented in Appendix D, together with derivations of the appropriate equations for evaluating the costs of each approach and some numerical examples.

6.2 IMPACT OF DECOMMISSIONING ON POWER COSTS

The effect on the cost of electricity of paying for decommissioning through an increased depreciation rate (as described in the third case, above) can be calculated. The monies collected are treated in the same way as if in a sinking fund created with annual payments. For a decommissioning cost of $44 million estimated in dollars at the time of reactor startup, and assumed annual inflation and interest rates of 6% and 8% respectively over an assumed 40 years of station operation, the total dollars collected by the time of reactor shutdown should be about $122 million, or, averaged over the station lifetime, about $3 million per year. For the reference PWR, with an on-line efficiency of 75%, the power sold per year is about 7.4 x 10^9 kWh. Thus, the incremental power cost associated with decommissioning is about 0.04¢/kWh, not a major factor in total power costs.

Expressions for computing the amounts of money required to be collected under various assumed conditions of inflation, interest, and discount rates for the three approaches for financing decommissioning are derived in Appendix D. Example calculations of the total costs and cash flows for each approach for a range of inflation, interest, and discount rates are also presented.

6.3 TAXATION AND PUBLIC ACCEPTANCE CONSIDERATIONS

A factor that could have considerable influence on the choice of mode and time frame for decommissioning is the way that the facility is viewed by the local taxing authorities. For example, will a reactor plant that is in safe storage be taxed at (1) its value as an operating plant, (2) at the value of the unimproved land, or, since the retired plant is a negative asset, (3) at the value of the land and structures, reduced by the expected decommissioning costs? The first alternative, which is unlikely, would force immediate dismantlement of the plant since the accumulated tax costs would, in a few years, exceed the cost of dismantlement. The third approach would reduce the costs to a very nominal amount since the cost of plant decontamination and restoration may well exceed the value of the land and the useable structures. In practice, the tax rate will be negotiated between the local tax assessor and the plant owner, and will likely be somewhere between situations (2) and (3) given above, with the land outside the inner exclusion area probably assessed at a value comparable with adjacent similar property and the property within the inner exclusion area valued at essentially zero. Since the outer areas of the site may not be restricted as to use once the reactor has been decommissioned, that land may be put to a productive use and might pay its own way with regard to property taxes. Therefore, property tax considerations should have little influence on the choice of decommissioning options.

Another more intangible consideration is that of public acceptance of the long-term presence of retired facilities. There is a reasonable probability that once the plant is no longer providing tax revenue and payroll to the community, the structures will represent a perceived hazard to the public and pressures may mount for the removal of the retired structures. While it is beyond the scope of these studies to evaluate the likelihood of this concern today, the plant owner should sample local public opinion on this question well in advance of setting his plans for decommissioning, to avoid unpleasant surprises later on.

REFERENCES

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

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

3. Docket PRM 50-22, Petition for Rulemaking on Decommissioning of Nuclear Power Plant, Federal Register, Vol. 42, pp. 40063, August 8, 1977.

7.0 CHARACTERISTICS OF THE REFERENCE PWR POWER STATION

This section briefly describes the characteristics of the reference PWR power station used as the basis for this decommissioning study. Summaries of the detailed information developed during this study are presented. Included are a description of the reference site, a description of the reference facility, estimates of the inventories of radionuclides on the station, estimates of radiation dose rates throughout the station, and an estimate of the chemical inventory in the station at the time of final reactor shutdown.

The information presented is typical of present generation, large PWR power stations. While some details may vary from station to station, these differences are not expected to have any major impacts on the results of the study.

Detailed characteristics of the reference PWR power station necessary to develop work plans, to estimate manpower requirements, radiation exposure, waste disposal volumes, and costs for decommissioning the station are contained in appendices. Detailed descriptions of the station site and facilities are presented in Appendix B, and Appendix A, respectively. Data for estimating inventories of radionuclides are presented in Appendix C. Estimates of radiation dose rates throughout the reference PWR are also presented in Appendix C.

7.1 THE REFERENCE SITE

A reference environment was developed to aid in assessing the public safety and potential environmental effects of conceptually decommissioning a LWR by various alternative methods. The meteorology parameters and population distributions used were taken from the ALAP Study[1] for the river site in the year 2000. The ecological information was derived from

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

the environment of one operating nuclear reactor.[2] The remainder of the information was obtained from a variety of sources and is felt to be representative of potential sites for nuclear reactor facilities in the midwestern or south mideastern United States. This generic site decription was developed for use in a series of studies examining decommissioning of nuclear fuel cycle facilities, and the detailed supporting information relating to this abbreviated description is found in Appendix B.

Individual features of a specific site will vary from those of a generic site for any specified nuclear fuel cycle facility. However, it is believed that use of a generic site will result in a more meaningful overall analysis of potential impacts associated with most nuclear fuel cycle facilities. Site specific assessments will be required for the safety analysis and the Environmental Report submitted with the request for license modification prior to decommissioning the facility.

The generic site occupies 4.7 square kilometers (1160 acres) in a rectangular shape of 2 kilometers (1.24 miles) by 2.35 kilometers (1.46 miles). A river of moderate size runs through one corner of the site.

The site is located in a rural area that has relatively low population density. Higher population densities are located at distances 16 to 64 kilometers (10 to 40 miles) away, and gradually reducing population densities are encountered out to 177 kilometers (110 miles). The closest moderately large city, population 40,000, is about 32 kilometers (20 miles) distant. The closest large city, population 1,800,000, is about 48 kilometers (30 miles) away. The total population in a radius of 80 kilometers (50 miles) is 3.52 million.

The plant facilities are located inside a 0.10-square kilometer (26 acre) fenced portion of the site. The minimum distance from the point of plant airborne releases to the outer site boundary is one kilometer. In most of the surrounding area, about 80% of the land is used for farming.

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

The relatively clean river flowing through the site has an average flow rate of 1420 m^3/sec. The river is used for irrigation, fishing, boating and other aquatic recreational activities, and is a source of drinking water for larger communities. Large supplies of flowing groundwater exist at modest depths around the site. This water is widely used for drinking and irrigation.

The reference site occupies a relatively flat terrace that has a low bluff forming one bank of the river. Biologically young soils cover the old basement rocks in the area. This site is in a relatively passive seismic area and is located at an elevation above the estimated maximum probable flow level.

The climate at the site is typical for internal continental areas. It has wide temperature variations and moderate precipitation. Meteorology used in this study is an average taken from 16 nuclear reactor sites, with annual average $\bar{\chi}/Q$ (atmospheric dispersion factor) at the closest site boundary of about 5×10^{-8} sec/m^3.(1)

Less than 20% of the land around the site is covered with pristine vegetation. The original vegetation was primarily a climax deciduous forest. A number of migratory birds are present in the area, as well as some annual birds. A few of these are considered to be rare, endangered, or threatened by extinction. A number of mammals occupy the general area.

The site is slightly contaminated with radioactive material as a result of deposition from the release of normal operating effluents over the 40-year plant operating life. It is assumed that any accidental releases of radioactive material will be cleaned up immediately following the event. Estimates of the maximum site contamination levels at the time of plant shut-down are given in Section 7.3.2. The site contamination estimates compare the effect of two annual atmospheric release cases, and their 40 year ground depositions. The assumptions and calculational methods for relating the normal plant effluents to site surface contamination can be found in Section 7.3.2.

7.2 THE REFERENCE FACILITY

The reference reactor power plant described in this section is a 3500 MW(t) [1175 MW(e)] pressurized water reactor of the Westinghouse design, specifically the TROJAN Nuclear Plant at Rainier, Oregon, operated by the Portland General Electric Company. The information presented herein was obtained from the TROJAN FSAR,[3] Westinghouse RESAR,[4] SNUPPS PSAR,[5] and from detailed construction drawings, photographs and other data furnished by personnel of the Portland General Electric Company.

The principal systems, components, and structures are described briefly in the sections that follow. More detailed information can be found in Appendix A.

7.2.1 Nuclear Steam Supply System

The nuclear steam supply system is illustrated and described in the functional schematic diagram in Figure 7.2-1. The principal components of interest are the reactor vessel which contains the fuel and coolant and the reactor coolant system (RCS) which transfers the heat from the fuel to the secondary coolant system via the Steam Generator heat exchangers where steam is produced for use in the turbine generator.

7.2.1.1 Reactor Vessel and Internals

The reactor vessel is a right circular cylinder with a welded hemispheric bottom and a removable hemispheric top, as illustrated in Figure 7.2-2. The vessel is constructed of carbon steel, about 0.216 meters in thickness and is clad on the inside with stainless steel or Inconel about 4 mm in thickness. The approximate dimensions of the vessel are 12.6 m high, 4.6 m outer diameter. The vessel weighs nearly 400 Mg.

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

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

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

PRESSURIZED WATER REACTOR (PWR)

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

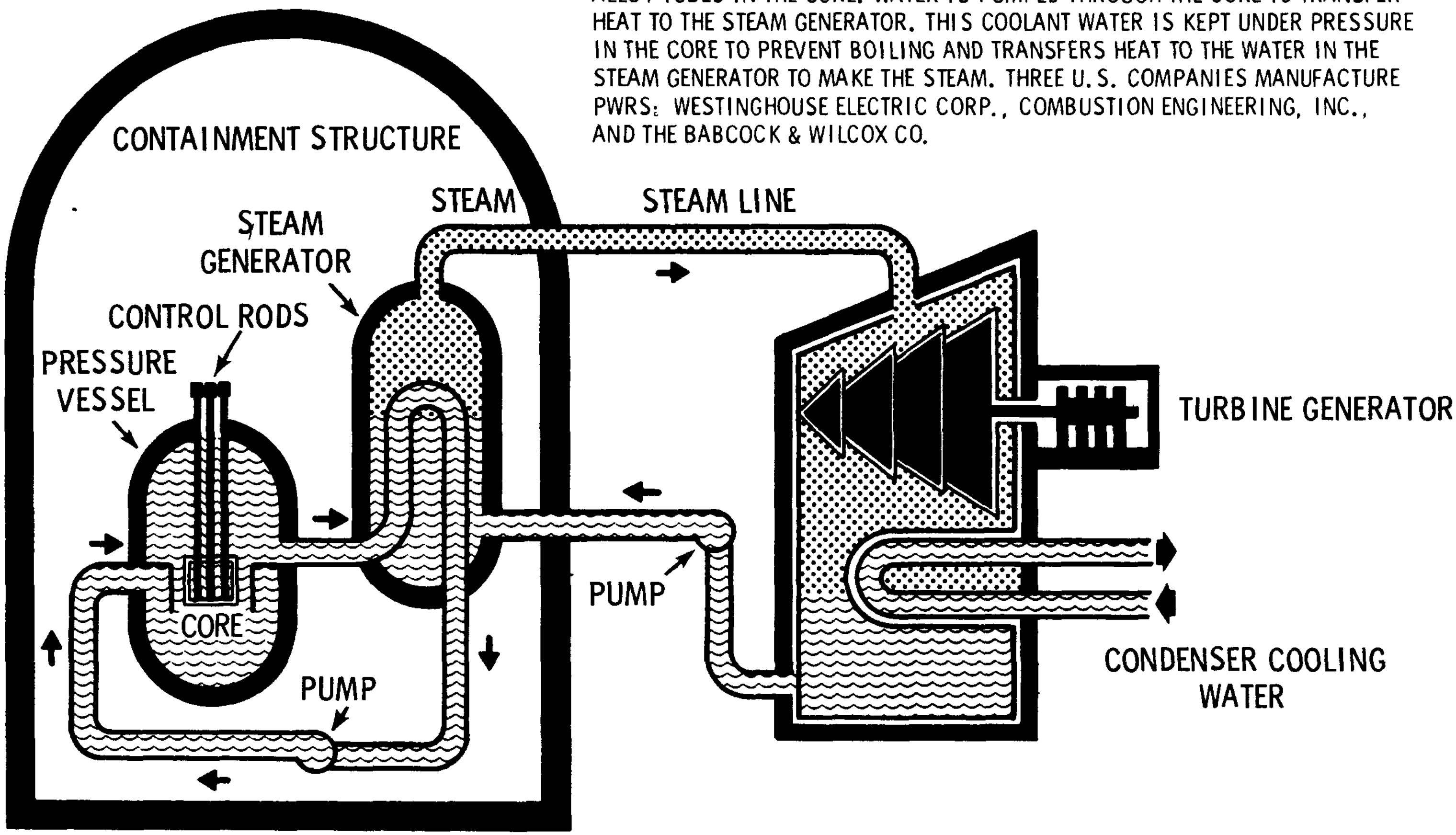


FIGURE 7.2-1. Pressurized Water Reactor (PWR)

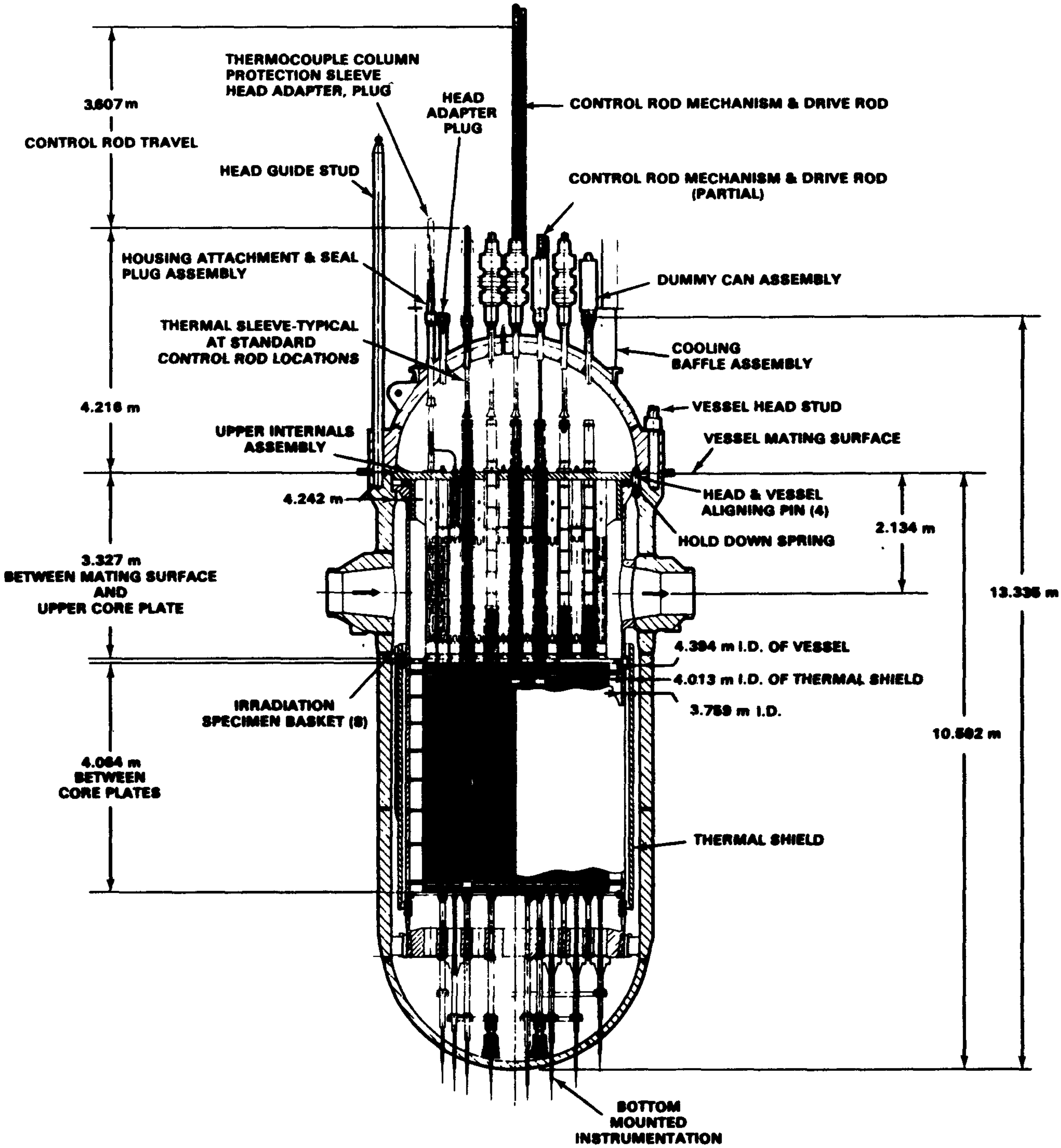


FIGURE 7.2-2. Reactor Vessel Internals

The vessel internal structures support and constrain the fuel assemblies, direct coolant flow, guide in-core instrumentation, and provide some neutron shielding. The principal components are: the lower core support assembly which includes the core barrel and shroud, with neutron shield pads and the lower core plate and supporting structure; and the upper core support and in-core instrumentation support assemblies. These structures are made of 304 stainless steel and have a total weight of about 190 Mg.

7.2.1.2 Reactor Coolant System

The reactor coolant system schematically illustrated in Figure 7.2-3 consists of four loops for transferring heat from the reactor to the secondary coolant system. Each loop contains a U-tube steam generator, a reactor coolant pump, and connecting piping. Each steam generator, illustrated in Figure 7.2-4, is about 20.6 m in height, 3.4 m in diameter, weighs about 312 Mg, and contains nearly 3400 Inconel U-tubes. The interior surfaces exposed to the reactor coolant are clad with austenitic stainless steel or Inconel.

Each coolant pump is a vertical, single stage, centrifugal, shaft seal pump capable of moving 335 cubic meters per minute. Its overall height is about 8.7 m and it weighs about 85.4 Mg. An air-cooled electric motor, which uses about 4.5 MW of electrical energy, drives each pump.

A total of 81 m of large diameter ($\sim$0.7 m I.D) piping connects the four loops of the reactor coolant system to the reactor vessel, This piping has wall thicknesses in the 59-66 mm range, and weighs slightly over 100 Mg.

Another major component is the pressurizer which controls the level of pressure on the reactor coolant. The pressurizer, illustrated in Figure 7.2-5, is a vertical, cylindrical vessel with hemispheric ends, made of carbon steel and clad on the inside with austenitic stainless steel. It is about 16.1 m in height and 2.3 m in outside diameter, and weighs about 88.7 Mg.

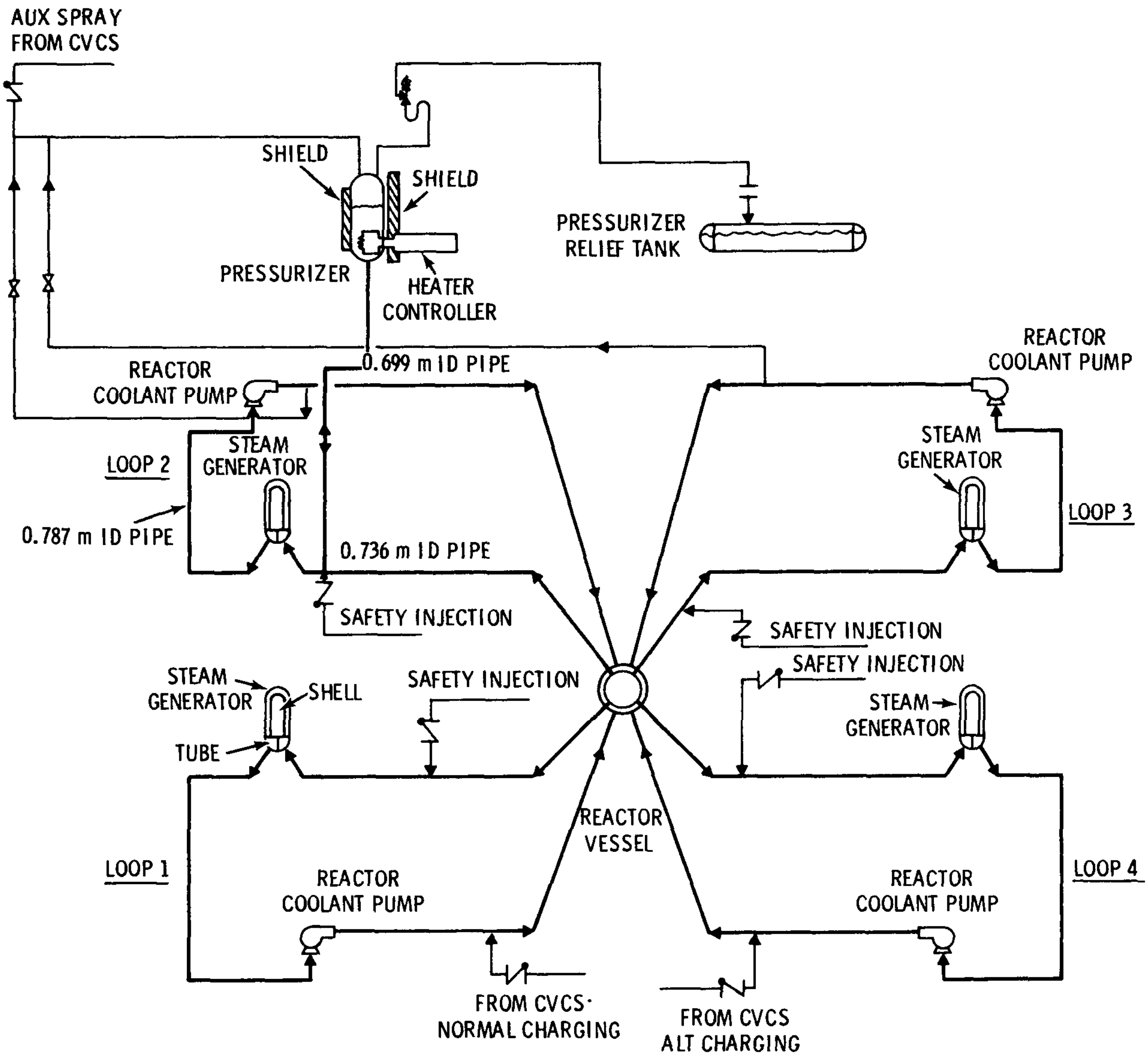


FIGURE 7.2-3. Reactor Coolant System

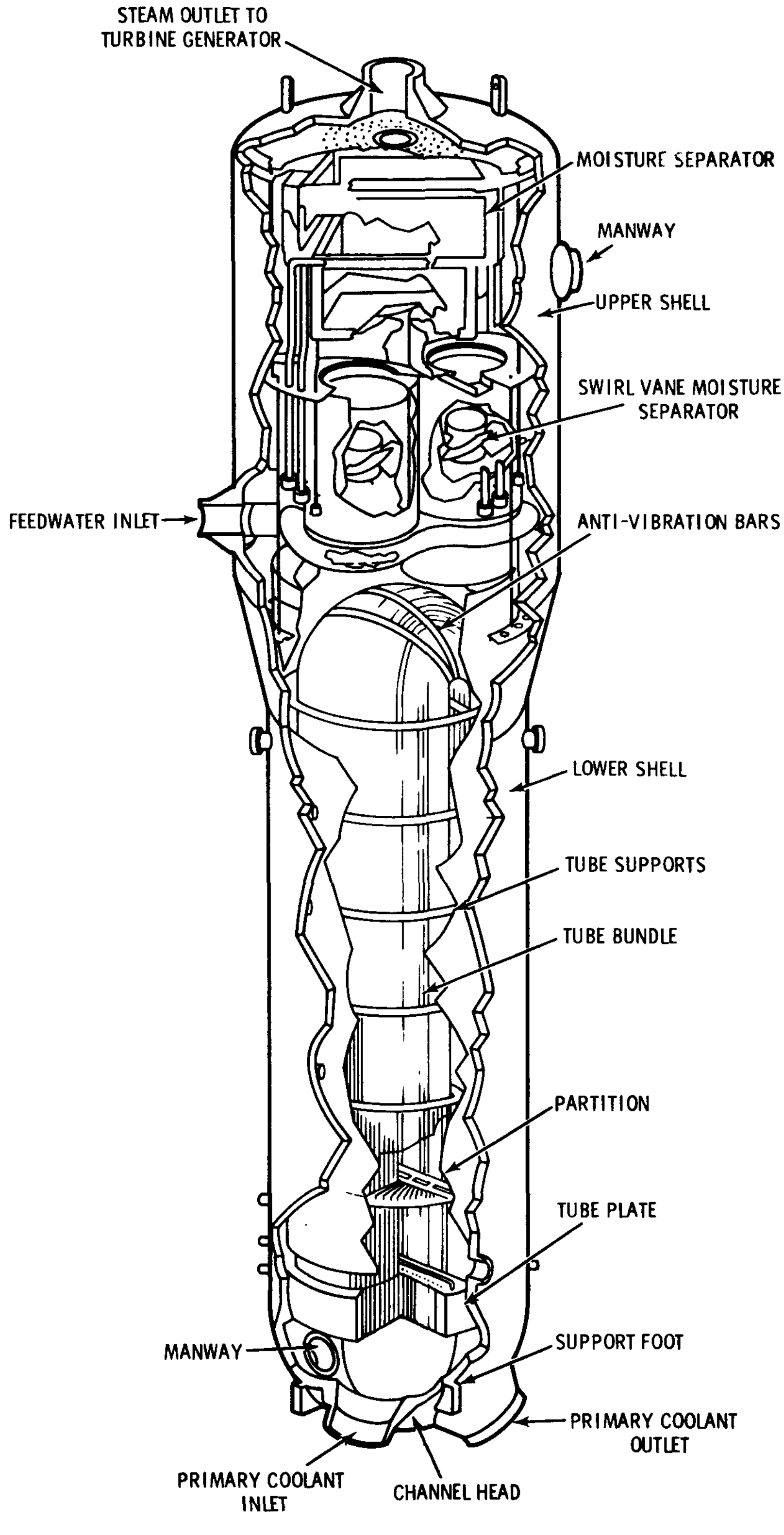


FIGURE 7.2-4. Steam Generator

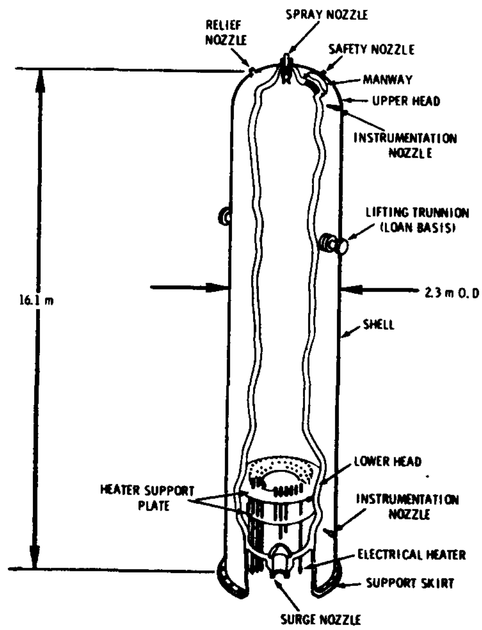


FIGURE 7.2-5. Cutaway of Pressurizer

7.2.2 General Plant Arrangements

The arrangement of the structrues on the plant site is illustrated in Figure 7.2-6. The structures of primary interest during decommissioning are those which contain radioactive material, i.e., the Containment, Fuel, Auxiliary, and Control Buildings. Other structures are primarily a demolition problem. All of the structures are discussed briefly in the following sections, with more detailed descriptions given in Appendix B.

7.2.2.1 Containment Building

The nuclear steam supply system is located within the Containment Building. This structure is a right circular cylinder with a hemispheric top and a flat base, as illustrated in Figure 7.2-7. It is constructed of reinforced concrete, with post-tensioned tendons in the cylindrical walls and dome, and lined with a welded steel skin. Major interior structures include the biological shield, the steam generator and pressurizer cubicles, and the refueling cavity. The Containment Building is about 64 m in height and about 22-1/2 m in diameter.

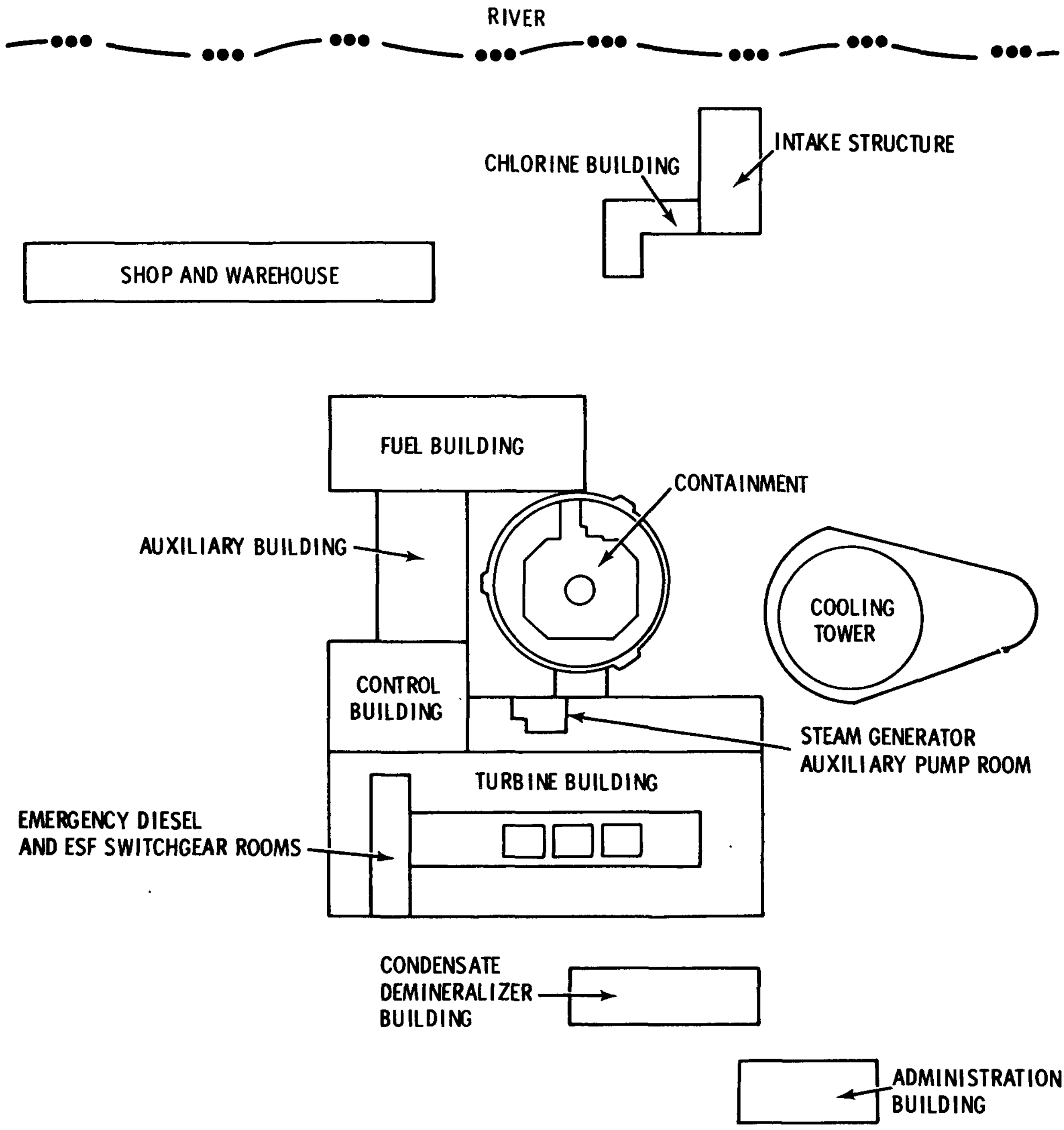
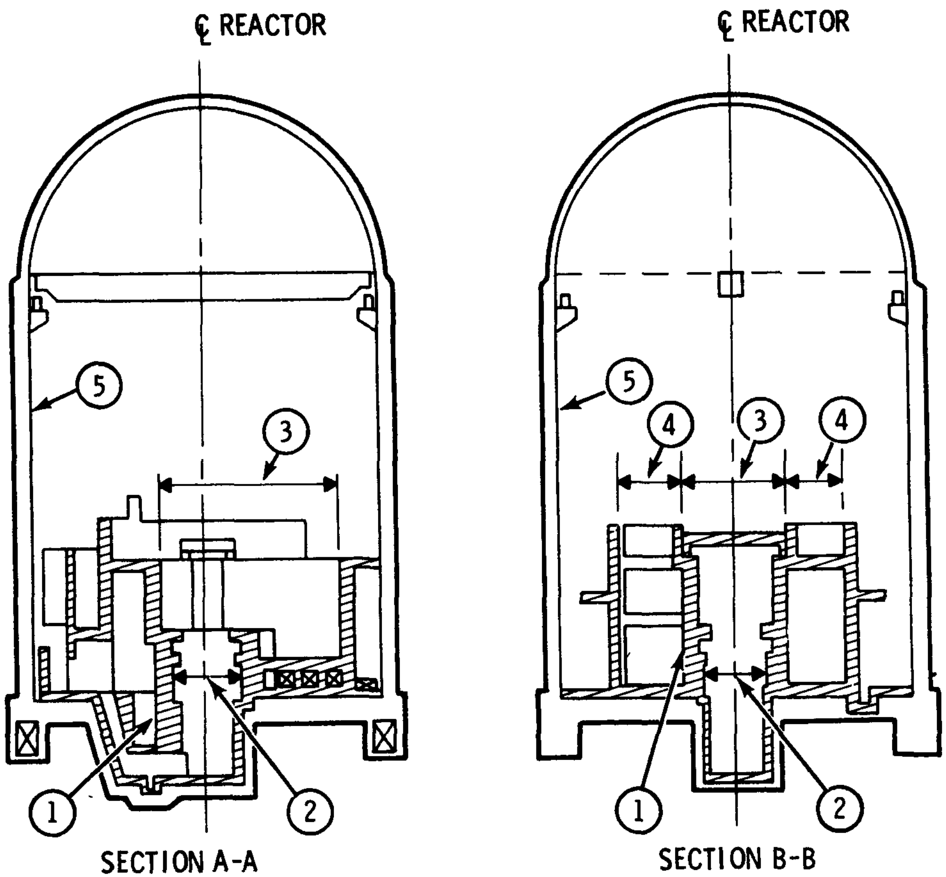
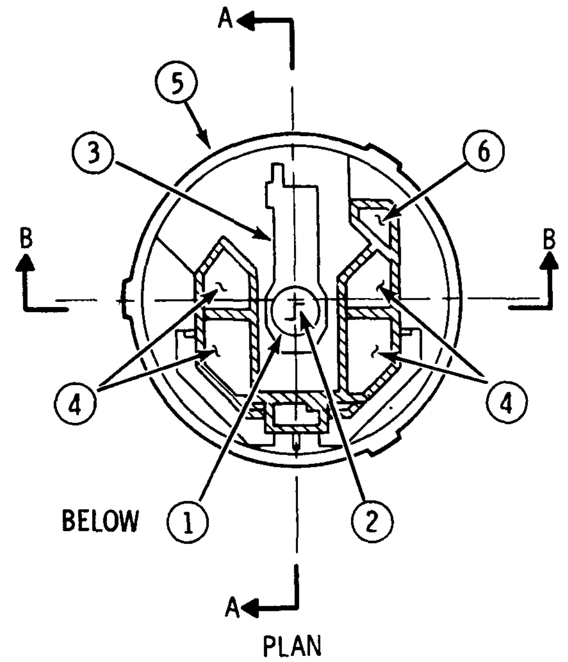


FIGURE 7.2-6. Typical Plant Layout





	COMPARTMENT
1	BIOLOGICAL SHIELD
2	REACTOR CAVITY
3	REFUELING CAVITY
4	STEAM GENERATOR
5	REACTOR CONTAINMENT
6	PRESSURIZER VESSEL

FIGURE 7.2-7. Containment Building

7.2.2.2 Fuel Building

The Fuel Building is a steel frame and reinforced concrete structure with four floors. It is approximately 27 m in height and about 54 m by 19 m in lateral dimension. The spent fuel storage pool and its cooling system, much of the Chemical and Volume Control System and the solid radioactive waste handling equipment are located in the Fuel Building.

7.2.2.3 Auxiliary Building

The Auxiliary Building is a steel and reinforced concrete structure, with two floors below grade and four floors above grade. It is approximately 30 m in overall height and has lateral dimensions of about 35 m by 19 m. Principal systems contained in the Auxiliary Building include the liquid radioactive waste treatment systems, the filter and ion exchanger vaults, waste gas treatment system, and the ventilation equipment for the Containment, Fuel and Auxiliary Buildings.

7.2.2.4 Control Building

The Control Building as a steel and reinforced concrete structure with four floors above grade. It is structurally connected to the Auxiliary Building and is approximately 18 m in height and has lateral dimensions of about 31 m by 24 m. The principal contents of the Control Building are the reactor control room, process control laboratories, counting rooms, and personnel facilities.

7.2.2.5 Turbine Building

The Turbine Building is framed with structural steel and has a reinforced concrete slab floor with the turbine pedestals poured into the grade level floor and with two operating floors above. The structure has lateral dimensions of about 95 m by 49 m and is about 33 m in height. The principal systems contained in the Turbine Building are the turbine generator, condensers, associated power production equipment, steam generator auxiliary pumps, and the emergency diesel generator units.

7.2.2.6 Cooling Tower

The hyperbolic natural draft cooling tower is a reinforced concrete structure with a height of about 152 m and a diameter at the base of about 119 m. About 5.2 million gallons of water are contained in the reservoir beneath the cooling fins.

7.2.2.7 Other Structures

The remaining structures on the reference site are of conventional construction. They are assumed to be uncontaminated with radioactive materials for purposes of this study. Brief descriptions of these structures are presented below.

Chlorine Building and Intake Structure

The Chlorine Building is a steel-framed structure on a concrete slab and contains the chlorination equipment for treatment of water coming from and being discharged into the river. The intake structure is a reinforced concrete structure which houses the raw water pumps and the intake screens and is located at the edge of the river.

Condensate Demineralizer Building

The Condensate Demineralizer Building has two levels of reinforced concrete below grade and one level of structural steel above grade. Its lateral dimensions are about 43 m by 10 m and is about 14 m in overall height. The principal systems and components contained in the Condensate Demineralizer Building are the condensate demineralizer ion exchangers and facilities for disposal of expended resins.

Shop and Warehouse

This building is a single story, steel frame structure on a concrete slab. Its dimensions are about 90 m in length and about 13 m in width. Facilities contained are the general machine shop, paint shop, warehouse, offices, lockers and lunchroom.

Administration Building

This building is a two story, steel frame structure. The general administrative offices and the plant security control station are located therein.

7.3 ESTIMATED INVENTORIES OF RADIONUCLIDES AT THE REFERENCE PWR

Information about the levels and nature of the radioactive contamination present at the reference PWR at the time of decommissioning is needed to evaluate the impacts of decommissioning on occupational and public safety. The residual radionuclide inventories are also used in determining the decontamination levels required for unrestricted use which directly influence the costs and methods of decommissioning.

The following sections contain summaries of the radionuclide inventories expected to be found in the reference PWR and on its site after forty years of normal operation. A more detailed discussion of both the methodology used in determining the neutron activation in the Reactor Containment Building and the methodology used in determining the corrosion product deposition on piping internal surfaces is contained in Appendix C. Annual atmospheric releases of radionuclides from the operating PWR are derived from calculated and reported releases to illustrate the wide variation between these two sources of information. These releases are used in calculations to obtain estimates of the accumulation of radionuclides on the site from forty years of normal PWR operation.

7.3.1 Accumulated Radionuclides within the Reference PWR

Significant quantities of radionuclides remain in a nuclear power station at the time of final reactor shutdown even after the irradiated fuel has been removed. Neutron-activated structural materials in and around the reactor pressure vessel contain large, relatively immobile quantities of radioactivity. Radioactive corrosion products and fission products from failed fuel, which are transported throughout the station by the reactor coolant streams, are the principal contributors to the more mobile radioactive contamination on piping, floors, and pool surfaces.

7.3.1.1 Neutron-Activated Reactor Components and Structural Material

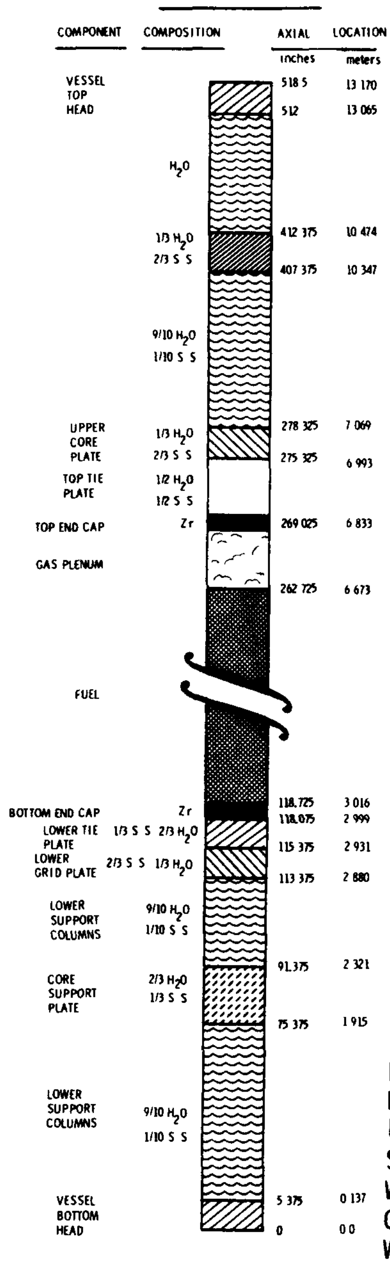
Production of radioactive reactor components and structural materials by neutron activation is a normal result of reactor operation. The concentration of a particular radionuclide in a given location in the reactor depends upon the neutron flux level at that location, the duration of the exposure to the neutron flux, the concentration of the parent isotope, and the cross section of that isotope for the production of the radioactive species. The concentrations of radionuclides present in the reactor vessel and its internal structures and in the surrounding shielding enclosure were calculated for the reference PWR, assuming 30 effective full power years (EFPY), equivalent to 40 calendar years at 75% of full power operation. Summary descriptions of the calculations and the results therefrom are presented in the following sections, with more details given in Appendix C.

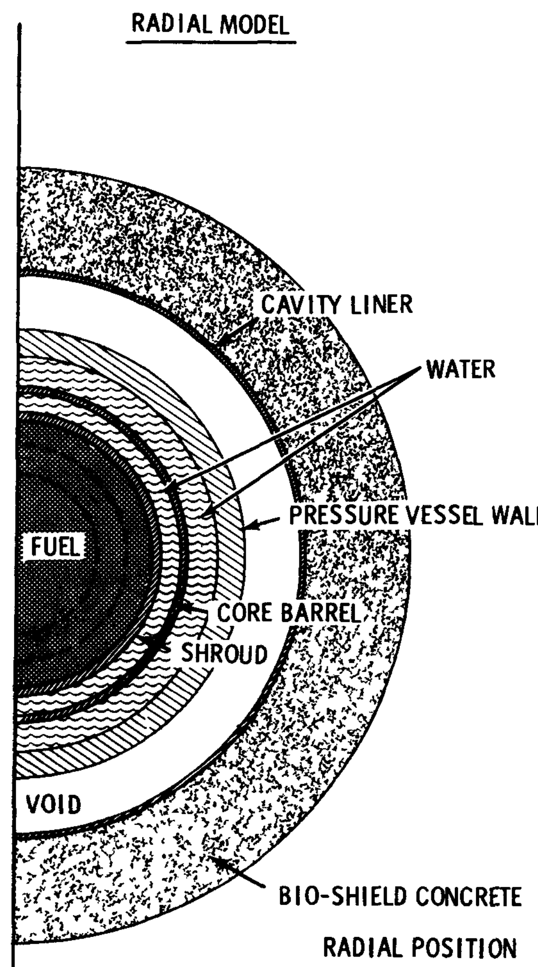
Neutron flux levels throughout the reference PWR pressure vessel and surrounding shield were calculated using the multi-energy group transport theory code, ANISN.[(6)] The physical models used in calculations are shown in Figure 7.3-1. The calculated neutron fluxes, averaged over the various reactor components, are listed in Table 7.3-1.

The calculated neutron fluxes were used in a series of calculations with the ORIGEN[(7)] code to compute the production, decay, and removal by neutron capture of each of the radioactive species produced in the reactor vessel components, for various periods of time, up to 30 EFPY of reactor operation. The buildup of a selected group of the longer-lived radionuclides is illustrated in Figure 7.3-2. The activity of each of the radionuclides, in Ci/m^3, is normalized to unity at 30 EFPY of operation for this illustration. It is seen that the shorter-lived radionuclides, ^{55}Fe and ^{60}Co, reach an equilibrium state in less than thirty years of reactor operation. The power reactors that have been decommissioned in the past operated for relatively short time periods. The Elk River Reactor, for example, had operated for the equivalent of only 2.5 EFPY when it was dismantled. Thus, the concentrations of the longer-lived radionuclides in the Elk River reactor were quite small compared to the concentrations that will be present in a large PWR after 30 EFPY of operation.

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

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





REGION	RADIAL POSITION INNER BOUNDARY inches	meters
FUEL (~28,000 MWD/MTHM)	0.0	0.0
FUEL (~17,000 MWD/MTHM)	38.21	0.971
FUEL (~ 6,000 MWD/MTHM)	54.04	1.373
SHROUD	66.36	1.686
WATER	67.48	1.714
CORE BARREL	74.00	1.880
WATER	75.50	1.918
VESSEL WALL	86.50	2.197
VOID	95.00	2.413
CAVITY LINER	121.00	3.073
BIO-SHIELD CONCRETE	121.25	3.080
OUTER EDGE AND SHIELD	198.0	5.029

FIGURE 7.3-1. Models for ANISN Calculations

TABLE 7.3-1. Calculated Neutron Flux Levels in Reactor Components[a] (neutrons/cm^2/sec)

				Vessel		Bio-Shield	
Neutron Energy Group	Shroud	Core Barrel	Thermal Shields	Clad	Wall	Cavity Liner	Concrete
Thermal Flux (0-0.632 eV)	9.36×10^{12}	1.33×10^{12}	2.09×10^{11}	2.8×10^{10}	2.79×10^{9}	1.70×10^{9}	(b)
Epithermal Flux (0.632 eV - 1 MeV)	3.86×10^{13}	1.42×10^{12}	4.58×10^{10}	4.09×10^{10}	3.05×10^{10}	1.33×10^{10}	(b)
Fast (1 - 14.92 MeV)	7.64×10^{12}	4.11×10^{11}	2.08×10^{10}	1.33×10^{10}	3.83×10^{9}	3.26×10^{8}	(b)

(a) Values listed are from the radial ANISN calculation, and are averaged over the length of the fuel in the core. The axial peak-to-average ratio in the core is 1.22. The listed values are also averaged over the thickness of the component, thus including local self-shielding effects.

(b) Values range from $\sim 4 \times 10^9$ near the inner surface to essentially zero at the outer boundary. A volume-weighted average value would be misleading.

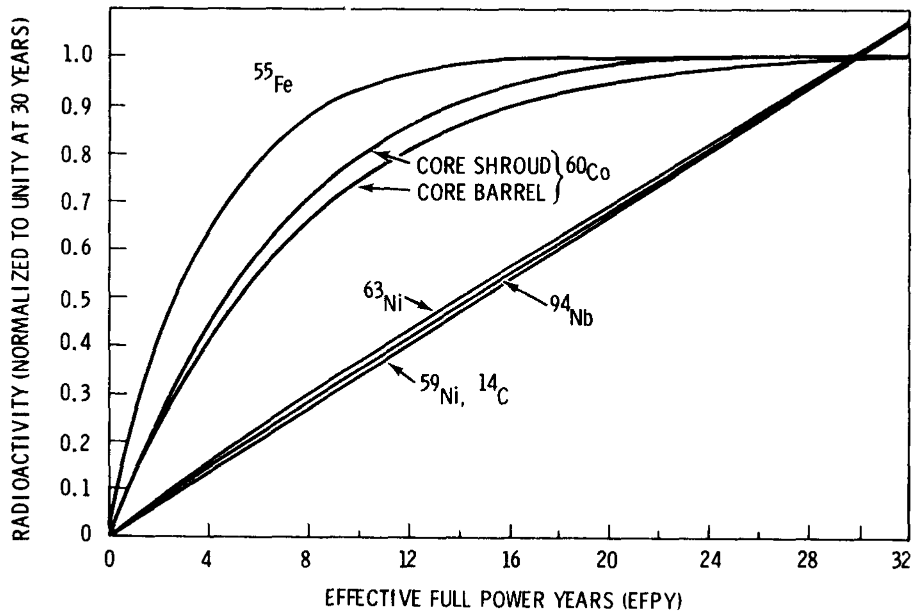


TABLE 7.3-2. Buildup of Activation Products in PWR Vessel Internals as a Function of EFPY

The specific activities at the time of final reactor shutdown of the principal radionuclides of interest in decommissioning are listed in Table 7.3-2 for each of the major reactor components. Radionuclides having half-lives shorter than thirty-five days have been omitted from this tabulation. The upper and lower values for ^{60}Co activity listed in Table 7.3-2 are based on assumed initial concentrations of the ^{59}Co impurity in the 304 stainless steel components ranging from 0.05 to 0.15 percent by weight in the alloy, and in the SA533 carbon steel pressure vessel ranging from 0.006 to 0.012 percent by weight. The ^{93}Nb trace impurity, parent of the ^{94}Nb radionuclide, was assumed to have a concentration in the 304 stainless steel of 0.016 percent by weight in the alloy. A complete listing of the isotopic composition of the stainless steel, carbon steel and concrete shield assumed for these calculations is given in Appendix C. A total of over 4.8 million curies of radioactivity was calculated to be present in the activated reactor vessel and components at the time of final reactor shutdown. The estimated fractional radionuclide inventories for stainless steel and carbon steel reactor components are shown in Tables 7.3-3 and 7.3-4. These inventories

TABLE 7.3-2. Radioactivity Levels in Major Activated Reactor Components at Time of Reactor Shutdown

Isotope	Half-Life	Core Mid-Plane Radioactivities (Ci/m^3)						
		Shroud	Lower 4.72 m of Core Barrel	Thermal Shields	Vessel Inner Cladding	Lower 5.02 m of Vessel Wall	Upper Grid Plant[a]	Lower Grid Plate[a]
^{95}Nb	35 day	2.0×10^3	7.6×10^0	3.5×10^0	5.6×10^{-3}	1.7×10^{-3}		
^{59}Fe	45 day	4.6×10^4	4.4×10^3	2.0×10^3	1.0×10^2	2.7×10^1		
^{58}Co	72 day	1.5×10^5	1.0×10^4	4.6×10^3	3.3×10^2	6.6×10^0		
^{95}Zr	65 day	1.1×10^{-1}	6.2×10^{-3}	2.9×10^{-3}	2.0×10^{-4}	7.2×10^{-4}		
^{65}Zn	245 day	1.2×10^2	1.1×10^0	5.0×10^{-1}	6.7×10^{-4}	3.5×10^{-5}		
^{54}Mn	~300 day	6.8×10^4	3.7×10^3	1.7×10^3	1.2×10^2	4.7×10^1		
^{55}Fe	2.7 yr	1.3×10^6	1.5×10^5	6.7×10^4	3.5×10^3	7.2×10^2		
^{60}Co[b] upper	5.27 yr	9.6×10^5	9.3×10^4	4.7×10^4	2.5×10^3	7.5×10^1		
lower		3.2×10^5	3.1×10^4	1.6×10^4	8.2×10^2	2.5×10^1		
^{63}Ni	~100 yr	1.2×10^5	1.5×10^4	6.8×10^3	3.6×10^2	3.8×10^0		
^{93}Mo	~3500 yr	3.6×10^{-1}	5.2×10^{-2}	2.4×10^{-2}	1.2×10^{-3}	1.3×10^{-3}		
^{14}C	~5,750 yr	1.5×10^2	1.8×10^1	8.3×10^0	4.0×10^{-1}	1.9×10^{-2}		
^{94}Nb	~20,000 yr	5.4×10^0	2.6×10^{-1}	1.2×10^{-1}	9.5×10^{-3}	--		
^{59}Ni	~80,000 yr	7.4×10^2	1.3×10^2	5.0×10^1	3.0×10^0	3.2×10^{-2}		
Sum (Ci/m^3)		2.97×10^6	3.07×10^5	1.45×10^5	7.73×10^3	9.04×10^2	2.97×10^6	2.97×10^6
Average/Peak		0.755	0.637	0.778	0.637	0.637	0.003×4.74[c]	0.08×4.74[c]
Ci/kg[d]		2.787×10^2	2.433×10^1	1.403×10^1	7.621×10^{-1}	7.164×10^{-2}	5.254×10^0	1.403×10^2
Weight of Material (kg)		12,312	26,783	10,413	2,074	245,582	4,627	3.946
Sum (Ci)[e]		3.431×10^6	6.516×10^5	1.461×10^5	1.581×10^3	1.759×10^4	2.431×10^4	5.534×10^5
TOTAL - Radioactivity					4.826×10^6 Curies 1.786×10^{17} becquerels			

(a) Normalized to shroud

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

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

(d) Conversion factor assumes stainless steel density of 8.038×10^3 kg/m^3 (0.29 lb/in.^{3}).

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

TABLE 7.3-3. Stainless Steel Activation Products-Core Shroud (Fractional Activity Normalized at Reactor Shutdown)

Radionuclide	Activity Composition at Decay Times of: Shutdown	10 Years	30 Years	50 Years	100 Years
^{54}Mn	2.6×10^{-2}	7.7×10^{-6}	--[a]	--	--
^{55}Fe	4.9×10^{-1}	3.7×10^{-2}	2.2×10^{-4}	1.3×10^{-6}	--
^{59}Fe	1.7×10^{-2}	--	--	--	--
^{58}Co	5.7×10^{-2}	--	--	--	--
^{60}Co	3.6×10^{-1}	9.6×10^{-2}	6.9×10^{-3}	5.0×10^{-4}	6.8×10^{-7}
^{59}Ni	2.8×10^{-4}	2.8×10^{-4}	2.8×10^{-4}	2.8×10^{-4}	2.8×10^{-4}
^{63}Ni	4.5×10^{-2}	4.2×10^{-2}	3.6×10^{-2}	3.2×10^{-2}	2.2×10^{-2}
^{65}Zn	4.5×10^{-5}	1.3×10^{-9}	--	--	--
^{93}Mo	1.4×10^{-7}	1.4×10^{-7}	1.4×10^{-7}	1.4×10^{-7}	1.4×10^{-7}
^{94}Nb	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}
TOTAL	1.0	1.8×10^{-1}	4.3×10^{-2}	3.3×10^{-2}	2.2×10^{-2}

[a] A dash indicates values less than 1×10^{-10}.

TABLE 7.3-4. Carbon Steel Activation Products-Lower Vessel Wall (Fractional Activity Normalized at Reactor Shutdown)

Radionuclide	Activity Composition at Decay Times of: Shutdown	10 Years	30 Years	50 Years	100 Years
^{54}Mn	5.3×10^{-2}	1.6×10^{-5}	--[a]	--	--
^{55}Fe	8.2×10^{-1}	6.3×10^{-2}	3.7×10^{-4}	2.1×10^{-6}	--
^{59}Fe	3.1×10^{-2}	--	--	--	--
^{58}Co	7.5×10^{-3}	--	--	--	--
^{60}Co	8.5×10^{-2}	2.3×10^{-2}	1.6×10^{-3}	1.2×10^{-4}	1.6×10^{-7}
^{59}Ni	3.6×10^{-5}	3.6×10^{-5}	3.6×10^{-5}	3.6×10^{-5}	3.6×10^{-5}
^{63}Ni	4.3×10^{-3}	4.0×10^{-3}	3.5×10^{-3}	3.0×10^{-3}	2.1×10^{-3}
^{93}Mo	1.5×10^{-6}	1.5×10^{-6}	1.5×10^{-6}	1.5×10^{-6}	1.5×10^{-6}
TOTAL	1.0	9.0×10^{-2}	5.5×10^{-3}	3.2×10^{-3}	2.1×10^{-3}

[a] A dash indicates values less than 1×10^{-10}.

are normalized to the total activity present at reactor shutdown, from data in Table 7.3-2. Radioactive decay of these isotopes is also shown in Tables 7.3-3 and 7.3-4 for time periods of 10, 30, 50, and 100 years after reactor shutdown.

The radionuclide inventory in the concrete biological shield is difficult to define with any great precision since the actual initial composition, particularly with regard to elements present in trace quantities, is not well known. Calculations of activation products in the bio-shield, using the constitutents listed in Table C.1-2 of Appendix C, result in the listing given in Table 7.3-5. No rare earths were included in the calculation due to lack of quantitative data on their probable initial concentrations. The levels of radioactivity (Ci/m^3) are those calculated to be present at final reactor shutdown.

An analysis of a bio-shield core sample from the Elk River Reactor was performed using gamma-ray spectroscopy. Gamma rays from ^{22}Na, ^{60}Co and ^{152}Eu were found to be present.[(8)] These measured levels of radioactivity, corrected for decay since reactor shutdown, are also listed in Table 7.3-5, for comparison with the calculated values. Apparently, the concrete contained some rare earths as trace constitutents, as evidenced by the presence of ^{152}Eu. The neutron capture cross section of ^{151}Eu is large, about 8,700 barns. Thus, a trace constitutent of ^{151}Eu could result in a significant production of ^{152}Eu over the forty years of reactor operation. A review of the (n, γ) reactions in other rare earth isotopes found only one other reaction likely to be of interest, ^{153}Eu (n, γ) ^{154}Eu. The neutron capture cross section of ^{153}Eu is about 320 barns and the activated ^{154}Eu has a half-life of about sixteen years. It is surprising that ^{154}Eu was not also found in the Elk River measurements.

It appears that to obtain a definitive estimate of the radionuclide inventory in the concrete bio-shield, either a quantitative analysis for trace elements in the concrete is needed or a core sample from the shield will have to be taken after final reactor shutdown and analyzed using gamma-ray spectroscopy. Based on the calculated radioactivity levels in Table 7.3-5,

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

an estimated 1,200 curies of radioactivity will be present in the concrete bio-shield at final reactor shutdown.

TABLE 7.3-5. Radioactivity Levels at the Inner Surface of the Activated Biological Shield at Reactor Shutdown

			Radioactivity (Ci/m^3)	
Radionuclide	Half-Life	Principal Radiation Emitted	Reference PWR (calculated)	Elk River Reactor (measured)
^{3}H	12.4 yr	β	2.29×10^{-5}	
^{14}C	∿5,750 yr	β	6.94×10^{-4}	
^{22}Na	2.6 yr	γ	Not calculated	3.6×10^{-2}
^{33}P	∿25 days	β	3.24×10^{-1}	
^{35}S	∿88 days	β	3.17×10^{-2}	
^{36}Cl	∿3 x 10^{5} yr	γ	8.40×10^{-6}	
^{37}Ar	35 days	γ	2.15×10^{-1}	
^{39}Ar	∿265 yr	β	3.96×10^{-2}	
^{40}K	∿1.26 x 10^{9} yr	γ	3.76×10^{-5}	
^{41}Ca	8 x 10^{4} years	γ	7.00×10^{-3}	
^{45}Ca	165 days	β	3.66×10^{0}	
^{46}Sc	84 days	γ	1.86×10^{-4}	
^{51}Cr	28 days	γ	1.04×10^{-1}	
^{54}Mn	∿300 days	γ	1.68×10^{-1}	
^{55}Fe	2.7 yr	γ	$3.01 \times 10^{+1}$	
^{59}Fe	45 days	γ	9.99×10^{-1}	
^{58}Co	72 days	γ	2.15×10^{-2}	
^{60}Co	5.27 yr	γ	6.69×10^{-1}	1.01×10^{0}
^{59}Ni	∿80,000 yr	γ	1.19×10^{-3}	
^{63}Ni	∿100 yr	β	1.40×10^{-1}	
^{65}Zn	245 days	γ	4.47×10^{-7}	
^{93m}Nb	13.6 yr	γ	2.77×10^{-5}	
^{95}Nb	35 days	γ	5.40×10^{-6}	
^{93}Mo	∿3,500 yr	γ	6.69×10^{-5}	
^{99}Tc	2.12 x 10^{5} yr	β	4.93×10^{-5}	
^{152}Eu	12.7 yr	γ	Not calculated	8.7×10^{-1}

Those radionuclides in Table 7.3-5 whose half-lives and/or shutdown concentrations result in a significant contribution to the total activity after one year and/or after 100 years of decay are listed in Table 7.3-6. The fractional contribution of each of these radionuclides to the total radioactivity of the bioshield concrete at reactor shutdown is shown, and the reduction of those fractional levels for decay periods of 10, 30, 50 and 100 years is also shown.

TABLE 7.3-6. Calculated Radioactivity Levels in Activated Biological Shield Concrete[a] (Fractional Activity Normalized at Reactor Shutdown)

Radionuclide	Radioactivity[c] at Shutdown (Ci/m^2)	Fractional Radioactivity at Decay Times of: Shutdown	10 Years	30 Years	50 Years	100 Years
^{39}Ar	3.96×10^{-2}	1.14×10^{-3}	1.11×10^{-3}	1.05×10^{-3}	1.00×10^{-3}	8.78×10^{-4}
^{41}Ca	7.00×10^{-3}	2.01×10^{-4}	2.01×10^{-4}	2.01×10^{-4}	2.01×10^{-4}	2.01×10^{-4}
^{45}Ca	3.66×10^{0}	1.05×10^{-1}	2.30×10^{-8}	--(b)	--	--
^{54}Mn	1.68×10^{-1}	4.83×10^{-3}	1.05×10^{-6}	--	--	--
^{55}Fe	$3.01 \times 10^{+1}$	8.65×10^{-1}	6.64×10^{-2}	3.91×10^{-4}	2.30×10^{-6}	--
^{60}Co	6.69×10^{-1}	1.92×10^{-2}	5.15×10^{-3}	3.71×10^{-4}	2.67×10^{-5}	3.73×10^{-8}
^{59}Ni	1.19×10^{-3}	3.42×10^{-5}	3.42×10^{-5}	3.42×10^{-5}	3.42×10^{-5}	3.42×10^{-5}
^{63}Ni	1.40×10^{-1}	4.02×10^{-3}	3.75×10^{-3}	3.27×10^{-3}	2.84×10^{-3}	2.01×10^{-3}
	$3.478 \times 10^{+1}$	1.0	7.70×10^{-2}	5.32×10^{-3}	4.10×10^{-3}	3.12×10^{-3}

(a) The radionuclides listed include only those whose half-life and/or initial concentration result in a significant contribution after one year's decay and/or one hundred years decay.
(b) Dashes mean the fractional level of radioactivity was less than 10^{-10}.
(c) Data from Table 7.3-5.

The radionuclide inventories shown in Table 7.3-3, 7.3-4, and 7.3-6 are used in estimating public radiation exposure from airborne releases during the decommissioning operations.

7.3.1.2 Neutron-Activated Corrosion Products

Numerous radioactive materials are present in the reactor coolant streams during reactor operation. Some of these materials, principally the neutron-activated corrosion products, tend to deposit on the inner surfaces of the piping systems creating distributed sources of radiation throughout the facility.

The composition of these activated corrosion product sources is derived from information available in the literature.[9,10] The fractional activities of the various corrosion products deposited on the primary side of steam generator surfaces, measured during steam generator repairs, are shown in Table 7.3-7. It can be seen that the cobalt isotopes comprise 78% of the total activity at reactor shutdown. The effect of radioactive decay on the mixture, for decay times up to 100 years is also illustrated in Table 7.3-7.

TABLE 7.3-7. Estimates of the Deposition of Neutron-Activated Corrosion Products on Piping Internal Surfaces(a) (Fractional Activity Normalized at Reactor Shutdown)

Radionuclide	Deposited Radioactivity ($\mu Ci/m^2$)	Fractional Radioactivity at Decay Times of: Shutdown	10 Years	30 Years	50 Years	100 Years
^{51}Cr	5.3×10^3	2.4×10^{-2}	--(b)	--	--	--
^{54}Mn	8.0×10^3	3.6×10^{-2}	1.1×10^{-5}	--	--	--
^{59}Fe	1.8×10^3	8.2×10^{-3}	--	--	--	--
^{58}Co	1.0×10^5	4.6×10^{-1}	--	--	--	--
^{60}Co	7.1×10^4	3.2×10^{-1}	8.6×10^{-2}	6.2×10^{-3}	4.4×10^{-4}	6.0×10^{-7}
^{95}Zr	8.8×10^3	5.6×10^{-2}	--	--	--	--
^{95}Nb	1.2×10^4	5.6×10^{-2}	--	--	--	--
^{103}Ru	5.9×10^3	2.6×10^{-2}	--	--	--	--
^{137}Cs	2.6×10^2	1.2×10^{-3}	9.5×10^{-4}	6.0×10^{-4}	3.8×10^{-4}	1.2×10^{-4}
^{141}Ce	1.5×10^4	6.6×10^{-2}	--	--	--	--
TOTAL	2.3×10^5	1.0	8.7×10^{-2}	6.3×10^{-3}	8.2×10^{-4}	1.2×10^{-4}

(a) Based on surface corrosion product activities on a steam generator in Reference 9.
(b) A dash indicates values less than 1×10^{-10}.

Note: The activities are based on actual data from the Turkey Point Reactors extrapolated to 7 years of commercial operation.

(9) U.S. NRC, Steam Generator Repair Report Revision 1. Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251, December 1977.

(10) Design, Inspection, Operation and Maintenance Aspects of the W NSSS to Maintain Occupational Radiation Exposures as Low as Reasonably Achievable. WCAP-8872, Westinghouse Electric Corporation, Nuclear Energy Systems, Pittsburg, PA, April 1977.

The details for calculating the quantities of corrosion products deposited on the internal surfaces of the reactor systems are contained in Appendix C. The quantities were estimated for the steam generators, the pressurizer, and for piping ranging in size from 51 mm diameter to 737 mm diameter. The results of these calculations, shown in Table 7.3-8, indicate that with 90% decontamination less than 5,000 curies of deposited radioactive corrosion products would be removed from the interior of the reactor piping systems by a vigorous chemical decontamination program immediately following reactor shutdown.

TABLE 7.3-8. Estimates of Quantities of Radioactive Corrosion Products Deposited on the Interior of Reactor Systems

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

The information in Tables 7.3-7 and 7.3-8 is used in Appendix J in estimating public radiation exposure from airborne releases during decommissioning operations, which is summarized in Section 11.

7.3.1.3 Fission Products and Transuranics from Failed Fuel

Fission products and transuranics enter the reactor coolant system as a result of leaching from fuel rods which contain cladding defects. A study[11] of leach rates for selected fission products and transuranics from

[11] Y. B. Katayama, Leaching of Irradiated LWR Fuel Pellets in Deionized and Typical Ground Water, BNWL-2057, Battelle Pacific Northwest Laboratories, July 1976.

irradiated light water reactor fuel immersed in room temperature water shows that the leach rates of plutonium and californium are factors of about 25 and 300 smaller respectively than the leach rate for cesium. It is therefore assumed for this study that transuranics are in sufficiently small concentration to permit the waste material to be treated as non-transuranic waste.

The fission products found in the reactor coolant from failed fuel tend to stay in solution rather than plating out as the corrosion products do, and are removed from solution by the RCS cleanup ion exchangers. However, fluid leaks from the RCS could result in the deposition of these materials on surfaces in the vicinity of the leaks, resulting in sources of radioactive contamination. Leaks occurring in areas normally accessable to operating personnel would probably be fixed and cleaned up according to operating procedures. Leaks in areas not normally accessable to operating personnel, such as in the ion exchange vaults, would not be cleaned up until actual decommissioning begins. Buildup of radioactive contamination would occur in this case during the operating life of the plant. No data related to the composition and levels of radioactivity on the surfaces of non-accessable areas was found in the literature. Since these conditions will probably exist at the end of the operating life of the reactor, for this study it is assumed that an undetected leak of 1 liter per day of reactor coolant occurs and is allowed to buildup over the forty year plant lifetime. The fractional inventory of radionuclides in the reactor coolant(12) is listed in Table 7.3-9. The resulting radioactivity from such a leak accumulated on surfaces near the leak for forty years is listed in Table 7.3-10. Assuming that all of this leak occurs in one ion exchange vault with a floor area of 6 m^2, the estimated accumulated surface contamination level is 3.75×10^4 $\mu Ci/m^2$ at reactor shutdown. With this surface contamination level and radionuclide mixture, the estimated dose rate at 1 meter above the floor is $\sim$90 mrem/hr. The fractional contributers to the radionuclide inventory at shutdown, and the effect of radionuclide decay on the fractional inventory for time periods of up to 100 years after plant shutdown are also shown in Table 7.3-10. The fractional contributers for each decay period, used in calculating the acceptable surface contamiantion levels for public use of the reference PWR facility in Section 8, are listed in Table 7.3-11.

(12) U.S. NRC, Calculation of Release of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactor (PWR-GALE Code), NUREG-0017, Table 2-2, p. 2-3, April 1976.

TABLE 7.3-9. Reactor Coolant Radionuclide Concentrations(12) in an Operating PWR

Radionuclide	Half-life (days)	Quantity (μCi/gm)	Fractional Activity
^{3}H	4.5×10^{3}	1.0[a]	
^{51}Cr	2.8×10^{1}	1.9×10^{-3}	1.1×10^{-2}
^{54}Mn	3.0×10^{2}	3.1×10^{-4}	2.0×10^{-3}
^{55}Fe	9.5×10^{2}	1.6×10^{-3}	1.0×10^{-2}
^{59}Fe	4.5×10^{1}	1.0×10^{-3}	8.5×10^{-3}
^{58}Co	7.2×10^{1}	1.6×10^{-2}	4.5×10^{-2}
^{60}Co	1.9×10^{3}	2.0×10^{-3}	1.7×10^{-2}
^{86}Rb	1.9×10^{1}	8.5×10^{-5}	2.4×10^{-4}
^{89}Sr	5.3×10^{1}	3.5×10^{-4}	9.9×10^{-4}
^{90}Sr	1.0×10^{4}	1.0×10^{-5}	2.8×10^{-5}
^{90}Y	2.7	1.0×10^{-5}	2.8×10^{-5}
^{91}Y	5.9×10^{1}	6.4×10^{-5}	1.8×10^{-4}
^{95}Zr	6.5×10^{1}	6.0×10^{-5}	1.7×10^{-3}
^{95}Nb	3.5×10^{1}	6.0×10^{-5}	1.7×10^{-3}
^{103}Ru	4.0×10^{1}	4.5×10^{-5}	1.3×10^{-4}
^{106}Ru	3.7×10^{2}	1.0×10^{-5}	2.8×10^{-5}
^{125m}Te	5.8×10^{1}	2.9×10^{-5}	8.2×10^{-5}
^{129m}Te	3.4×10^{1}	1.4×10^{-3}	4.0×10^{-3}
^{131}I	8.0	2.7×10^{-1}	7.6×10^{-1}
^{134}Cs	7.5×10^{2}	2.5×10^{-2}	7.1×10^{-2}
^{136}Cs	1.4×10^{1}	1.3×10^{-2}	3.7×10^{-2}
^{137}Cs	1.1×10^{4}	1.8×10^{-2}	5.1×10^{-2}
^{140}Ba	1.3×10^{1}	2.2×10^{-4}	1.8×10^{-3}
^{140}La	1.7	2.2×10^{-4}	1.8×10^{-3}
^{141}Ce	3.2×10^{1}	7.0×10^{-5}	2.0×10^{-4}
^{144}Ce	2.8×10^{2}	3.3×10^{-5}	9.3×10^{-5}
^{143}Pr	1.4×10^{1}	5.0×10^{-5}	1.4×10^{-4}
TOTAL		3.54×10^{-1}	1.0

(a) Not included in total.

(12) U.S. NRC, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code). NUREG-0017, Table 2-2, pp 2-3, April 1976.

TABLE 7.3-10. Radioactive Surface Contamination in the Reference PWR Resulting from Accumulated Coolant Leakage in an Ion Exchanger Vault (Fractional Activity Normalized at Reactor Shutdown)

Radionuclide[a]	Accumulated Radioactivity at Shutdown (μCi)	Fractional Radioactivity at Decay Times of: Shutdown	10 Years	30 Years	50 Years	100 Years
^{51}Cr	1.5×10^{2}	6.9×10^{-4}	--[b]	--	--	--
^{54}Mn	3.2×10^{2}	1.4×10^{-3}	4.2×10^{-7}	--	--	--
^{55}Fe	5.1×10^{3}	2.2×10^{-2}	1.7×10^{-3}	9.9×10^{-6}	5.7×10^{-8}	--
^{59}Fe	1.9×10^{2}	8.7×10^{-4}	--	--	--	--
^{58}Co	1.6×10^{3}	7.5×10^{-3}	--	--	--	--
^{60}Co	1.7×10^{4}	7.5×10^{-2}	2.0×10^{-2}	1.4×10^{-3}	1.0×10^{-4}	1.4×10^{-7}
^{89}Sr	2.6×10^{0}	1.2×10^{-3}	--	--	--	--
^{90}Sr	9.3×10^{1}	6.9×10^{-4}	5.4×10^{-4}	3.4×10^{-4}	2.1×10^{-4}	6.3×10^{-5}
^{90}Y	9.3×10^{1}	6.9×10^{-4}	5.4×10^{-4}	3.4×10^{-4}	2.1×10^{-4}	6.3×10^{-5}
^{95}Zr	5.7×10^{1}	2.5×10^{-4}	--	--	--	--
^{95}Nb	5.7×10^{1}	2.5×10^{-4}	--	--	--	--
^{129m}Te	6.9×10^{1}	3.1×10^{-4}	--	--	--	--
^{131}I	3.1×10^{3}	1.4×10^{-2}	--	--	--	--
^{134}Cs	2.7×10^{4}	1.2×10^{-1}	4.1×10^{-3}	4.8×10^{-6}	5.4×10^{-9}	--
^{136}Cs	2.5×10^{2}	1.1×10^{-3}	--	--	--	--
^{137}Cs	1.7×10^{-1}	7.5×10^{-1}	5.9×10^{-1}	3.7×10^{-1}	2.4×10^{-1}	7.4×10^{-2}
TOTAL	2.25×10^{5}	1.0	6.2×10^{-1}	3.7×10^{-1}	2.4×10^{-1}	7.4×10^{-2}

(a) Radionuclides in reactor coolant that contribute more than 2.5 μCi to the total.
(b) A dash indicates values less than 1×10^{-10}.

TABLE 7.3-11. Isotopic Composition of Accumulated Radioactive Surface Contamination in the Reference PWR (Renormalized for Each Decay Time)

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

[a] A dash indicated values less than 10^{-4}.

7.3.2 Accumulated Radionuclides Deposited on the Reference Site from Normal PWR Operations

This section contains a discussion of the potential site radioactive contamination from normal PWR operation, and gives the expected contamination levels at and after plant shutdown. Accidental releases of radionuclides are not expected to significantly increase the radioactivity levels found on the site from normal operatons. Thus, accidental releases are not considered in this analysis.

Radioactive contamination is expected to be on the reference PWR site as a result of forty years of plant operation. Information about the levels and nature of the radioactive contamination present at the time of decommissioning is needed for a determination of the alternative uses of the site. Naturally occuring radionuclides and those resulting from nuclear weapons testing will be present on the site consistent with the level generally found in the region. The magnitude of natural background based on the UNSCEAR report(13) varies with elevation from about 80 mrem per year at sea level to about 170 mrem per year in Colorado. The annual dose from fallout attributed to weapons testing is very small by comparison.(13)

Annual atmospheric releases from operating PWR's are expected to vary widely and are dependent on such factors as the plant operating conditions, the plant gaseous effluent clean-up systems, and the plant size. A comparison of annual atmospheric release information from the GESMO(14) study and normalized information from operating PWR's for the year 1975(15) is presented in Table 7.3-12 to illustrate this variability in atmospheric release information.

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

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

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

TABLE 7.3-12. Comparison of Estimated Annual Airborne Radioactivity Releases from Operating PWRs

Radionuclide	Half-Life (days)	NUREG 0002[14] Release[a] (Ci/year)	NUREG 0218[15] Release[b] (Ci/year)
^{3}H[c]	4.5×10^{3}	1.1×10^{3}	1.1×10^{3}[17]
^{14}C[c]	2.1×10^{6}	8.0	8.0[16]
^{51}Cr	2.8×10^{1}	–[d]	3.3×10^{-3}
^{54}Mn	3.0×10^{2}	1.2×10^{2}	7.3×10^{-4}
^{57}Co	2.7×10^{2}	–	3.9×10^{-5}
^{58}Co	7.2×10^{1}	1.8×10^{-2}	6.7×10^{-3}
^{60}Co	1.9×10^{3}	2.4×10^{-2}	3.6×10^{-3}
^{59}Fe	4.5×10^{1}	1.2×10^{-3}	9.6×10^{-4}
^{89}Sr	5.3×10^{1}	1.2×10^{-3}	2.5×10^{-4}
^{90}Sr	1.0×10^{4}	1.2×10^{-3}	2.2×10^{-4}
^{90}Y	2.7	1.2×10^{-3}	2.2×10^{-4}
^{95}Zr	6.5×10^{1}	–	1.2×10^{-3}
^{95}Nb	3.5×10^{1}	–	1.2×10^{-3}
^{103}Ru	4.0×10^{1}	–	1.8×10^{-4}
^{110m}Au	2.6×10^{2}	–	6.4×10^{-5}
^{124}Sb	6.0×10^{1}	–	1.6×10^{-4}
^{125}Sb	9.9×10^{2}	–	1.4×10^{-5}
^{131}I	8.0	3.0×10^{-2}	1.8×10^{-1}
^{133}I	8.4×10^{-1}	2.8×10^{-2}	2.3×10^{-2}
^{134}Cs	7.5×10^{2}	5.9×10^{-3}	5.2×10^{-4}
^{136}Cs	1.4×10^{1}	–	1.7×10^{-4}
^{137}Cs	1.1×10^{4}	1.2×10^{-2}	1.1×10^{-3}
^{140}Ba	1.3×10^{1}	–	1.5×10^{-3}
^{140}La	1.7	–	1.5×10^{-3}
^{141}Ce	3.2×10^{1}	–	7.5×10^{-6}
^{144}Ce	2.8×10^{2}	–	4.2×10^{-5}

(a) Normalized to the Reference PWR power rating of 1175 MWe and plant capacity factor of 0.75.

(b) 1975 reported release information from 29 operating PWR's, normalized to a 1175 MWe power rating and plant capacity factor of 0.75.

(c) This radionuclide is assumed to be released as a gas, thus does not accumulate on the site via ground deposition.

(d) A dash indicates no value listed in the reference.

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

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

(17) H. Harty, Nuclear Energy Center Site Survey Reactor Plant Considerations, BNWL-B-457, Battelle, Pacific Northwest Laboratories, Richland, WA, 1976.

Ground deposition from the continuous ground level release for both annual atmospheric releases was estimated using the generic site meteorological information and standard atmospheric calculational methods.[16] Radioactive decay for the released radionuclides was accounted for over the duration of the release and the following ground deposition period. The ^{3}H and ^{14}C are assumed to be released as gases, and neither radionuclide is assumed to be accumulated via ground deposition on the site. The concentrations of radionuclides deposited on the site over the forty year plant life were calculated using the XOQDOQ computer program.[18] No credit was taken in the calculation for plume rise from either buoyancy or momentum. The average deposition values for an area of 1000 meters radius around the point of release were calculated for both of the annual atmospheric release inventories given in Table 7.3-12. Estimated ground concentrations at shutdown and for decay periods of 10, 30, 50, and 100 years for the GESMO[14] release and the 1975 reported release[15] are shown in Tables 7.3-13 and 5.3-14 respectively.

The procedure used to estimate relative deposition rates was based on numerical solutions to the flux-gradient ("K-Theory") diffusion equation.[19] The effluent was not allowed to diffuse beyond a height of 200 meters in a stable condition and 1000 meters in neutral and unstable conditions.

At the ground surface, a partial sink boundary condition involving the deposition velocity was assumed. The wind and eddy diffusivity profiles required as input to the diffusion equation were those presented by Markee.[19]

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

(18) J. F. Sagendorf and J. T. Goll, XOQDOQ - Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations, Draft NRC Report, December 1976.

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

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

(19) E. H. Markee, Jr., A Parametric Study of Gaseous Plume Depletion By Ground Surface Adsorption. USAEC Meteorological Information Meeting, AECL-2787, pp. 602-613, 1967.

TABLE 7.3-13. Estimated Accumulated Activity of Radionuclides Deposited on the PWR Site over the 40-Year Plant Lifetime from GESMO[13] Study Annual Releases[a]

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

[a] Normalized to the Reference PWR power rating of 1175 MWe and plant capacity factor of 0.75.

[b] A dash indicated values less than 1×10^{-15} Ci/m^2.

Deposition velocity was allowed to vary with wind speed in accordance with an empirical equation.[20] An areal grass density of approximately 70 g/m^2 was assumed.

No allowance has been made for normal weathering conditions such as runoff, soil and vegetational coverage, or for resuspension of deposited material.

The ground concentration values in Tables 7.3-13 and 7.3-14 are used in Section 8 of this report in determining the decontamination levels required for unrestricted use of the site.

7.4 ESTIMATES OF RADIATION DOSE RATES IN THE REFERENCE PWR

The calculated radiation dose rates expected from the neutron activated reactor components at the time of final reactor shutdown and the estimated radiation dose rates throughout the balance of the reference PWR plant during

[20] C. A. Pelletier and J. D. Zimbrick, "Kinetics of Environmental Radioiodine Transport Through the Milk-Food Chain," in Environmental Surveillance in the Vicinity of Nuclear Facilities, W. C. Reining, Editor, Thomas Publishers, Springfield, IL, 1970.

TABLE 7.3-14. Estimated Accumulated Activity of Radionuclides Deposited on the PWR Site over the 40-Year Plant Lifetime from 1975 Reported Release(15) Information(a)

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

(a) Normalized to the Reference PWR power rating of 1175 MWe and plant capacity factor of 0.75.

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

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

decommissioning operations are described in the following sections. The characterizations of the principal activation products are presented in Appendix C. Details of the postulated radiation levels used in making estimates of occupational exposure during decommissioning are also presented in Appendix C.

The buildup of radioactivity in the reference PWR stops at final shutdown. The highest dose rates that are encountered after shutdown and defueling are associated with the activated reactor core internals and reactor pressure vessel.

Radiation levels in work areas and in contact with equipment being disassembled and the duration of exposure to those levels determine the external radiation exposure to decommissioning personnel. These radiation levels are significant factors in decommissioning work because they directly affect occupational exposure and have a strong influence on work plans and methods.

The emphasis during the PWR's operational lifetime was to provide electrical power in the most cost-effective manner. Downtime was expensive and it is likely that while housekeeping was adequate for occupational safety and ALARA considerations, surface contamination cleanup was probably a second order priority, especially in areas that had high radiation levels from depositions in piping and other equipment. The amount of residual surface contamination will vary from location to location and from plant to plant, depending on the owner's operating philosophy. As a result, the occupational exposure from area background sources may exceed the occupational exposure related to the task at hand and must therefore be considered during decommissioning job planning. In such cases, task priorities would be subject to change.

7.4.1 Calculated Radiation Dose Rates from the Neutron-Activated Reactor Components and Materials

The computed concentrations of radionuclides in the highly activated reactor pressure vessel and its internal components, given in Table 7.3-2, were used in calculations to predict the radiation levels that might be encountered during the removal and disposal of these components. The radioactive transport of the monoenergetic gamma rays from ^{60}Co and ^{94}Nb, and the inner bremsstrahlung gamma spectra from ^{59}Ni, ^{55}Fe, and orbital electron capture were calculated using the ANISN[6] code modeled in the physical geometry of the reactor vessel and its internals. Similar calculations were performed using the PUSHLD[21] code, in slab geometry, and the results transformed to cylindrical geometry using the standard methods given by Rockwell[22] to check on possible differences due to method and built-in features of the codes such as attenuation factors and buildup factors. Both codes produce as an output the external dose rate to biological tissue at a distance of one centimeter from the surface of the activated material.

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

The dose rates from the activated PWR components at the time of final shutdown, computed using the radionuclide concentrations given in Table 7.3-2, are given in Table 7.4-1. These dose rates include only those radiations which contribute to the dose to biological tissue. The ^{63}Ni does not contribute to the external radiation dose since energies of the beta-rays from ^{63}Ni are insufficient to penetrate the surface dead skin layers of the human body.

The time dependence of the levels of radioactivity and radiation dose rates in the activated reactor components is illustrated in Figure 7.4-1 for 140 years after reactor shutdown. The specific activities of each of the major radionuclides having half-lives greater than one year are summed and the sum normalized to unity at the time of reactor shutdown for purposes of this illustration. Similarly, the surface radiation dose rates corresponding

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

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

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

to the specific activities shown are summed and the sum normalized to unity at the time of reactor shutdown. It can be seen in Figure 7.4-1 that, while the decay of ^{63}Ni controls the total specific activity after about 15 years, the radiation dose rate reduction is controlled by the decay of ^{60}Co for the first 80 years, after which time the dose rate is controlled by ^{94}Nb.

TABLE 7.4-1. Radiation Dose Rates from Activated Components, Calculated at the Time of Final Reactor Shutdown(a)

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

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

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

NOTE: IB means inner bremsstrahlung

No rare earths were included in the calculation of activation products due to lack of definitive data on the probable initial concentrations of rare earths in concrete.

It is clear from the data shown in Table 7.4-1 and Figure 7.4-1 that the radiation dose rate from the internal components of the reactor vessel will remain well above any acceptable release levels for thousands of years. For this reason, permanent entombment is not considered a satisfactory decommissioning approach, and disposal of this material in deep geologic storage may well be required.

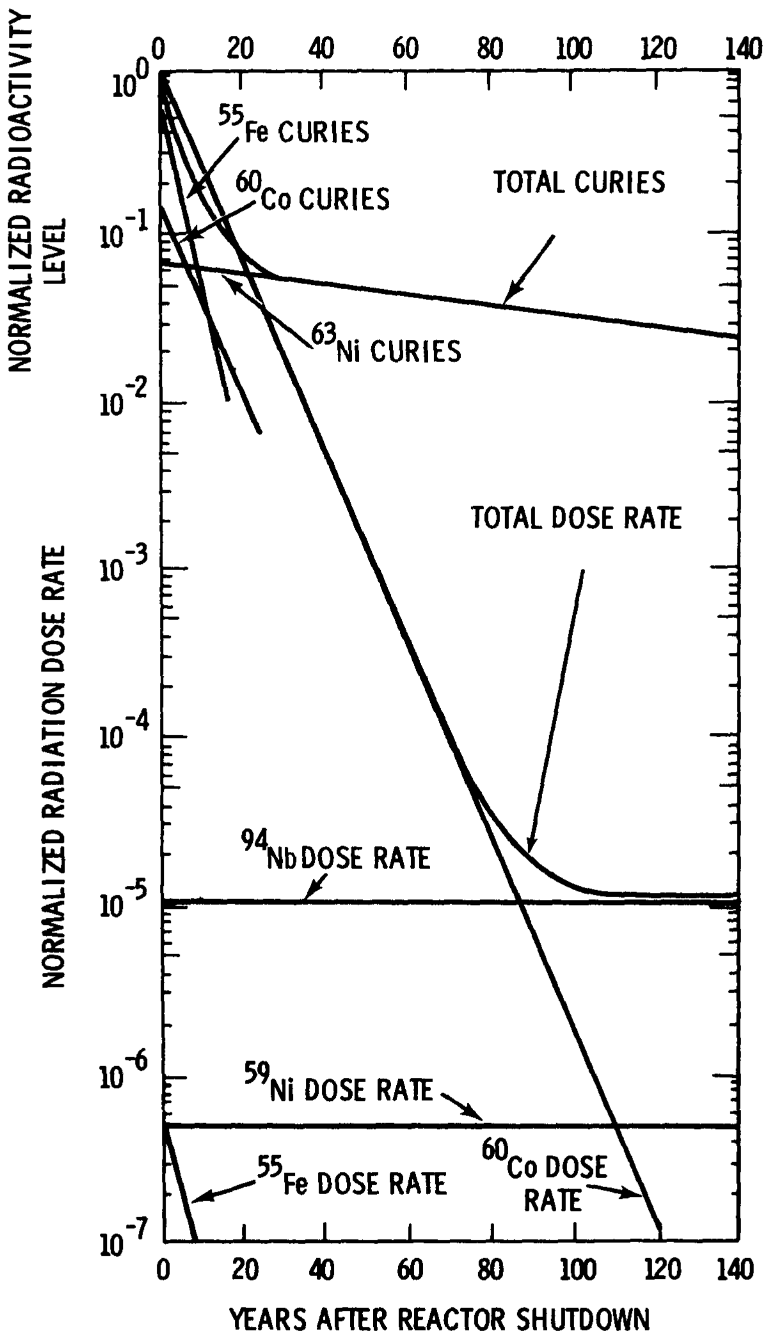

FIGURE 7.4-1. Time Dependence of Radioactivity Levels and Radiation Dose Rates in the Activated Reactor Components

7.4.2 Radiation Dose Rates Throughout the Balance of the Reference PWR Plant

The measured radiation dose rate data used as the basis for this study (see Appendix C) came from six reactor stations that had been operating from 3 to 6 years. The equilibrium levels of radiation dose rate from piping depositions described previously in Section 7.3.1.2 have probably not yet been reached. However, the data[(23)] presently available are not adequate to permit extrapolation to thirty years of full power operation, as shown in Figure 7.4-2. Therefore, composite radiation level values created from data from the six PWR's are used to estimate occupation radiation exposures to decommissioning personnel without further upward adjustment. A representative sample of these radiation level estimates is presented in Table 7.4-2. The wide range of dose rates seen in the table (from 0.001 to 30 r/hr) are postulated to be typical for the reference PWR after final shutdown and before any chemical decontamination efforts.

The RCS radiation levels are caused primarily by the activated corrosion products ^{58}Co and ^{60}Co. Measurements have shown that ^{60}Co increasingly dominates the radiation field after a few years of operation. Only ^{58}Co and ^{60}Co deposited activities need to be considered for any immediate decommissioning approach since these two radionuclides contribute more than 90% of the out-of-core radiation dose rates with 70% of the dose rate attributed solely to ^{60}Co.[(24)] The relative decay rates of the principal activated corrosion products as a function of time are presented in Figure 7.4-3. The curves on Figure 7.4-3 roughly indicate the relative contribution to the dose rate by the different corrosion product isotopes and the total activity as a function of time after reactor shutdown. The relative fraction of ^{60}Co activity shown here may be lower than might be encountered in a plant which has operated 30 to 40 years, but these values represent presently available information.

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(23) S. G. Sawochka, N. P. Jacob and W. L. Pearl, Primary System Shutdown Radiation Levels at Nuclear Power Generation Stations, EPRI 404-2, Figures 6 and 7, December 1975.

(24) A. J. Kennedy, PWR Corrosion Products: Synthesis and Significance. Presented to the American Nuclear Society, Novemer 27 - December 2, 1977 in San Francisco, CA, Babcock & Wilcox Document Number RDTPL 77-24, November 1977.

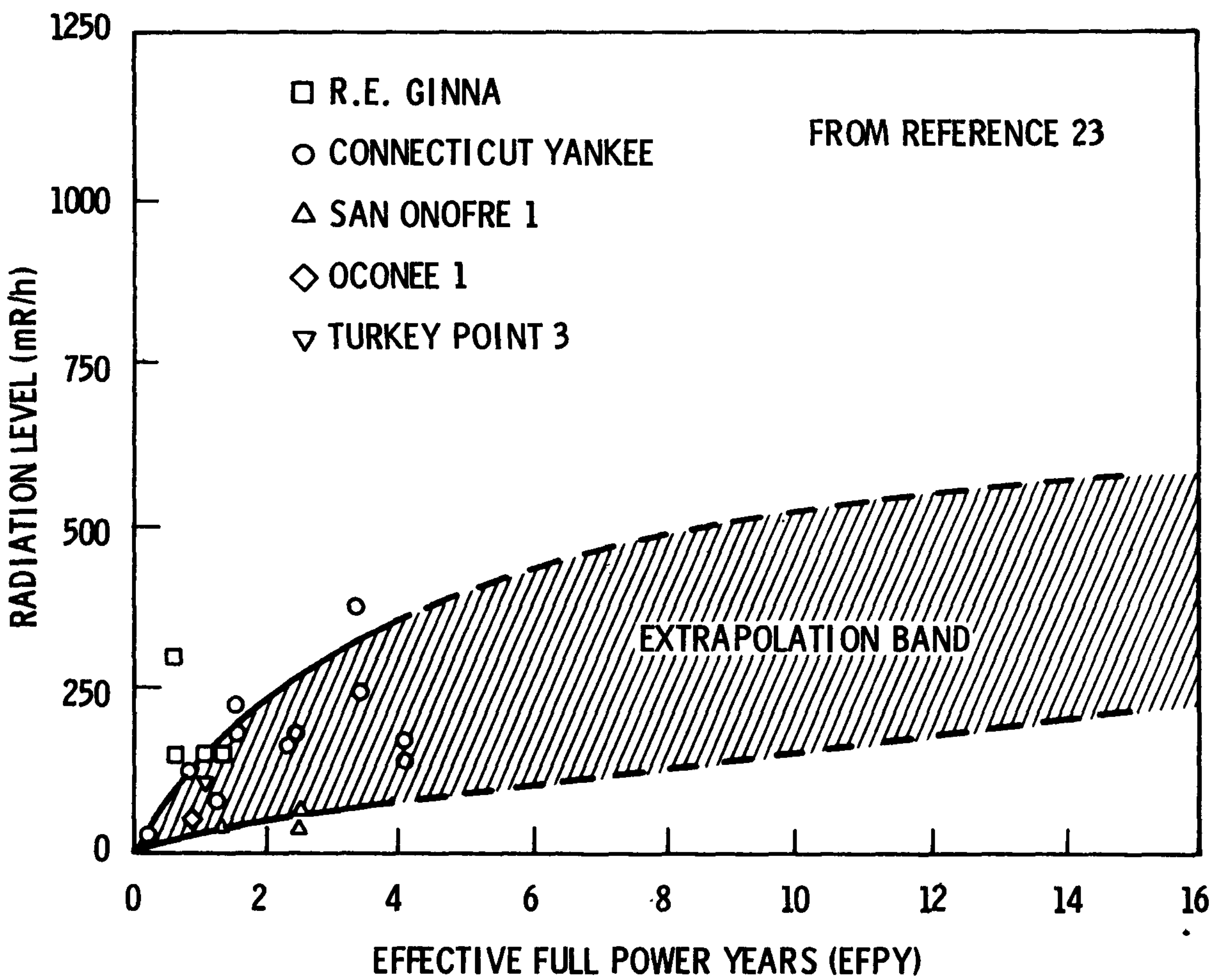

FIGURE 7.4-2. Shutdown Radiation Levels on PWR Coolant System

TABLE 7.4-2. Estimated Radioactivity Levels in the Reference PWR at Shutdown

• Areas are designated as follows:

Measurement Point Prefix	Corresponding Elevation, m (ft)
1A - xx	13.72 (45)
2A - xx	18.59 (61)
3A - xx	23.47 (77)
4A - xx	28.35 (93) - Operating Level

Measurement Point(a)	Location	Type of Measurement(b)	Dose Rate, R/hr
	• Reactor Containment Building		
1A-01	Reactor Coolant Pump Bowl	Contact	12-30[c]
1A-02	RCS Piping, cold leg	Contact	0.5-0.6
1A-03	Steam Generators	General Area	0.05-0.4
1A-04	Emergency Personnel Lock	Inside Lock Area	0.001-0.012
1A-05	Floor Drains	Contact	0.1-0.6
1A-22	Pressurizer Area	General Area	≤0.2
2A-02	Regenerative Heat Exchanger (Hx)	Contact	1-15
2A-04	Between Steam Generator (SG) Enclosure and Containment Vessel (CV) wall	General Area	≤0.025
2A-07	Between RCS Pumps and SG's	General Area	0.1-0.9
3A-01	Between Upper Internals Storage and CV Wall	General Area	0.02-0.1
3A-05	Near CV Wall	General Area	0.005-0.02
4A-01	Reactor Cavity, Inside Edge	General Area	0.1-1
4A-03	Steam Generators	General Area	≤0.2
	• Auxiliary Building		
1A-30	Component Cooling Water Pumps	General Area	≤0.15
1A-32	Waste Tank Room	General Area	0.2-0.4
1A-35	Treated Waste Monitor Tanks	Contact	0.01-0.3
2A-12	Pipeway	General Area	0.05-0.15
2A-13	Resin Storage Tank	General Area	≥0.4
2A-15	Volume Control Tank	General Area	1-3
2A-20	Radwaste Evaporator Room	General Area	0.25-0.5
3A-12	Waste Evaporator Panel	General Area	0.001-0.01
3A-25	Demineralizers	General Area	0.01 -0.2
4A-13	HEPA Exhaust Filters	Contact	≥0.005
	• Fuel Building		
1A-27	Waste Holdup Tank Rooms	General Area	2-5
1A-28	Water Heat Exchangers	General Area	0.07-0.14
1A-29	Gas Stripper Feed Pumps	General Area	≥0.025
2A-10	Drumming Room	General Area	0.2-1.5
2A-17	Drumming Room Entrance	General Area	0.2
2A-19	CVCS Monitor Tanks	Contact	≤0.3
3A-08	Boric Acid Evaporator Room	General Area	0.3-0.5
3A-10	Spent Fuel Pool Pump	General Area	≥0.05
3A-14	Spent Fuel Pool Skimmer Filters	General Area	≥0.1
4A-08	Controlled Access Machine Shop	General Area	0.02-0.1
4A-12	Spent Fuel Pit	General Area	≥0.025

(a) The specific location of the Measurement Points can be found in the facility plan view drawings in Appendix C.

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

(c) Example: 12-30 means in the range of from 12 to 30 R/hr.

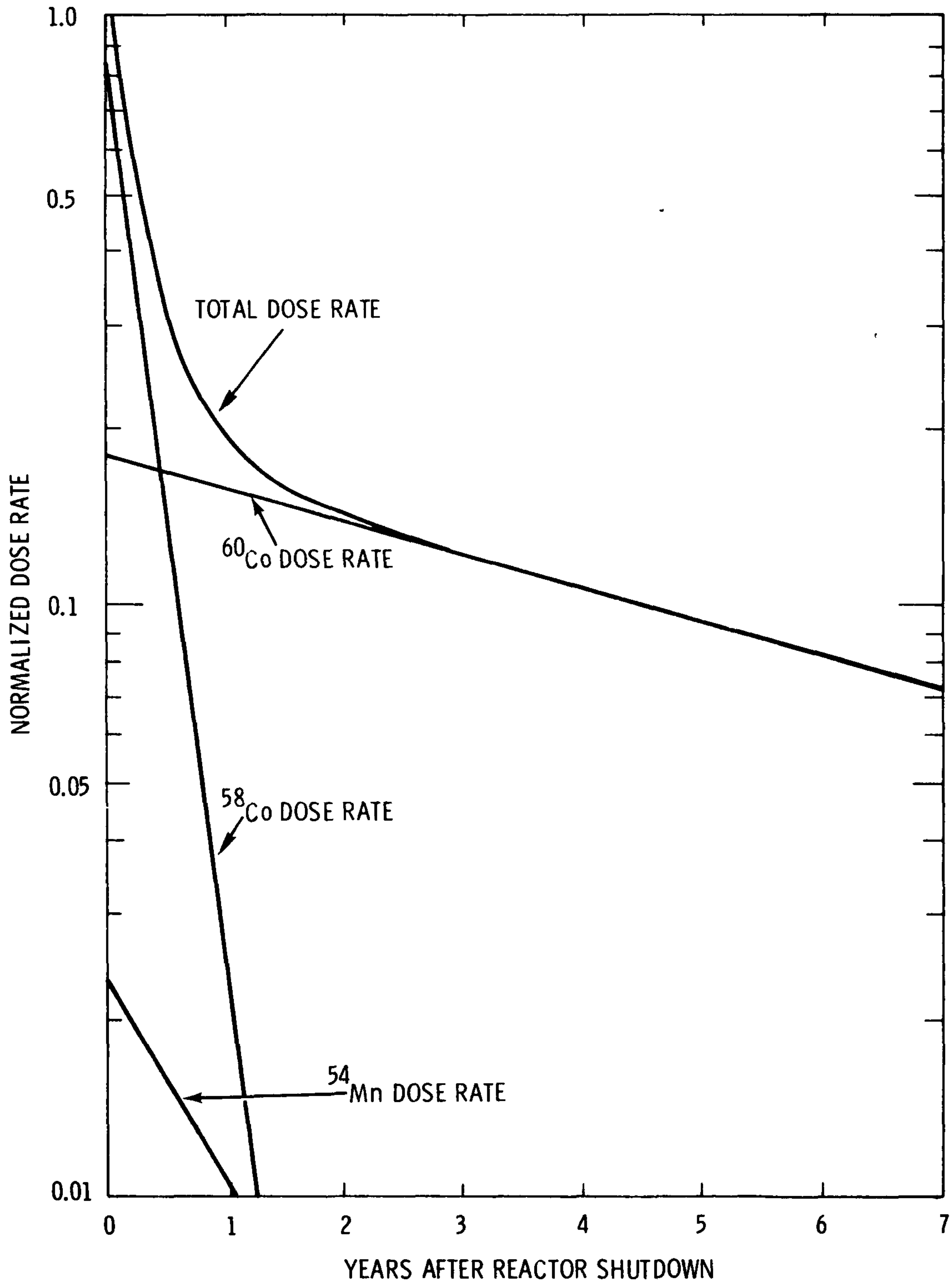

FIGURE 7.4-3. Radioactive Decay of Activated Corrosion Products

7.5 ESTIMATED INVENTORY OF CHEMICALS

Plant chemistry control limits the chemical usage to levels less than those specified in Table 7.5-1.[25]

TABLE 7.5-1. Annual Chemical Usage

Chemical	Expected Amount, kg (per year)	Limiting Amount, kg (per year)	Method of Addition
Sulfuric Acid	1.60×10^5	4.01×10^5	Continuous
Chlorine	1.11×10^5	2.78×10^5	Batch Basis
Volatile Amines	414	1035	Batch Basis
Boric Acid	331	828	Batch Basis
Sodium Hydroxide	2.81×10^4	7.03×10^4	Batch Basis
Alum	0.8×10^4	1.99×10^4	Batch Basis
Sodium Bisulfite	298	7451	Continuous

Plant chemical usage limits do not replace those limits on chemical discharges for a given plant but do provide assistance in meeting the discharge limits during plant operation by preventing excess and unnecessary usage. A nominal three month inventory of chemicals is kept on hand. Most of these chemicals are assumed to be removed prior to decommissioning. Inventories of these chemicals will therefore be limited to residuals in vessels and piping at the start of decommissioning.

[25] Environmental Technical Specifications for Trojan Nuclear Plant, Appendix B, Portland General Electric Co., November 1975.

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

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

3. TROJAN Final Safety Analysis Report, Portland General Electric Co., September 1973.

4. RESAR-3 Preliminary Safety Analysis Report, Portland General Electric Co., November 1973.

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

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

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

8. AEC-Elk River Reactor Dismantling Plan, SS-836, Rural Cooperative Power Association, August 1971.

9. U.S. NRC, Steam Generator Repair Report Revision 1. Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251, December 1977.

10. Design, Inspection, Operation and Maintenance Aspects of the W NSSS to Maintain Occupational Radiation Exposures as Low as Reasonable Achievable. WCRP-8872, Westinghouse Electric Corporation, Nuclear Energy Systems, Pittsburg, PA, April 1977.

11. Y. B. Katayama, Leaching of Irradiated LWR Fuel Pellets in Deionized and Typical Ground Water, BNWL-2057, Battelle, Pacific Northwest Laboratories, July 1976.

12. U.S. NRC, Calculation of Release of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code). NUREG-0017, Table 2-2, pp. 2-3, April 1976.

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

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

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16. Meteorology and Atomic Energy 1968. TID 24190, p. 73, edited by D. H. Slade, July 1968.

17. H. Harty, Nuclear Energy Center Site Survey Reactor Plant Considerations, BNWL-B-457, Battelle, Pacific Northwest Laboratories, Richland, WA, 1976.

18. J. F. Sagendorf and J. T. Goll, XOQDOQ - Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations, Draft NRC Report, December 1976.

19. E. H. Markee, Jr., A Parametric Study of Gaseous Plume Depletion By Ground Surface Adsorption. USAEC Meteorological Information Meeting, AECL-2787, pp. 602-613, 1967.

20. C. A. Pelletier and J. D. Zimbrick, "Kinetics of Environmental Radioiodine Transport Through the Milk-Food Chain, in Environmental Surveillance in the Vicinity of Nuclear Facilities, W. C. Reining, Editor, Thomas Publishers, Springfield, IL, 1970.

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

22. Theodore Rockwell III, Reactor Shielding Design Manual, D. Van Nostrand Co., Inc., Princeton, NJ, 1965.

23. S. G. Sawochka, N. P. Jacob and W. L. Pearl, Primary System Shutdown Radiation Levels at Nuclear Power Generation Stations, EPRI 404-2, Figures 6 and 7, December 1975.

24. A. J. Kennedy, PWR Corrosion Products: Synthesis and Significance. Presented to the American Nuclear Society, November 27-December 2, 1977 in San Francisco, CA, Babcock & Wilcox Document Number RDTPL 77-24, November 1977.

25. Environmental Technical Specifications for Trojan Nuclear Plant, Appendix B, Portland General Electric Co., November 1975.

8.0 METHODOLOGY FOR DETERMINING ACCEPTABLE CONTAMINATION LEVELS FOR PUBLIC USE OF THE DECOMMISSIONED REFERENCE PWR AND SITE

The ultimate disposition of decommissioned nuclear facilities and their surrounding sites depends upon the degree and type of radioactive contamination remaining. Examination of existing guidelines and regulations has led to the conclusion that there is a need for a general method to derive acceptable radioactive contamination levels that can be applied to the public release of any decommissioned nuclear facility or site.[1] Some guidance currently exists defining the levels of radioactive surface contamination which are acceptable to the USNRC for the termination of operating licenses.[2,3] Other suggested guidance is directed toward specific types of nuclear facilities, or accident situations involving radioactivity.[4-9]

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(1) K. J. Schneider and C. E. Jenkins, Study Coordinators, Technology, Safety and Cost of Decommissioning a Reference Nuclear Fuel Reprocessing Plant. Report of U.S. Nuclear Regulatory Commission by Battelle, Pacific Northwest Laboratory, NUREG-0278, October 1977.

(2) U.S. NRC, Termination of Operating Licenses for Nuclear Reactors. Regulatory Guide 1.86, June 1974.

(3) U.S. AEC, Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-product, Source or Special Nuclear Material. April 1970.

(4) U.S. Code of Federal Regulations. Title 49, Part 173, "Transportation." Superintendent of Documents, GPO, Washington, DC, 20555, 1976.

(5) U.S. ERDA, "Prevention Control and Abatement of Air and Water Pollution," U.S. ERDA Manual. Chapter 0510, April 2, 1975.

(6) J. W. Healy, A Proposed Interium Standard for Plutonium in Soils. LA-5483-MS, Los Alamos Scientific Laboratory, Los Alamos, NM, January 1974.

(7) C. E. Guthrie and J. P. Nichols, Theoretical Possibilities and Consequences of Major Accidents in ^{233}U and ^{239}Pu Fuel Fabrication and Radioisotope Processing Plants. ORNL-3441, Oak Ridge National Laboratory, Oak Ridge, TN 37380, April 1964.

(8) A. J. Hazle and B. L. Crist, Colorado's Plutonium-Soil Standard. Colorado Department of Health, Occupational and Radiological Health Division, Denver, CO. 1975.

(9) ANSI Standard N328, Control of Radioactive Surface Contamination on Materials, Equipment and Facilities to be Released for Uncontrolled Use. In publication for ANSI National Trail and Use.

None of these guidelines is sufficiently flexible to accomodate the various radionuclide mixtures or site specific features found at each unique nuclear facility. This fact suggests that the methodology used to calculate the acceptable levels of residual radionuclide contamination at decommissioned nuclear facilities should be based on a more general concept that is capable of accomodating these unique radionuclide mixtures and site specific features. One general concept is to compare established annual dose limits with calculated annual doses to members of the public to determine acceptable radioactive contamination levels. The contamination levels derived from a maximum annual dose concept take into account the exposure of individuals to contamination remaining at a decommissioned facility or on its site following release for public use.

The purpose of this section is to describe and demonstrate the methodology for determining the acceptable contamination levels for public use of the decommissioned reference PWR and its site based on the maximum annual dose to an individual. The following terminology is used in developing the annual dose based methodology:

- Disposition Criteria The acceptable radioactive contamination levels for public use of decommissioned nuclear facilities, based on a maximum annual dose limit.
- Organs of Reference Radiation doses are calculated for specific organs of the human body. In this study these organs of reference are the thyroid glands, lungs, total body, and bone.
- Exposure Pathways The radiation exposure pathways represent ways by which people are exposed to radiation. Exposure pathways of concern in calculating the dose to personnel in the PWR facility are inhalation of radionuclides, submersion in airborne radioactivity, and external exposure from radioactive surface contamination. In the environment, the same pathways are considered, plus the ingestion of food products containing radionuclides.
- Decay Periods The continually changing mixture of the radionuclide inventories results in annual doses that are time dependent due to radioactive decay. This dependence is demonstrated by calculating the doses at the time of the PWR final shutdown, and at 10, 30, 50, and 100 years after shutdown.

- <u>Annual Dose</u> The annual dose is the radiation dose calculated during any year following continuous exposure. It is the sum of the doses received during the year of interest from all pathways including the dose resulting in that same year from the intake of radionuclides during previous years. The highest value found is referred to as the maximum annual dose. For internal emitters, this methodology differs from the method of calculating the 50-year dose commitment from one year's intake often used in performing environmental dose assessments of operating facilities.

Additional terminology, the radiation dose models and parameters, and a derivation of the equations used to determine the annual doses are contained in Appendix E, Radiation Dose Methodology.

8.1 TECHNICAL APPROACH

The basic premise for the disposition criteria methodology described in this study is that no member of the public will receive an annual dose in excess of a limit yet to be established by the U.S. Federal agencies. Discussion of future use categories and the methodology, based on maximum annual dose, for determining the acceptable contamination levels are contained in the following subsections.

8.1.1 Definition of Use Categories

During the planning stages of decommissioning, a variety of future uses may be considered for the facility and its site. For this study, three general use categories after decommissioning are considered.

They are:

- <u>Restricted Use</u> Restricted use permits only nuclear activities to be conducted at the decommissioned PWR facility and/or site. The residual radioactive contamination levels for this category are expected to be similar to the levels found at licensed operating nuclear facilities. Therefore, the exposure of workers and the public is controlled by the restrictions imposed by the nuclear license.

- Conditional Use Conditional use of the decommissioned PWR would permit use of all or part of the facility and/or site under some form of control to ensure public safety. A conditional use category is an interim category that may permit higher residual radioactive contamination levels than the unrestricted use category, assuming that controls to ensure public safety can be adequately enforced. The enforcement of the controls or restrictions imposed, such as physical barriers or signs and other radiation exposure controls, may require some form of nuclear license or some form of zoning requirements. The interim period for the enforcement of the restrictions would last until the residual contamination has decayed or until additional decontamination has reduced the radiation to levels that permit unrestricted use. The intent of such conditions is to ensure that annual doses to any member of the public using these conditionally released facilities or sites would be below the maximum annual dose limit as may be established for unrestricted use by U.S. Federal agencies. While conditional use categories do not exist in the regulations, such a category could be defined for some types of nuclear facilities. The maximum annual dose limit to members of the public for this category would be the same as for the unrestricted use category.
- Unrestricted Use Unrestricted use of the decommissioned PWR facility and/or site means that the potential exposure to members of the public from residual radioactive contamination levels attributable to the facility will not exceed the maximum annual dose limit as may be established by U.S. Federal regulatory agencies. Decommissioning a site will, in general, result in the unrestricted public use of land areas that the public had been denied use of during the normal 40-year PWR operational life.

No attempt has been made to define all of the possible specific uses that may fall into each of these general categories. The ability to enforce the license restrictions required for the first two use categories for long periods of time requires ongoing surveillance. Each potential use restriction will require its own specific analysis for acceptable contamination levels. Furthermore, the restriction can best be ensured if the responsibility lies with a government agency. For this reason, the only category for which example disposition criteria are derived in this study is for unrestricted use of the decommissioned reference PWR facility and site. The acceptable residual contamination levels for the reference PWR and site are therefore based on the assumption that no member of the public will receive an annual dose in excess of the maximum annual dose limit, yet to be established, for the unrestricted use category.

8.1.2 Disposition Criteria

Determination of the disposition criteria for the reference PWR is a procedure that is necessarily linked with other decommissioning considerations. The relationship between the objectives of this study, site specific studies, and the disposition criteria methodology is shown in Figure 8.1-1. The disposition criteria for the reference PWR are calculated using previously developed methodology.[1]

There are no unique regulations or specific guidance on acceptable maximum annual dose to individuals living on or near a decommissioned site. Guidance that could be interpreted as annual dose limit recommendations specifically for the cases of interest here include:

1. Recommendations of the International Committee on Radiation Protection (ICRP), Publication 9[10]
2. Surgeon General's Guidelines (DHEW)[11]

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[1] K. J. Schneider and C. E. Jenkins, Study Coordinators, Technology, Safety and Cost of Decommissioning a Reference Nuclear Fuel Reprocessing Plant Report of U.S. Nuclear Regulatory Commission by Battelle, Pacific Northwest Laboratory, NUREG-0278, October 1977.

[10] "Recommendations of the International Commission on Radiological Protection." ICRP Publication 9. Pergamon Press, London, 1966.

[11] Surgeon General, U.S. Public Health Service, Surgeon General's Guidelines, "Use of Uranium Mill Tailings for Constructive Purposes." Hearings Before the Subcommittee on Raw Materials of the Joint Committee on Atomic Energy, October 28 and 29, 1971. (1971), pp. 52-54.

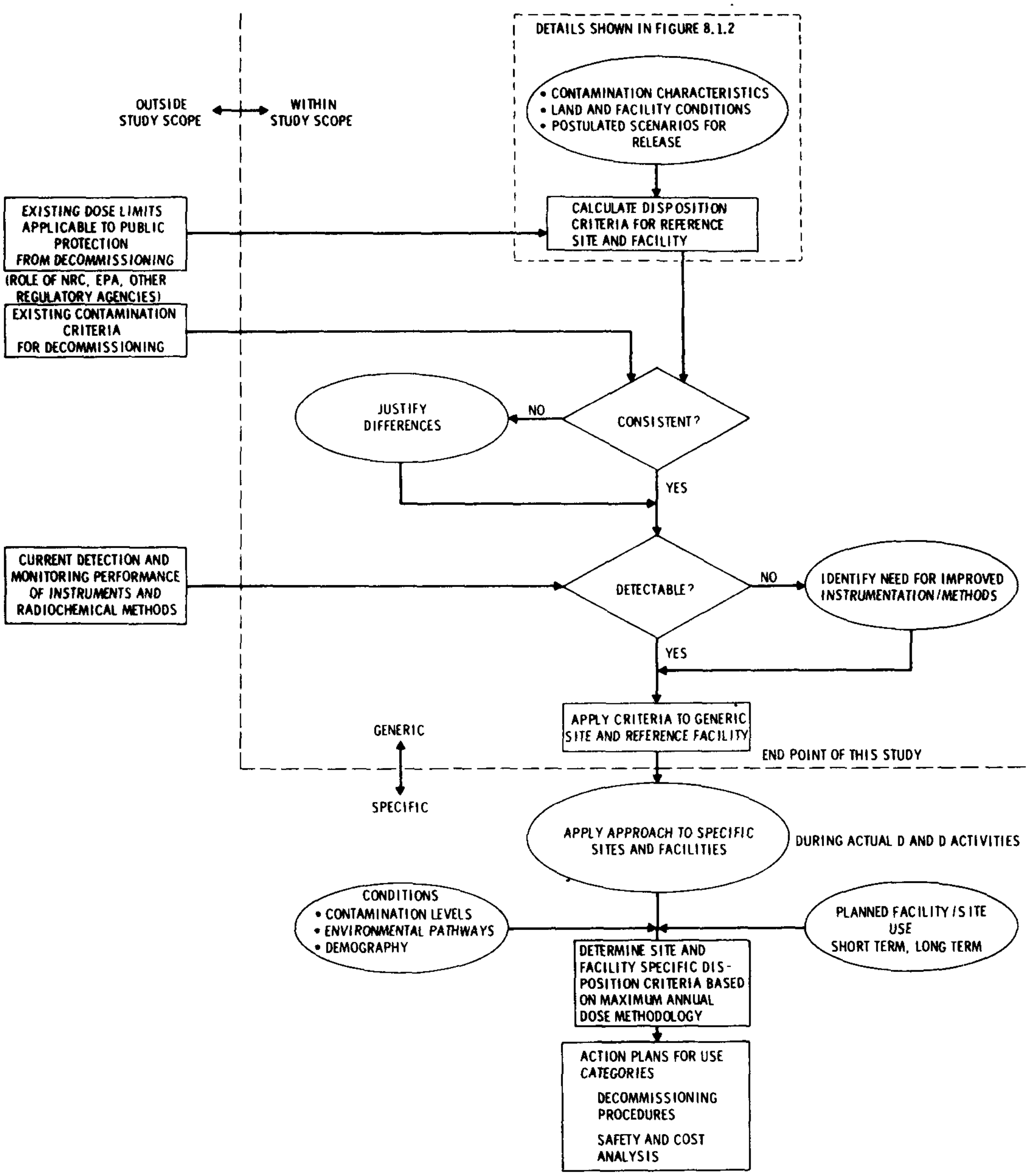

FIGURE 8.1-1. Logic Diagram for the Relationship of Disposition Criteria to Reference and Specific Decommissioning Studies

3. Appendix I of 10 CFR 50, Guides for Design Objectives for Light-Water-Cooled Nuclear Power Reactors (NRC)[12]

4. Proposed Federal Guidance for the Environmental Limits of Transuranium Elements (EPA)[13]

5. 40 CFR 190 Environmental Radiation Protection Requirements for Normal Operations of Activities in The Uranium Fuel Cycle (EPA).[14]

None of the guidance, which provides limits on the dose rate to the public from nuclear facilities, was proposed specifically for decommissioned property. Such guidance as is provided however, does suggest different annual dose limits, or an equivalent to an annual dose limit ranging from annual doses to the total body of 3 to 500 mrem per year.[1]

It is not within the scope of this study to recommend annual dose limits for the exposure of the public to radioactive materials. Instead, acceptable residual contamination levels are to be calculated for the bounds of a range of possible annual dose limits. It is reasonable to expect that an annual dose limit established uniquely for the control of public exposure from decommissioned nuclear facilities will probably fall in the range of the lowest values in References 10 through 14. This range appears to be from 1 to 25 mrem per year. The selection of this range is not intended nor should it be implied as a recommendation for limiting radiation exposure of the public from decommissioned nuclear facilities. It is also assumed that any limit so established will apply to the maximum annual dose to any organ of reference; thus

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(12) Code of Federal Regulations. Title 10, Part 50, Appendix I, "Licensing of Production and Utilization Facilities," Superintendent of Documents, GPO, Washington, DC, 20555, 1976.

(13) U.S. Environmental Protection Agency, Proposed Guidance On Dose Limits For Persons Exposed To Transuranium Elements in The General Environment. EPA 520/4-77-016, September 1977.

(14) U.S. Code of Federal Regulations. Title 40, Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations." Superintendent of Documents, GPO, Washington DC, 20555, January 1977.

(1) K. J. Schneider and C. E. Jenkins, Study Coordinators, Technology, Safety and Cost of Decommissioning a Reference Nuclear Fuel Reprocessing Plant, Report of U.S. Nuclear Regulatory Commission by Battelle, Pacific North-West Laboratory, NUREG-0278, October 1977.

assuring that applicable regulatory annual dose limits will not be exceeded. Example disposition criteria are developed for the reference PWR for this range of annual dose limits.

The methodology for the determination of disposition criteria based on annual dose is shown in Figure 8.1-2. A discussion of the three steps in this methodology, shown in Figure 8.1-2, is given below:

- Compute the Maximum Annual Doses The maximum annual doses to the organs of reference resulting from the radioactive contamination present at the radioactive decay times of interest are calculated using the Radiation Dose Methodology discussed in Appendix E. A derivation of the annual dose equations and listing of the calculated dose values are found in subsections E.1.5 and E.2.1. The reference radionuclide inventories used in these calculations are discussed in Section 7. These inventories are listed in Appendix E, to show the relationship between the calculated doses and the radionuclide inventories.

- Compute the Contamination Levels The residual contamination levels, or disposition criteria, expressed in units of microcuries per square meter ($\mu Ci/m^2$) are calculated for the organs of reference, and for the maximum annual dose values of 1 and 25 mrem per year.

- Determine the Maximum Acceptable Contamination Level The maximum acceptable contamination level at the assumed maximum annual dose is determined by selecting the largest calculated organ dose derived from all exposure pathways. This value is dependent on the composition of the radionuclide inventory.

8.2 EXAMPLE CALCULATION OF THE ACCEPTABLE CONTAMINATION LEVELS AT THE DECOMMISSIONED REFERENCE PWR

The methodology for developing disposition criteria is best demonstrated by calculating example criteria for the reference PWR facility and site.

8.2.1 Acceptable Residual Contamination Levels in The Reference PWR Facility

Example disposition criteria for the decommissioned reference PWR facility are calculated following the logic illustrated in Figure 8.1-2, and using

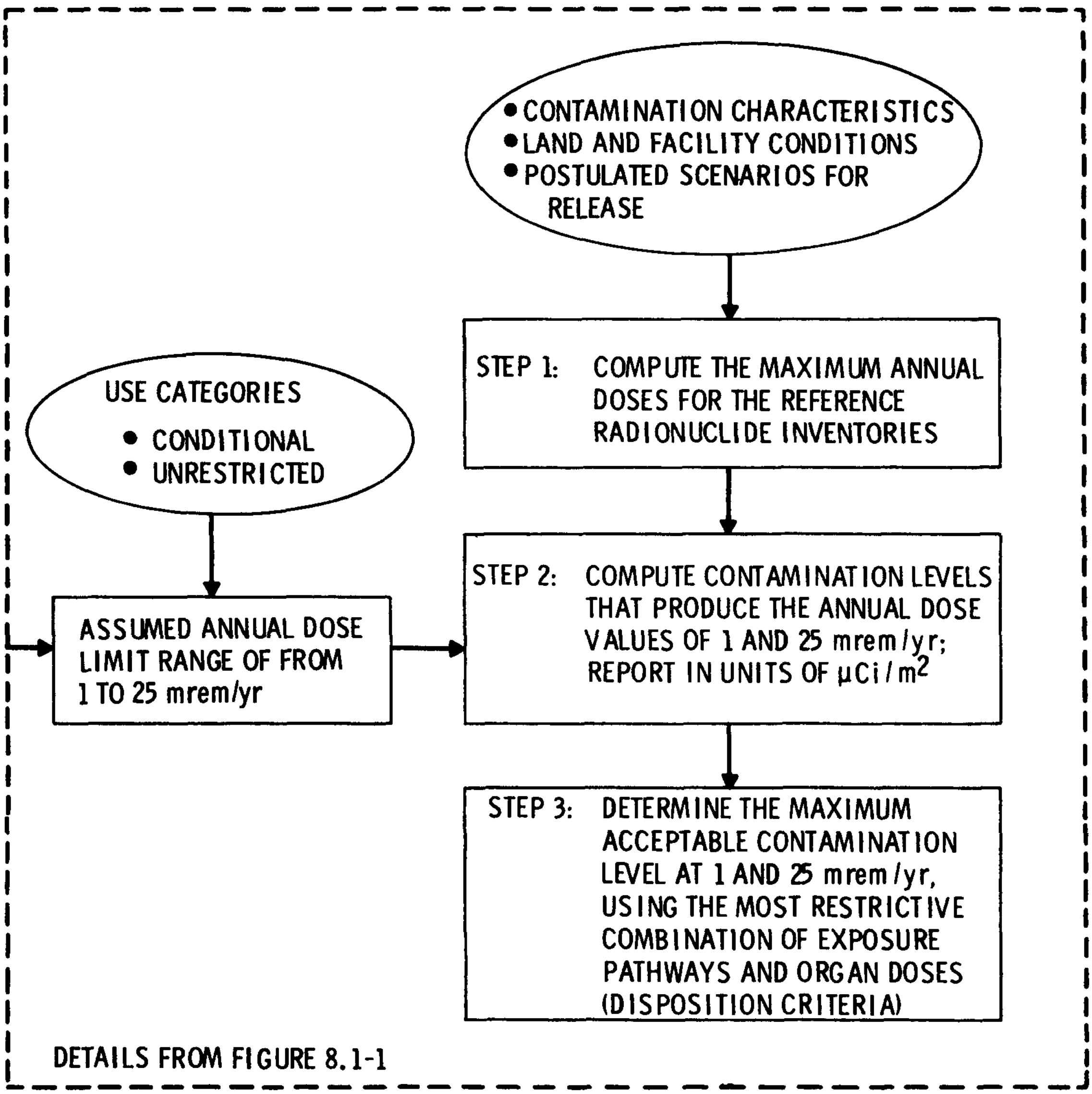

FIGURE 8.1-2. Details of the Annual Dose-Based Disposition Criteria Methodology

the terminology listed previously and in Appendix E. The surface contamination inventory of radionuclides in the reference PWR is derived in Section 7. Contamination is assumed to accumulate on a surface for the entire 40-year PWR operating life as a result of a postulated primary coolant leak. The quantity of surface contamination for the reference PWR is difficult to predict. It will be specific to each PWR and its operating history. It can be estimated for a specific PWR, however, it is best determined by measurement at the time of shutdown on a case by case basis. It is reasonable, however, to predict the isotopic composition of this contamination. Therefore, surface contamination levels have been normalized to 1 μCi per square meter at each decay time. The actual radioactivity levels and isotopic composition experienced at the facility are important in determining the degree of decontamination required; however, only the isotopic composition is necessary in determining the disposition criteria.

The residual radioactive contamination levels present during decommissioning operations are assumed to be appropriately monitored and surface radiation measurements suitably recorded. The decommissioning operations, discussed in Section 9 and Appendix F, are designed to remove surface radioactive contamination until the radiation levels acceptable for unrestricted use are achieved. These levels, or disposition criteria, for the facility are derived in this section based on radioactive surface contamination, with the assumption that all volumetric wastes generated during decommissioning in achieving these criteria are disposed of as radioactive wastes.

For the maximum annual dose calculations, airborne radionuclide concentrations in the PWR facility are calculated using a constant resuspension factor of 5×10^{-6} m^{-1} as discussed in Appendix E, subsection E.1.1. Results of actual measurements of airborne radionuclide concentrations in decommissioned facilities could alter the allowable contamination levels calculated here.

The maximum annual radiation doses to non-nuclear workers (Step 1 in Figure 8.1-2) are calculated using a 40-hour work week, and are listed in Table E.2-9 through E.2-11 in Appendix E. These tables contain doses calculated for selected organs of reference from all exposure pathways considered, and for all radionuclides that contribute more than 1% to the total dose. Acceptable contamination levels (Step 2 in Figure 8.1-2) are next calculated for doses of 1 and 25 mrem per year. These calculated acceptable residual contamination levels, expressed in units of microcuries per square meter ($\mu Ci/m^2$), for the organs of reference and radioactive decay periods of interest, are shown in Table 8.2-1.

TABLE 8.2-1. Contamination Levels Within the Reference PWR at Selected Times After PWR Shutdown (Unrestricted Use)

Time After Shutdown (Years)	Organ of Reference	Dominant Radionuclide Contributer To Dose	Deposition ($\mu Ci/m^2$) Corresponding to A Maximum Annual Dose[a] of: 1 mrem	25 mrem
0	Thyroid	^{137}Cs	0.26	6.4
	Lungs	^{137}Cs	0.23	5.7
	Body	^{137}Cs	0.25	6.2
	Bone	^{137}Cs	0.26	6.4
10	Thyroid	^{137}Cs	0.36	8.9
	Lungs	^{137}Cs	0.29	7.4
	Body	^{137}Cs	0.30	7.6
	Bone	^{137}Cs	0.29	7.4
30[b]	Thyroid	^{137}Cs	0.40	10
	Lungs	^{137}Cs	0.37	9.3
	Body	^{137}Cs	0.32	8.1
	Bone	^{137}Cs	0.32	8.1

(a) Ingestion pathways in the released decommissioned facility for non-radiation workers are non-existent.
(b) At 30 to 100 years after shutdown, ^{137}Cs is the sole component in the radionuclide inventory, so the normalized contamination levels are constant for these times.

The disposition criteria based on the largest annual dose to any organ of reference at the decay periods of interest (Step 3 in Figure 8.1-2) are listed in Table 8.2-2. At final reactor shutdown, the disposition criteria are controlled by the annual dose to the lungs, and for other decay periods shown they are controlled by the annual dose to the bone. This change in organ of reference reflects the changing composition of the radionuclide inventory with time. The major radionuclide contributer to the maximum annual dose at all times is ^{137}Cs. Additional dose to the lungs at shutdown is a result of ^{60}Co and ^{134}Cs, which rapidly decay from the radionuclide mixture. The resulting disposition criteria range between 0.23 and 8.1 $\mu Ci/m^2$, depending upon the time after shutdown considered and the annual dose limit selected.

TABLE 8.2-2. Maximum Acceptable Residual Contamination Levels Within The Reference PWR (Unrestricted Use)

Time After Shutdown (Years)	Limiting Organ Of Reference	Deposition ($\mu Ci/m^2$) Corresponding To A Maximum Annual Dose[a] of: 1 mrem	25 mrem/yr
0	Lungs	0.23	5.7
10	Bone	0.29	7.4
30[b]	Bone	0.32	8.1

[a] Ingestion pathways in the released decommissioned facility for non-radiation workers are non-existent.

[b] At 30 to 100 years after shutdown, ^{137}Cs is the dominant component in the radionuclide inventory, so the normalized contamination levels are constant for these times.

8.2.2 Acceptable Radioactivity Levels on The Reference PWR Site

Information about the levels and nature of the radionuclide contamination present on the reference PWR site was derived from two references.[15,16] The first reference lists the calculated annual atmospheric releases during the operation of a generic PWR, while the second reference lists the reported releases from 29 operating PWR power stations for the year 1975. Disposition criteria are calculated from the 40-year accumulated depositions on the site from these annual releases. Results calculated using both radionuclide inventories illustrates the dependence of the results on the mixture of the radionucides assumed.

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

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

Airborne concentrations of radionuclides in the environment are calculated using a time-dependent resuspension factor discussed in Appendix E, subsection E.1.1. The radionuclide inventories, showing the 40-year accumulated group depositions and the values at each decay period of interest, are listed in Tables 7.3-13 and 7.3-14. At plant shutdown, these radionuclides are assumed to be mixed to a depth of one centimeter in the soil with no mechanical mixing or weathering effects. After decommissioning, the site is assumed to be used for farming, and plowing is assumed to mix the radioactive contamination to a depth of 15 cm. A dry soil "surface density" factor of 224 kg per square meter mixed to a depth of 15 cm is used to determine the soil radioactivity concentration. This factor is discussed in Section E.1.2.2 in Appendix E. For calculational convience, the site disposition criteria calculated in this section are based on the surface contamination estimtes reported in Tables 7.3-13 and 7.3-14. The radioactive concentrations in activity per unit mass are summarized in Section 8.3.

It should be noted that the contamination levels defined for the site by Tables 7.3-13 and 7.3-14 are probably higher than might be encountered at a real PWR. This is primarily because no credit was taken for weathering effects on the radioactive contamination either during the 40-year PWR operating life or during the decay periods. For specific sites, comprehensive measurements will be necessary at shutdown to characterize the quantity and mixture of the deposited radioactive contamination. These inventories are not normalized to one μCi per square meter at each decay period, as was previously done in the reference PWR facility, to permit realistic environmental transport calculations.

Maximum annual doses calculated for both site radionuclide inventories (Step 1 in Figure 1.1-2) are listed in Tables E.2-12 through E.2-21. Again, these tables contain the calculated doses for the environmental exposure pathways considered, the organs of reference, and the radionuclides that contribute 1% or more to the total dose. Acceptable contamination levels (Step 2 of Figure 8.1-2) are calculated for 1 and 25 mrem per year dose values, and are listed in Tables 8.2-3 and 8.2-4. The dominant radionuclide contributer to the organ doses are listed in these tables to help illustrate the dependence of the calculated doses on the assumed radionuclide mixture. For both radionuclide inventories, the maximum annual doses at all decay periods are delivered

TABLE 8.2-3. Residual Contamination Levels On The Reference PWR Site Based On The GESMO[15] Study PWR Releases (Unrestricted Use)

Time After Shutdown (Years)	Organ of Reference	Dominant Radionuclide Contributor To Dose	Deposition ($\mu Ci/m^2$) Corresponding To A Maximum Annual Dose[a] of: 1 mrem/yr	25 mrem/yr
0	Thyroid	^{60}Co	1.1×10^{-1}	2.7×10^{0}
	Lungs	^{60}Co	1.1×10^{-1}	2.7×10^{0}
	Body	^{60}Co	8.9×10^{-2}	2.2×10^{0}
	Bone	^{90}Sr	1.4×10^{-2}	3.5×10^{-1}
10	Thyroid	^{137}Cs	1.6×10^{-1}	4.0×10^{0}
	Lungs	^{137}Cs	1.6×10^{-1}	4.0×10^{0}
	Body	^{90}Sr	4.2×10^{-2}	1.1×10^{0}
	Bone	^{90}Sr	1.1×10^{-2}	2.8×10^{-1}
30	Thyroid	^{137}Cs	2.4×10^{-1}	5.9×10^{0}
	Lungs	^{137}Cs	2.4×10^{-1}	5.9×10^{0}
	Body	^{90}Sr	3.8×10^{-2}	9.4×10^{-1}
	Bone	^{90}Sr	1.0×10^{-2}	2.6×10^{-1}
50	Thyroid	^{137}Cs	2.4×10^{-1}	6.1×10^{0}
	Lungs	^{137}Cs	2.4×10^{-1}	6.1×10^{0}
	Body	^{90}Sr	3.8×10^{-2}	9.6×10^{-1}
	Bone	^{90}Sr	1.0×10^{-2}	2.6×10^{-1}
100	Thyroid	^{137}Cs	2.4×10^{-1}	5.9×10^{0}
	Lungs	^{137}Cs	2.4×10^{-1}	5.9×10^{0}
	Body	^{90}Sr	4.3×10^{-2}	1.1×10^{0}
	Bone	^{90}Sr	1.1×10^{-2}	2.8×10^{-1}

[a] Surface contamination is assumed to be mixed to a uniform soil depth of 15 cm after decommissioning for the dose calculations, and is presented here in units of $\mu Ci/m^2$.

to the bone from ingestion of ^{90}Sr in food products. The disposition criteria in units of $\mu Ci/m^2$ (Step 3 in Figure 8.1-2) based on these bone doses, are listed in Tables 8.2-5 and 8.2-6.

8.2.3 Acceptable Contamination Levels On PWR Equipment

Disposal of much of the non-activated PWR equipment after decontamination could be covered by standards developed by the ANSI Committee N328. The complexities of decontaminating equipment for public release are great, and are briefly explored in Appendix J of this report. Decommissioning an actual PWR will probably require special procedures to dispose of equipment on a piece by piece basis.

TABLE 8.2-4. Residual Contamination Levels on the Reference PWR Site Based on Normalized NUREG-0218[16] PWR Releases (Unrestricted Use)

Time After Shutdown (Years)	Organ Of Reference	Dominant Radionuclide Contributer To Dose	Deposition: $\mu Ci/m^2$ Corresponding To A Maximum Annual Dose[a] Of: 1 mrem/yr	25 mrem/yr
0	Thyroid	^{60}Co	6.6×10^{-2}	1.7×10^{0}
	Lungs	^{60}Co	1.0×10^{-1}	2.6×10^{0}
	Body	^{90}Sr	4.1×10^{-2}	1.0×10^{0}
	Bone	^{90}Sr	1.1×10^{-2}	2.8×10^{-1}
10	Thyroid	^{60}Co	1.6×10^{-1}	3.9×10^{0}
	Lungs	^{60}Co	1.6×10^{-1}	3.9×10^{0}
	Body	^{90}Sr	2.6×10^{-2}	6.5×10^{-1}
	Bone	^{90}Sr	6.0×10^{-3}	1.5×10^{-1}
30	Thyroid	^{137}Cs	2.6×10^{-1}	6.5×10^{0}
	Lungs	^{137}Cs	2.5×10^{-1}	6.2×10^{0}
	Body	^{90}Sr	2.3×10^{-1}	5.8×10^{-1}
	Bone	^{90}Sr	6.0×10^{-3}	1.5×10^{-1}
50	Thyroid	^{137}Cs	2.8×10^{-1}	6.9×10^{0}
	Lungs	^{137}Cs	2.8×10^{-1}	6.9×10^{0}
	Body	^{90}Sr	2.3×10^{-2}	5.8×10^{-1}
	Bone	^{90}Sr	5.9×10^{-3}	1.5×10^{-1}
100	Thyroid	^{137}Cs	2.7×10^{-1}	6.8×10^{0}
	Lungs	^{137}Cs	2.7×10^{-1}	6.8×10^{0}
	Body	^{90}Sr	2.5×10^{-2}	6.2×10^{-1}
	Bone	^{90}Sr	6.6×10^{-3}	1.6×10^{-1}

[a] Surface contamination is assumed to be mixed to a uniform solid depth of 15 cm after decommissioning for the dose calculations, and is presented here in units of $\mu Ci/m^2$.

TABLE 8.2-5. Maximum Acceptable Residual Contamination Levels On The Reference PWR Site Based On The GESMO Study Annual Releases[15] (Unrestricted Use)[a]

Time After Shutdown (Years)	Deposition ($\mu Ci/m^2$) Corresponding To A Maximum Annual Dose[b] of: 1 mrem/yr	25 mrem/yr
0	1.4×10^{-2}	3.5×10^{-1}
10	1.1×10^{-2}	2.8×10^{-1}
30	1.0×10^{-2}	2.6×10^{-1}
50	1.0×10^{-2}	2.6×10^{-1}
100	1.1×10^{-2}	2.8×10^{-1}

(a) The organ of reference is bone.

(b) Includes the dose from all environmental exposure pathways.

TABLE 8.2-6. Maximum Acceptable Residual Contamination Levels On The Reference PWR Site Based On Normalized NUREG-0218 PWR Releases[16] (Unrestricted Use)[a]

Time After Shutdown (Years)	Deposition ($\mu Ci/m^2$) Corresponding To A Maximum Annual Dose[b] of: 1 mrem/yr	25 mrem/yr
0	1.1×10^{-2}	2.8×10^{-1}
10	6.8×10^{-3}	1.7×10^{-1}
30	6.0×10^{-3}	1.5×10^{-1}
50	5.9×10^{-3}	1.5×10^{-1}
100	6.6×10^{-3}	1.6×10^{-1}

(a) The organ of reference is bone.

(b) Includes the dose from all environmental exposure pathways.

8.3 EXISTING GUIDANCE ON RESIDUAL CONTAMINATION

Existing guidance on acceptable contamination levels for the unrestricted release of decommissioned nuclear facilities is found in Regulatory Guide 1.86[2] and ANSI Standard N328.[9] The levels reflected in these standards are listed in Table 8.3-1 and 8.3-2. These levels cannot be compared with the disposition criteria developed here, since the methodology used in this study is different from that used for either of these standards. Using the maximum annual dose as the general basis for determining disposition criteria allows more flexibility in the consideration of various mixtures of radionuclides realistically expected at decommissioned nuclear facilities.

The calculated disposition criteria, reported in Tables 8.2-2, 8.2-5, and 8.2-6, are summarized in Table 8.3-3. For the facility, the residual radioactivity is characterized as surface contamination. For the site, the surface contamination values are presented along with mass contamination values in units of radioactivity per unit mass. The conversion from surface to mass contamination value assumes the contamination is mixed to a depth of 1 cm before plowing, and to a depth of 15 cm after plowing.

For the PWR facility, the unrestricted use criterion is limited by the dose to the lungs at shutdown, predominantly from ^{137}Cs and ^{60}Co, and the dose to bone at 100 years from ^{137}Cs. At times between 30 and 100 years after shutdown the sole contributer to the dose is ^{137}Cs, therefore the disposition criterion calculated at 30 years is the same as for 100 years. The criterion for the next PWR facility increases with increasing time to about 20 years after shutdown and then remains constant for 80 years. After 20 years, external exposure is the dominant exposure pathway and ^{137}Cs is the sole contributor.

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(2) U.S. NRC, Termination of Operating Licenses for Nuclear Reactors. Regulatory Guide 1.86, June 1974.

(9) ANSI Standard N328, Control of Radioactive Surface Contamination on Materials, Equipment and Facilities to be Released for Uncontrolled Use. April 1978.

TABLE 8.3-1. ANSI N328 Surface Contamination Limits

Nuclide	Limit (Activity) dpm/1000 cm^2 Total	Removable
Group 1:		
Nuclides for which the nonoccupational MPC_a** is 2 x 10^{-13} Ci/m^3 or less or for which the nonoccupational MPC_w*** is 2 x 10^{-7} Ci/m^3 or less; includes Ac-227; Am-241, -242m, -243; Cf-249, -250, -251, -252; Cm-243, -244, -245, -246, -247, -248; I-125, I-129; Np-237; Pa-231; Pb-210; Pu-238, -239, -240, -242, -244; Ra-226, -228; Th-228, -230.****	Nondetectable(1)	20
Group 2:		
Those nuclides not in Group 1 for which the nonoccupational MPC_a** is 1 x 10^{-12} Ci/m^3 or for which the nonoccupational MPC_w*** is 1 x 10^{-6} Ci/m^3 or less; includes Es-254; Fm-256; I-126, -131, -133; Po-210; Ra-223; Sr-90; Th-232; U-232.****	Nondetectable$^{(\beta,\lambda)}$(2) 2,000(α)	200
Group 3:		
Those nuclides not in Group 1 or Group 2.	5,000	1,000

*See Note following Table 2 on application of limits.

**MPC_a: Maximum Permissible Concentration in Air applicable to continuous exposure of members of the public as published by or derived from an authoritative source such as NCRP, ICRP or NRC (10 CFR Part 20 Appendix B Table 2, Column 1.)

***MPC_w: Maximum Permissible Concentration in Water applicable to members of the public.

****Values presented here are obtained from 10 CFR Part 20. The most limiting of all given MPC values (e.g., soluble vs. insoluble) are to be used. In the event of the occurrence of mixtures of radionuclides, the fraction contributed by each constituent of its own limit shall be determined and the sum of the fractions must be less than 1.

(1)The instrument utilized for this measurement shall be calculated to measure at least 100 pCi of any group, contaminants, uniformly spread over 100 square centimeters.

(2)The instrument utilized for this measurement shall be calibrated to measure at least 1 nCi of any group 2 beta or gamma contaminants uniformly spread over an area equivlent to the sensitive area of the detector.
NOTE: Direct survey for unconditional release should be performed in areas where the background is $\leq$100 c/m. When the survey must be performed in a background exceeding 100 c/m, it may be necessary to use the indirect survey method to provide the additional sensitivity required.

TABLE 8.3-2. Regulatory Guide 1.86 Acceptable Surface Contamination Levels

Nuclide[a]	Average[b,c]	Maximum[b,d]	Removable[b,e]
U-nat, ^{235}U, ^{238}U and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, ^{226}Ra, ^{228}Ra, ^{230}Th, ^{228}Th, ^{231}Pa, ^{227}Ac, ^{125}I, ^{129}I	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, ^{232}Th, ^{90}Sr, ^{223}Ra, ^{224}Ra, ^{232}U, ^{126}I, ^{131}I, ^{133}I	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except ^{90}Sr and others noted above	5000 dpm $\beta\gamma$/100 cm^2	15,000 dpm $\beta\gamma$/100 cm^2	1000 dpm $\beta\gamma$/100 cm^2

[a] Where surface contamination by both alpha and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c] Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d] The maximum contamination level applies to an area of not more than 100 cm^2.

[e] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

TABLE 8.3-3. Summary of the Disposition Criteria for the Reference PWR Facility and Generic Site

	Time After Shutdown (Years)	Limiting Organ	Acceptable Residual Contamination Levels for an Annual Dose Limit of 1 mrem per Year: Surface Contamination ($\mu Ci/m^2$)	Soil Contamination: Mixed to 1 cm (pCi/g)	Soil Contamination: Mixed to 15 cm (pCi/g)
PWR Facility[a]	0	Lungs	2.3×10^{-1}	--	--
	100	Bone	3.2×10^{-1}	--	--
Site (GESMO)	0	Bone	1.4×10^{-2}	9.4×10^{-1}	6.2×10^{-2}
	100	Bone	1.1×10^{-2}	7.4×10^{-1}	4.9×10^{-2}
Site (NUREG-0218)	0	Bone	1.1×10^{-2}	7.4×10^{-1}	4.9×10^{-2}
	100	Bone	6.6×10^{-3}	4.4×10^{-1}	2.9×10^{-2}

[a] In the facility, surface contamination levels are assumed to be used to determine the necessary decommissioning procedures. All wastes generated during the decommissioning procedures are assumed to go to a nuclear waste disposal site.

On the site, the calculated disposition criteria for both radionuclide inventories is controlled by the dose to the bone from ^{90}Sr. The dominant exposure pathway is from the ingestion of radionuclides incorporated in food products raised on the site. The criteria both decrease with time, reflecting the increased fractional contribution of the more toxic ^{90}Sr in the radionuclide mixture as the short-lived radionuclides decay.

8.4 RADIATION DETECTION CAPABILITIES

Federal regulations require environmental monitoring of LWR nuclear power stations for radioactivity released during normal operations.[17] Other regulations[18] require that a licensee conduct surveys of radiation levels or concentrations of radioactive contaminants to ensure compliance with 10 CFR Part 20 limits. Specifically, paragraph 20.1(c) of 10 CFR Part 20 states that every reasonable effort should be made by the licensee to maintain radiation exposure "as low as practicably achievable." Guidance on environmental sampling techniques to help meet these regulations can be found in NRC Regulatory Guides,[19-21] and in procedures developed by the DOE Environmental Measurements Laboratory.[22]

To ensure compliance with these regulations, the staff at operating PWR power stations routinely monitor both effluent and environmental levels of

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(17) U.S. Code of Federal Regulations. Title 10, Part 50, Appendix A, "Licensing of Production and Utilization Facilities," General Design Criteria 64 Superintendent of Documents, GPO, Washington, DC 20555, 1976.

(18) U.S. Code of Federal Regulations. Title 10, Part 20, "Standards for Protection Against Radiation." Superintendent of Documents, GPO, Washington, DC, 20555, 1976.

(19) Directorate of Regulatory Standards, Measurements of Radionuclides in the Environment, Sampling and Analysis of Plutonium in Soil. Regulatory Guide 4.5, U.S. Nuclear Regulatory Commission, Washington, DC, May 1974.

(20) Directorate of Regulatory Standards, Measurements of Radionuclides in the Environment, Strontium-89 and Strontium-90 Analysis. Regulatory Guide 4.6, U.S. Nuclear Regulatory Commission, Washington, DC, May 1974.

(21) Directorate of Regulatory Standards, Environmental Technical Specifications for Nuclear Power Plants. Regulatory Guide 4.8, U.S. Nuclear Regulatory Commission, Washington, DC, December 1975.

(22) J. H. Harley, Ed., HASL Procedures Manual. HASL-300, Supplements 5, Health and Safety Laboratory, HASL-300 Rev, New York, NY, August 1977.

radioactivity. With the existence of annually recorded monitoring data and established sampling and measurement techniques, the ability to identify radioactive species and verify the radioactive contamination levels that correspond to the disposition criteria listed in Tables 8.2-2, 8.2-5, 8.2-6, and 8.2-7 should already exist. However, it is doubtful that these levels could be detected with field instrumentation on the site at the lower limit of the range of assumed maximum annual dose, i.e., 1 mrem per year, at the 95% confidence limit. Therefore, a laboratory technique may have to be used to accurately determine both the mixture and quantity of radioactivity present in the station environment.

The Lower Limit of Detection (LLD) is defined in Regulatory Guide 4.16[23] as being the smallest concentration of radioactive material in a sample which has a 95% probability of being detected above the system background. For a particular counting system, the LLD is mathematically expressed by:

$$LLD = \frac{4.66\ Sb}{3.7 \times 10^4\ E\ V\ Y\ \exp\ (-\lambda\ \nabla\tau)} \qquad 8.3$$

where:

4.66 = a factor relating the 95% confidence limit of a one sided confidence factor for measurements where the background counting time equals the sample counting time.

LLD is the lower limit of detection, µCi/ml

S_b is the standard deviation of the instrument background counting rate, counts/second

3.7×10^4 = the number of disintegrations per second per µCi

E is the detector counting efficiency, counts observed per disintegration

V is the sample volume, ml

[23] Directorate of Regulatory Standards, Measuring, Evaluating, and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Airborne Effluents from Nuclear Fuel Processing and Fabrication Plants. Regulatory Guide 4.16, U.S. Nuclear Regulatory Commission, Washington, DC, March 1978.

Y is the fractional radiochemical yield; only applies when a radiochemical separation is done on the sample

λ is the radioactive decay constant for the particular radionuclide, seconds^{-1}

$\nabla\tau$ is the elapsed time between sample collection and counting.

The values of these parameters should be based on the actual characteristics of the system used, not on theoretically predicted values.

The LLD will vary with the type of instrumentation used, mixture of radionuclides in the sample, counting time selected, the sample size and the counting geometry. The LLD levels for samples containing single or simple parent-daughter radionuclide pairs, using Sodium iodide (NaI) detectors and other laboratory methods, are listed in Table 8.4-1.[21]

TABLE 8.4-1. Detection Capabilities for Environmental Sample Analysis[a]

	Lower Limit of Detection (LLD)[b]		
Analysis	Water (pCi/)	Vegetation (pCi/kg, Wet)	Soil (pCi/kg, Dry)
^{3}H (H_2O)	300	300[c]	-[d]
^{54}Mn	15	150	50
^{59}Fe	30	300	100
$^{58,60}Co$	15	150	50
^{65}Zn	30	300	100
^{89}Sr[c]	10	10	150
^{90}Sr[c]	2	2	30
^{95}Zr-Nb	10	150	100
^{106}Ru-Rh	10	150	100
^{129}I[c]	2	10	---
^{131}I[c]	0.4	2	---
$^{134,137}Cs$	15	150	100
^{140}Ba-La	15	150	100
U[c]	2	50	30
Pu-Alpha[c]	0.01	5	1

(a) This table is based on similar values given in Regulatory Guide 4.8,[21] with adjustments and additions reflecting current experience at a commercial radioanalytical laboratory.

(b) The normal lower limit of detection is defined in HASL 300, Appendix D (Rev. 8/74),[24] at the 95 confidence level. The LLD for radionuclides analyzed by gamma spectrometry will vary according to the number of radionuclides encoutered in environmental samples.

(c) After chemical extraction.

(d) None reported.

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(21) Directorate of Regulatory Standards, Environmental Technical Specifications for Nuclear Power Plants. Regulatory Guide 4.8, U.S. Nuclear Regulatory Commission, Washington, DC, December 1975.

With mixtures of fission products such as those assumed in this report, it is expected that the LLD for NaI detectors will be larger than those listed in Table 8.4-1. This is due to interference problems in differentiating between numerous gamma or beta radiations of similar energies. As a result, it is unlikely that radioactive contamination levels corresponding to the lower limit of the assumed range of the maximum annual dose i.e., 1 mrem per year can be detected with either field instruments or NaI detection systems. In this situation, a more sophisticated detection system, such as germanium-lithium [Ge(Li)] semiconductor detectors may be required. Typical values for the LLD reported for mixed fission-product samples on air filters for this system are listed in Table 8.4-2.(24) These values are for counting times of 1000 minutes with an absolute detector efficiency of 1.2% for ^{137}Cs.

TABLE 8.4-2. Lower Limits of Detection for a Mixture of Fission Products Using a Typical Ge(Li) Detection System

Radionuclide	dpm/Sample	Radionuclide	dpm/Sample
^{7}Be	68	^{106}Ru	68
^{54}Mn	4	^{125}Sb	21
^{57}Co	3	^{131}I	7
^{58}Co	4	^{137}Cs	7
^{60}Co	5	^{140}Ba	5
^{65}Zn	9	^{141}Ce	5
^{88}Y	5	^{144}Ce	24
^{95}Zr	11	^{147}Nd	59
^{103}Ru	8		

NOTES: 1. The sample was in a 5 cm diameter by 2.5 cm deep (2" diameter by 1" deep)
2. The counting time was 1000 minutes.
3. The absolute efficiency for ^{137}Cs for this sample geometry was 1.2%.

Using Ge(Li) detectors for gamma spectra measurements gives both qualitative and quantitative information about environmental samples. To assure compliance, this technique may be required to assess samples with activities corresponding to maximum annual doses of less than 10 mrem per year.

(24) J. H. Harley, ed., HASL Procedures Manual. HASL-300, Supplement 2, Health and Safety Laboratory, New York, NY, August 1974.

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9.0 DECOMMISSIONING ACTIVITIES

Two approaches to the decommissioning of the reference PWR nuclear power station have been selected for evaluation in this study: Immediate Dismantlement, and Safe Storage with Deferred Dismantlement, as defined in Section 4. An outline of the major components of a Master Decommissioning Plant (MDP) for each of these approaches is presented in this section, together with discussions of the major work items in each plan.

9.1 DECOMMISSIONING BY IMMEDIATE DISMANTLEMENT

In choosing to decommission his facility by immediate dismantlement, the owner trades significant radiation exposure accumulations to the workers for the relatively rapid release of the site and facility for unrestricted use. The program plan for accomplishing the job, the estimated staffing requirements, and the postulated work schedules and sequences are presented.

9.1.1 Program Plan

The dismantlement of a reference PWR facility can be divided into five general areas of effort: Planning and Preparation, Decontamination, Disassembly and Transport, Demolition, and Site Restoration. In any job of this magnitude, many of these areas of effort are going on simultaneously in different sections of the facility. To minimize conflicts and accidents, a well-defined sequence and schedule for disassembly of the various portions of the plant must be created and followed carefully. The major activities considered in this study of dismantling a PWR facility are illustrated in Figure 9.1-1.

9.1.1.1 Planning and Preparation

Essential to the results of this study is the assumption that the facility owner/operator is also the prime contractor the the dismantling work. Approximately 2 years prior to final reactor shutdown, work will begin in the engineering and operations departments of the utility organization to prepare the planning and paperwork needed to convert an operating license to a possession-only license following final reactor shutdown. An important part of the

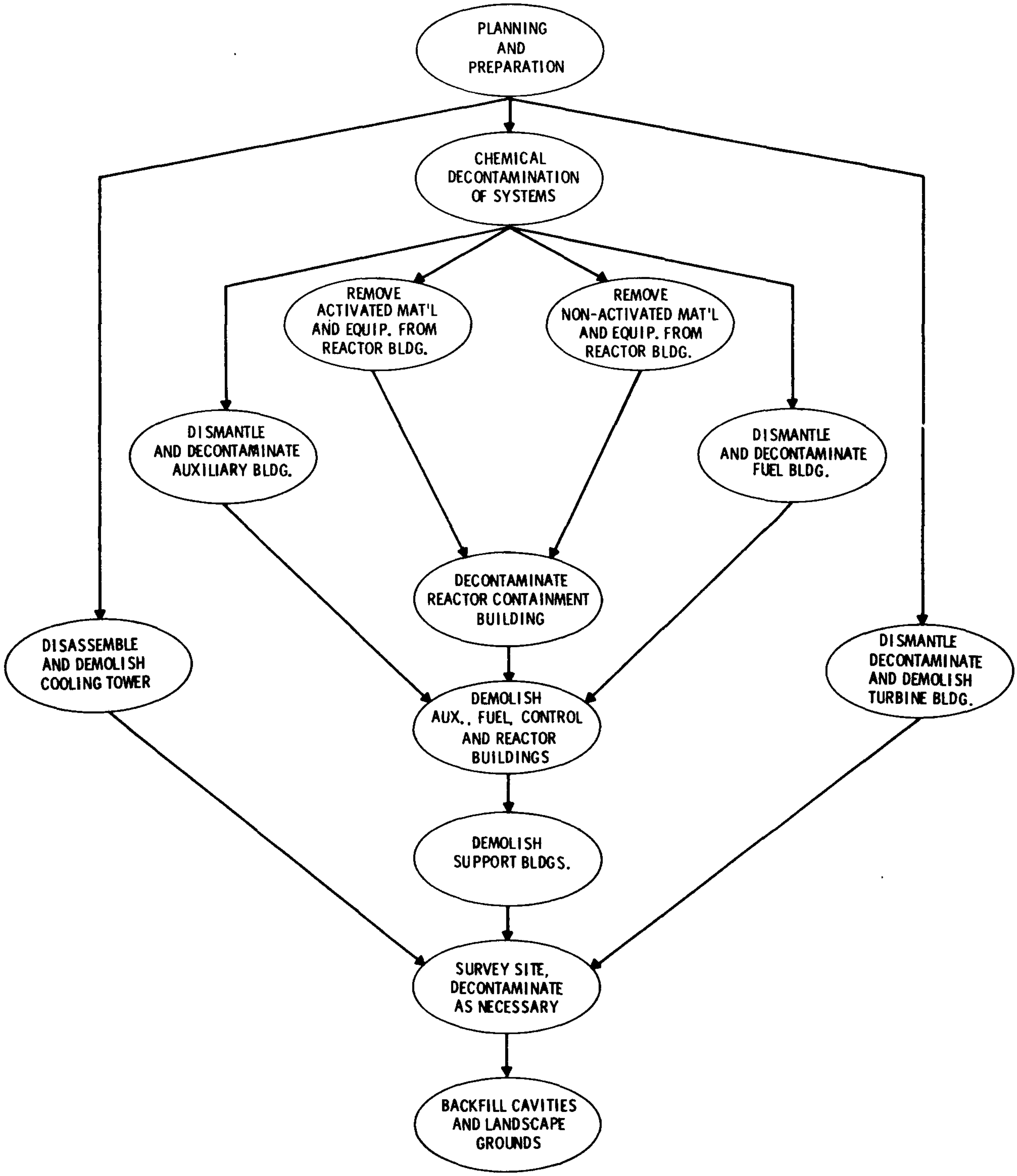

FIGURE 9.1-1. Sequence of Major Dismantlement Activities

planning involves a review of all regulations and guides applicable to decommissioning. A detailed compilation and review of such regulations has been presented in Section 5 of this report. The applicable regulations are also mentioned in the section to which they apply. For example, the following regulations and guides must be consulted during the planning and preparation phase of decommissioning.

- Regulatory Guide 1.86 <u>Termination of Operating License</u>
- 10 CFR 50.33 <u>Financial Qualifications</u>
- 10 CFR 50.59 <u>Authorization of Changes, Tests and Experiments</u>
- 10 CFR 50.82 <u>Application for Termination of License</u>
- 10 CFR 50.90 <u>Application for Amendment to License</u>
- 10 CFR 51.5 <u>Environmental Impact Assessment of Decommissioning</u>
- 10 CFR 20 <u>Standards for Protection Against Radiation</u>

Included in the above regulations are the requirements for preparation of changes in Technical Specifications, deleting those related to reactor operation; preparation and submittal of a Decommissioning Plan for NRC review and approval; preparation of detailed plans and procedures for chemical decontamination of intact systems; sectioning and disposal of the reactor vessel and its internals; and detailed sequences for component and system removal. In addition, design, procurement and testing of special devices and equipment must be initiated during the 2-year period prior to final reactor shutdown, to assure that work can proceed without undue delay after shutdown.

Creation of a decommissioning organization within the utility is initiated about two years prior to final reactor shutdown, with the structure and staffing requirements identified, and commitments obtained from key engineering and operating personnel to fill critical positions. Orientation and training of personnel identified as members of the decommissioning organization is carried on during the final year of reactor operation. A suggested organizational structure and staffing requirements are given in Section 9.1.3.

Selection of the various specialty contractors required for the dismantlement effort is accomplished during the final year of reactor operation. The types of specialty contractors anticipated to be required for dismantlement of a PWR facility are also listed in Section 9.1.3.

Also included in the Planning and Preparation efforts are such things as the shipment any spent fuel to a repository, and the removal from the Spent Fuel Pool of fuel storage racks in excess of those needed to hold the final core upon defueling the reactor. Removal of the excess storage racks is necessary to provide floor space in the Spent Fuel Pool for temporary storage of the sectioned vessel internals and the sectioned vessel walls while awaiting shipment.

Another preparatory step is the installation of a set of open top tanks in the former New Fuel Storage area to provide a final stage of electrochemical (electropolishing) decontamination for piping, sectioned tank walls, cavity liners, and similar materials that have significant salvage value. This decontamination facility is described in Appendix F.3.6.

The final preparatory step is the comprehensive survey of radiation dose rates and contamination levels within the facility, taken after final reactor shutdown and following the defueling of the reactor. This survey provides the baseline data for decisions about which systems need chemical decontamination, which areas need surface decontamination, which components may need temporary shielding to reduce personnel exposure during disassembly, and provides initial data on radiation dose rates likely to be encountered during the various dismantlement activities.

9.1.1.2 Decontamination

Decontamination can involve both chemical and physical attacks to remove radioactive materials. Chemical decontamination is performed on all systems directly connected with the reactor coolant system (RCS), and on all other systems that have measurable residual contamination. The objectives of the decontamination effort are two-fold: First, to reduce the radiation levels throughout the facility in order to minimize personnel exposure during disassembly; and second, to attempt to clean as much material as possible to unrestricted use levels, as defined in Section 8, thereby permitting salvage of valuable material and reducing the quantities of material that must be packaged and shipped to a disposal site (assumed to be shallow land burial in this study).

Chemical decontamination generally follows existing plant procedures and selected plant operating personnel are retained to assist in these operations. Records of any previous plant decontamination compaigns are reviewed to identify potential problem areas and to make maximum beneficial use of past plant experience.

Since this decontamination compaign takes place just prior to disassembly, damage to the systems and equipment from the decontamination solutions is of secondary concern. Solutions can be used that will remove the surfaces of the stainless steel in the tanks and piping, resulting in decontamination factors (DF) greater than the factors of 10 to 40, typical of chemical decontamination results from operating reactors.

Several chemical solutions can be used for in-place decontamination of the reactor coolant system. For this study, ethylenediaminetetraacetic acid (EDTA)/citric acid/oxalic acid has been selected, as is discussed in Appendix F.1.

Following the in situ chemical decontamination, the systems are disassembled. Equipment such as heat exchangers that cannot be satisfacorily decontaminated internally are sealed and externally decontaminated by washing in the Cask Wash Pit prior to shipment to a burial site. Valves, piping, and plate material are further decontaminated after disassembly in the final decontamination station which utilizes a large-scale electropolishing system, described in Appendix F.3.6. Electropolishing systems have been demonstrated on the laboratory scale to be capable of removing stubborn surface contamination to background levels in minutes.

In this study, no credit is taken for the potential effectiveness of the electropolishing system in reducing the quantities of contaminated material that must be packaged and shipped for burial, for several reasons. First, the systems have not been demonstrated in the type of large-scale usage postulated here. Second, the levels of residual radioactivity that would be permitted on material that is returned to the commercial stream are not yet defined by any regulation. Studies sponsored by the NRC are in progress to develop the bases for the establishment of regulations in this area, but the outcome of these studies is as yet unknown. Therefore, there is no way to know whether or not the electropolishing decontamination technique can clean material to release levels since the release levels are not defined. Third, depending upon the allowable levels of residual radioactivity (when defined), the costs of adequate surveys and possibly repeated cleanings of the material to get the material releasable may be greater than the savings achievable through release of the material. Therefore, in this study, all of the potentially contaminated material is assumed to remain contaminated, even after treatment, and is packaged and shipped to a shallow land burial site for disposal.

Decontamination of the floors, walls, and other surfaces within the facility structures is accomplished using standard techniques as described in Appendix F.1. A similar problem exists here as did with the piping material, i.e., what levels of residual radioactivity are allowable in structures that are to be released for unrestricted use, and how are these levels monitored for compliance with the standards? For this study it is assumed that acceptable methods of measurement are in place so that it can be determined readily when the residual radioactivity on or in the concrete of the structures is less than the allowable levels for unrestricted use. These same methods are applied to concrete that is contaminated by absorption of contaminated liquids into the pores of the concrete, and to concrete that contains radionuclides produced by neutron activation. Removal of the surfaces of concrete walls and floors is a relatively time-consuming operation. Therefore, the surface removal method is assumed to remove most of the contaminted material in one operation, with repeated operations needed only in isolated instances. It is further assumed that the cost of decontamination of these surfaces is essentially independent of the level to which it must be decontaminated, as long as that level is one that produces an annual radiation dose to the maximally-exposed individual in the range of 1 to 25 mrem, as discussed in Section 8.

The removal of the contaminated or activated concrete is accomplished using mechanical (jackhammers, rocksplitters) and/or explosive techniques. The removed rubble is packaged as radioactive waste and shipped to a licensed burial site.

9.1.1.3 Disassembly/Transport

Disassembly and disposal of contaminated and potentially contaminated equipment and materials is accomplished by the utility's decommissioning staff. In general, the disassembly work begins in the containment building and progresses through the fuel and auxiliary buildings in sequences designed to assure that essential services remain in place as long as they are needed. As a given section of a building is disassembled, the extensions of the essential services into those areas are removed.

The exact removal sequences within a given system is dictated by accessibility and the anticipated personnel exposures during removal. Where possible, items that contribute significantly to the general level of exposure in the work area are either removed first, or are temporarily shielded while the work goes on. Systems are unbolted at flanges when possible, and cut into manageable sections when necessary, using an appropriate cutting device (plasma torch, arc saw, oxyacetylene torch, power saw, or shaped charge explosives). Piping is cut into lengths compatible with standard-size waste disposal boxes and within the capability of the final decontamination station. Similarly, tanks and pool liners are cut into plate segments appropriately sized. For this study, all initially contaminated materials are assumed to remain contaminated to greater unrestricted use levels, even after final decontamination, and are packaged for disposal as radioactive waste.

Packaging of contaminated materials for disposal is accomplished in accordance with DOT regulations published in 49 CFR, Parts 173 through 178, NRC regulations published in 10 CFR, part 71 and Regulatory Guide 7.1. Containers are lined with shielding material when necessary to reduce surface dose rates to acceptable levels. Some items such as heat exchangers, may have openings welded shut and shipped using the outer shell of the exchanger as the packaging.

Shipping of packaged contaminated materials from the facility to a waste burial site is accomplished using trucking companies that specialize in transporting special materials. The volume of these materials to be transported and the number of shipments required are estimated in Appendix G.4.2, and the costs are summarized in Table 10.1-4.

The final step in the dismantlement of a building is a comprehensive radiation survey to insure that the radioactivity remaining on any materials in the structure is less than the amount allowable for unrestricted use as defined in Section 8. Surveys of the surfaces of the site are also required to assure that the accumulation of radioactive releases from the operating reactor during 40 years of operation on the site surfaces is sufficiently small to permit unrestricted use of the site. Decontamination of portions of the site is accomplished by removing and packaging a few centimeters of soil surface or sections of paving material from those areas requiring decontamination. When the facility and site can be released for unrestricted use, application is made to NRC to terminate the possession-only license.

9.1.1.4 Demolition

The NRC's responsibility is to assure that the public is protected from excessive exposure to radioactivity from nuclear facilities. Once the site and the facility have been decontaminated to levels that permit unrestricted use of the property, the nuclear license is terminated and the NRC's responsibilities at the station are also terminated. There are no provisions in any regulations that imply that the decontaminated structures must be demolished and the site restored to pre-facility conditions. Therefore, demolition and site restoration are not required to complete decommissioning. For completeness, the costs and time required for demolition of the decontaminated structures have been estimated and the costs are explicitly included in the total cost for decommissioning, developed in Section 10 and Appendix G.

Once a building's interior is dismantled and decontaminated to levels of radioactivity that are releasable, as defined in Section 8, it is turned over to a demolition contractor for removal of the structure. Normally no contaminated materials will remain in a structure when it is released for demolition. However, in some instances, contaminated piping or similar material that is imbedded in the structure will be sealed during dismantling, identified and removed during demolition.

All above-grade structures are removed. Below-grade structures are demolished to at least 1 m. below grade level with the base floors broken up sufficiently to permit normal drainage of surface waters through these floors. Large quantities of uncontaminated rubble are used in backfilling the below-grade cavities, thus reducing the quantities of fill material that must be hauled in to complete the site restoration. Uncontaminated rubble in excess of that required to backfill the below-grade cavities are transported to a local landfill for disposal.

9.1.1.5 Site Restoration

The amount of effort expended in site restoration is determined by the planned future usage of the site by the owner. For this study, it is assumed that the land is restored to a reasonable facsimile of the conditions that existed before the facility was constructed. Other choices for land usage, such as the building of new facilities on the site or the use of existing structures would have significant impacts on the nature of the restoration work.

This work is performed by fixed-price contractors. Sufficient top soil is supplied to adequately cover the back-filled cavities and to restore the land contours to approximately those that existed before plant construction. Planting of native vegetation on the restored land surfaces is accomplished to complete the task of returning the site to approximately its original condition.

9.1.1.6 Quality Assurance Program

An extensive quality assurance program is carried on throughout the decommissioning effort, to assure that all applicable regulations are met, to assure that the work is performed according to plan, to assure that the work does not endanger public safety, and to assure the safety of the decommissioning staff.

During the 2-year period prior to shutdown, QA personnel are active in the following areas:

- Review decommissioning plans for quality assurance involvement.
- Prepare inspection/test procedures as work plans are developed.
- Review designs of test equipment for quality input.
- Order any inspection/test equipment required to perform quality assurance/quality control function.
- Receive procured equipment, verify acceptance.

- Qualify suppliers for fabrication of radioactive shipping containers.
- Prepare inspection/test procedures to be imposed on subcontractors.
- Prepare inspection plans for shipment of radioactive materials, containers, trucks, etc.
- Finalize formal Quality Assurance Plan.

The QA efforts during the actual dismantlement period include the following:

- Perform QA functions for procurements.
- Qualify suppliers.
- Audit all program activities.
- Monitor performance of Specialists, Utility Operators, Laborers, Craftsmen, and Health Physics Technicians for compliance with work procedures.
- Verify compliance of radioactive shipments with appropriate procedures and regulations.
- Perform dimensional, visual, nondestructive examinations or other required inspection services to assure compliance with work plans.
- Maintain auditable files on the QA audits.
- Prepare a final report on overall performance of the dismantlement program with regard to the QA function.

A more detailed review of the anticipated elements of an appropriate Quality Assurance Program for the dismantlement effort is given in Appendix F.4.

9.1.1.7 Environmental Surveillance Program

An abbreviated version of the environmental monitoring program carried on during plant operation is continued during the dismantlement period. The purpose of the program is to identify and quantify any releases of radioactivity to the surrounding areas resulting from the dismantlement activities. The proposed program, detailed in Appendix F.5, is sufficient to permit evaluation of any significant releases. For emergency situations involving releases from events such as fires or malicious acts that may necessitate prompt emergency action to minimize the risk to the public, additional short-term surveillance efforts will be required.

After dismantlement is complete, a reduced one year follow-up program of environmental monitoring will be carried out by the same organization that performed the earlier programs.

9.1.2 Work Schedule Estimates

A proposed overall schedule and sequence of events for the dismantlement effort is presented in Figure 9.1-2, based on detailed schedules developed in Appendix G.2. Initial work begins about 2 years prior to final reactor shutdown, with 1) preparation of a decommissioning plan for NRC approval, 2) preparation of the revisions to the facility Technical Specifications necessary to change from an operating license to a possession-only license, 3) preparation of the data needed to make an assessment of the environmental impact of the dismantlement work, 4) preparation of detailed work plans and procedures for accomplishing the dismantlement of the facility, 5) design, procurement and testing of all special equipment needed for dismantlement, and 6) selection and training of the personnel for the decommissioning staff.

Following shutdown, the reactor is defueled and disabled as required to obtain the possession-only license. The spent fuel is shipped to an off-site location after the initial 90 day cooling period. Initial efforts center around the chemical decontamination of the Reactor Coolant System (RCS), the Chemical Volume Control System (CVCS) and related systems. Dismantlement begins with the reactor vessel and the interior of the reactor containment, and progresses through the various supporting structures such as the Condensate Demineralizer Building, the Fuel Building, the Auxiliary Building, the Control Building, and the Turbine Building. Demolition of the Cooling Tower can progress in parallel with the disassembly and decontamination of the other structures. Removal of all remaining structures, backfilling, grading and landscaping of the site completes the program. As can be seen in Figure 9.1-2, about 4 years of effort will be required after reactor shutdown to complete the dismantlement effort. The estimated costs for accomplishing the dismantlement are presented in Section 10.

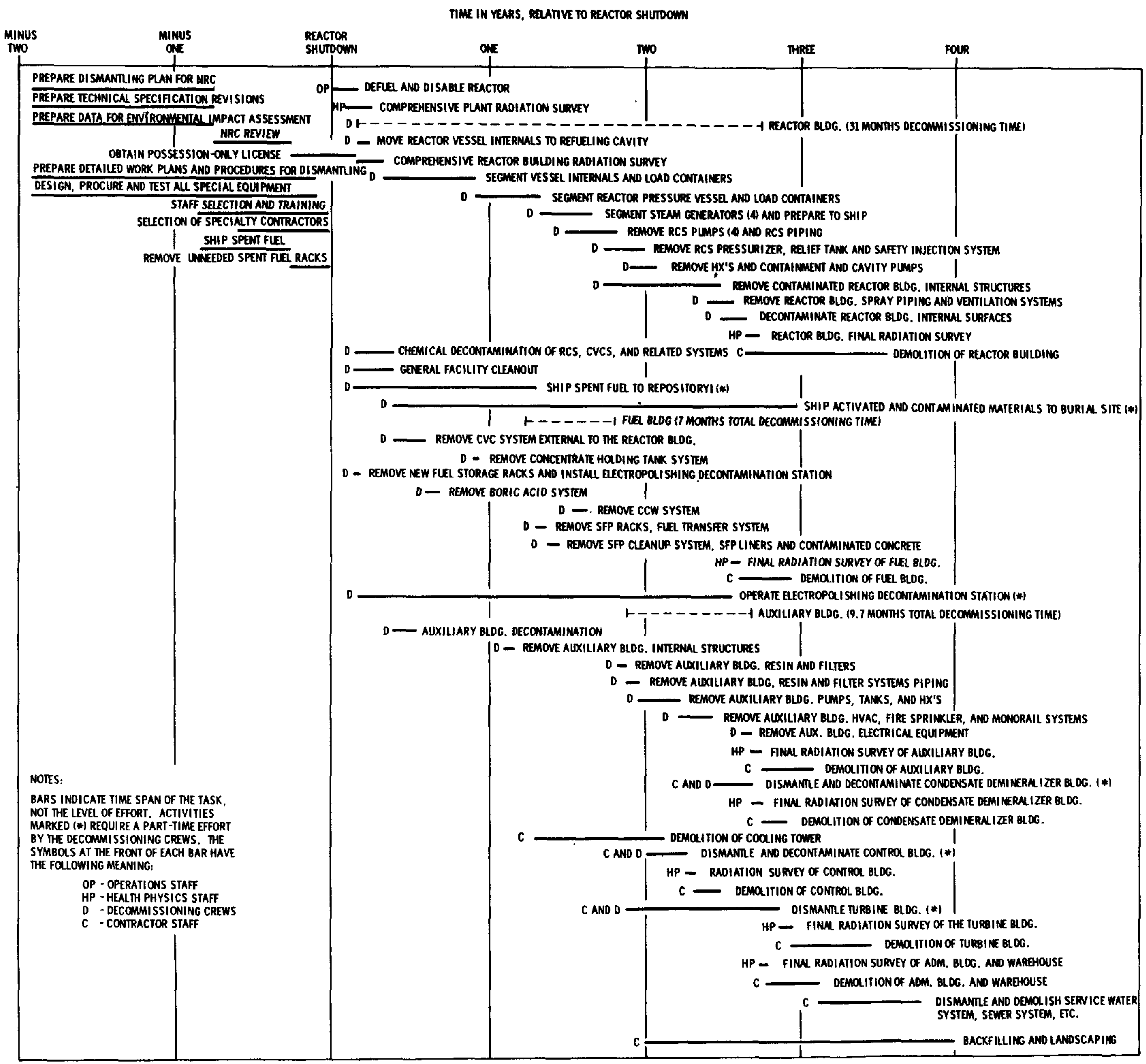

FIGURE 9.1-2. Schedule and Sequence of Dismantlement Events

9.1.3 Decommissioning Staff Requirements

The manpower requirements set forth in this section are for the decommissioning staff only. The many utility staff members who may contribute during the 2 years of planning and preparation prior to final reactor shutdown as part of their normal work assignments are not explicitly included.

The staff is sized and structured on a two-shift, 5-day week. Certain operations such as chemical decontamination and security will be carried out on three shifts. A suggested organizational structure is shown in Figure 9.1-3. The personnel are drawn from the operations and maintenance staff at the facility, to the maximum extent possible, to capitalize on their detailed knowledge of and familiarity with the facility. Specialists and consultants are brought in as required to assist the dismantling staff.

The decommissioning staff organization for the reference PWR shown in Figure 9.1-3 consists basically of two parallel branches reporting to the Decommissioning Superintendent. The operational branch, under the Decommissioning Engineer, plans and carries out the actual decommissioning activites. The safety branch, under the Health and Safety Supervisor, plans and conducts the industrial and radiological safety programs. The duties of key individuals on the decommissioning staff are as follows:

- Decommissioning Superintendent is responsible to Corporate Management for the complete coordination and supervision of all decommissioning activities. He directs the Decommissioning Engineer and Health and Safety Supervisor to assure that the decommissioning plan is developed and implemented in a safe and cost-effective manner. He provides necessary liaison with regulatory agencies. Security, Quality Assurance, and Contracts and Accounting supervision also report to the Superintendent.
- Health and Safety Supervisor (generally a Senior Health Physicist) advises the Decommissioning Superintendent and is responsible for matters of industrial and radiation safety. With the help of the Industrial Safety Specialist he develops and implements the industrial safety program. Additionally, he develops and implements the radiological safety program and environmental survey program, maintains occupational radiation exposure records, and supervises the Health Physics Technicians.

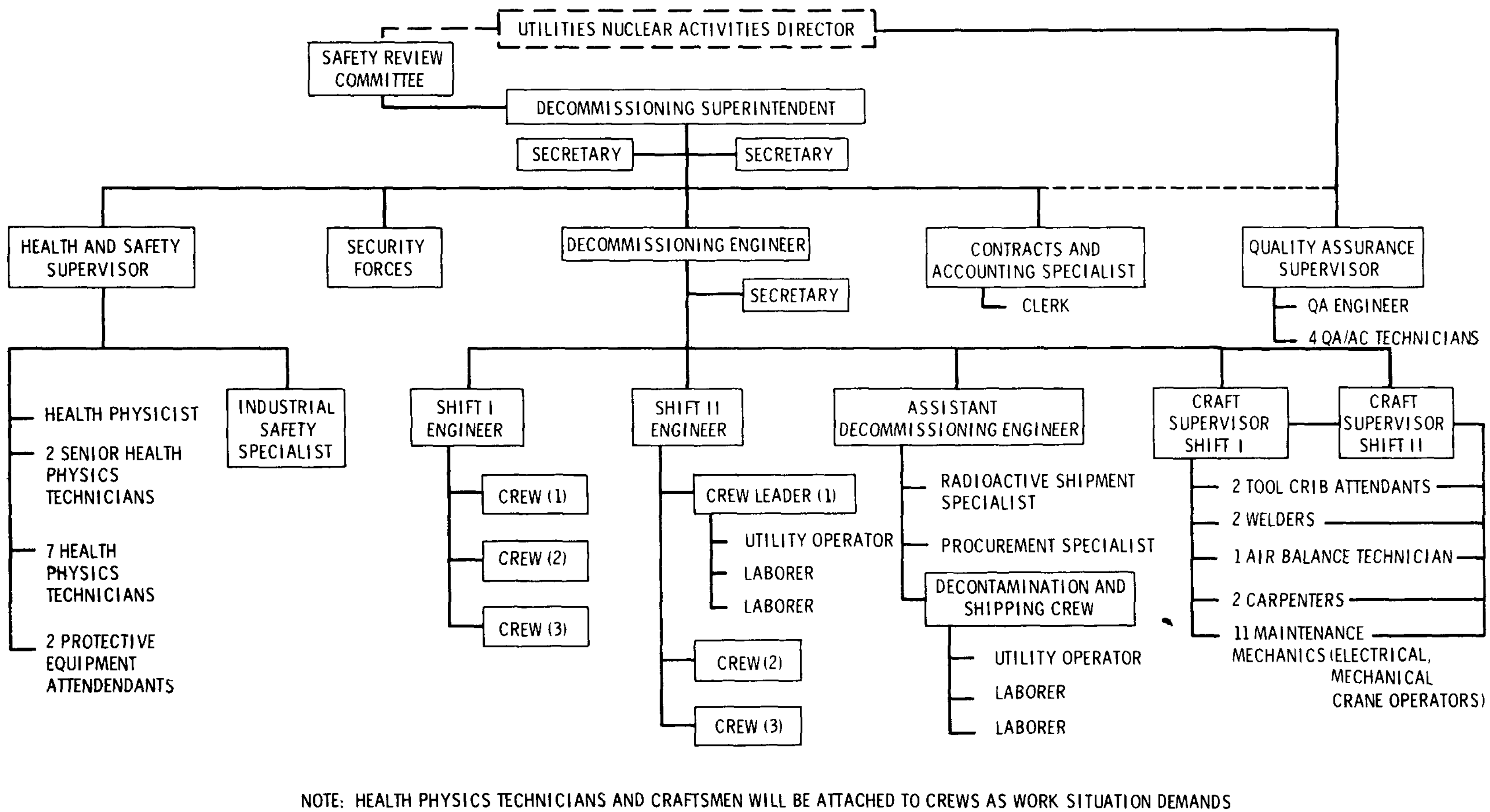

FIGURE 9.1-3. Suggested Organization for Decommissioning by Immediate Dismantlement

- Health Physics/Laboratory Technicians conduct on-the-job radiation dose rate measurements and ensure compliance with radiation working procedures.

  Additionally, they operate the plant laboratory facilities and conduct the onsite surveillance programs including sampling and analysis.

- Security Force Supervisor is responsible for all site security matters. He supervises the security forces during decommissioning.

- Decommissioning Engineer is responsible for planning, coordinating and carrying out the decommissioning activities in a safe manner. He provides engineering services and detailed procedures necessary to implement the decommissioning plan. He is responsible for the prepearation of all routine and special reports and compiles a chronological history of the entire project.

- Craft Supervisor is responsible for operating and maintaining plant equipment and services required to remain in operation during part or all of the project. He ensures that plant operating areas are maintained by qualified personnel.

- Project Accountant maintains a complete record of all costs incurred during the decommissioning activities. He disburses funds with approval of the Decommissioning Superintendent and the Decommissioning Engineer.

- Assistant Decommissioning Engineer assists the Decommissioning Engineer in developing detailed working procedures and supervises the Decontamination and Shipping crew. He writes specifications for special equipment and tools that must be procured or fabricated to carry out the project and prepares routine and special reports as requested by the Decommissioning Engineer.

- Crew Leaders report to the shift Engineers. They supervise working crews performing the decommissioning activities.

- Quality Assurance Supervisor/Engineer prepares the quality assurance plan and works with the Decommissioning Engineer to implement the quality assurance program. He reports directly to corporate headquarters in quality assurance matters. The Quality Assurance Supervisor is responsible for audit records, job performance and the assurances that the established safety review procedures are followed in all decommissioning activities.

- Safety Review Committee advises Corporate Headquarters (Utilities Nuclear Activities Director) and the Decommissioning Superintendent on safety-related matters. It is composed of six voting members-two from corporate headquarters and four independent consultants. The Decommissioning Superintendent, Quality Assurance Supervisor, Decommissioning Engineer, and Health and Safety Supervisor are nonvoting members. A majority of the independent consultants must agree on the resolution of all safety-related issues. The Safety Review Committee will generally meet about once a month during the active dismantlement phase. All decisions of the Safety Review Committee are implemented by the Decommissioning Superintendent.

The basic working unit is the crew, consisting of the Crew Leader (typically a Senior Operator), a Utility Operator, and two laborers, plus health physics technicians and craftsmen who are assigned to a crew as the work situation dictates. Each shift will have three crews, supervised by a Shift Engineer. To maintain continuity in ongoing jobs, Crew 1 of Shift I and Crew 1 of Shift II will be assigned to the same job, and similarly, for Crews 2 and 3 of each shift. One additional day-shift crew will handle final electropolishing decontamination and shipping functions.

The projected utilization of manpower during immediate dismantlement as a function of time is shown in Table 9.1-1. This summary is based on the detailed manpower assignments presented in Appendix G.2. The manpower shown in Table 9.1-1 exceeds the total manpower shown in Figures G.2-1 through G.2-4 in Appendix G.2. This additional manpower is utilized in planning and preparation, training activities, preparation of reports, and numerous small unspecified work items.

Specialty Contractors

Specialty contractors will be employed to perform unique services during decommissioning. Principal among these contractors are:

- Temporary waste solidification support, a transportable evaporator-solidifier system and operating personnel, to provide additional waste handling capacity and final cleanup capability after the installed waste handling systems have been removed;

TABLE 9.1-1. Projected Utilization of Manpower During Immediate Dismantlement

Title or Function	Time in Years Relative To Reactor Shutdown -2	-1	1	2	3	4	Total Man Years
Decommissioning Superintendent	0.3	1	1	1	1	1	5.3
Decommissioning Engineer	1	1	1	1	1		5
Asst. Decommissioning Engineer	1	1	1	1	0.5		4.5
Health & Safety Supervisor	0.3	1	1	1	1		4.3
Industrial Safety Specialist	0.3	1	1	1	1		4.3
Health Physicist		0.5	1	1	1		3.5
Senior Health Physics Technician		1	2	2	1		6
Health Physics Technician		3	7	7	5		22
Protective Equipment Attendant			2	2	1.5		5.5
Quality Assurance Supervisor	0.3	1	1	1	1	1	5.3
Quality Assurance Engineer	0.5	2	1	1	0.6		5.1
Quality Assurance Technicians		0.5	4	4	3	1	12.5
Contract and Accounting Specialist	0.3	1	1	1	1	1	5.3
Accounting Clerk		0.3	1	1	1		3.3
Security Supervisor			1	1	1	1	4
Security Patrolman			15	10	8	8	41
Radioactive Shipment Specialist		1	1	1	1		4
Procurement Specialist	0.3	1	1	1			3.3
Craft Supervisor			2	2	2		6
Craftsman			16	16	7		39
Tool Crib Attendant			2	2	1.5		5.5
Shift Engineer			2	2	2		6
Crew Leader			7	7	5		19
Utility Operator			10	7	3.4		20.4
Laborer			18	18	11.2		47.2
Secretary	1	2	3	3	2	1	12
Safety Review Committee Consultants	0.5	0.5	0.5	0.5	0.2		2.2
Man Years/Year	5.8	18.8	103.5	95.5	63.9	14	
Cost (millions of dollars)	0.2653	0.7164	2.9456	2.7578	1.8975	0.4035	
Total Man Years	------	------	--------	-------	-------	-------	301.5

NOTE: The number of significant figures shown is for computational accuracy and does not imply precision to the nearest tenth of a man year.

- Explosives specialists, for removal of the activated portions of the biological shield and for cutting of pipes, plates, etc., by the use of shaped charges;
- Hauling contractors, for transport of packaged radioactive materials from the facility to a licensed burial site;
- Demolition contractors for demolition of the decontaminated structures, disposal of clean rubble, and backfilling of below-grade areas to grade level;
- Landscaping contractors, for final grading and planting of vegetation to complete the site restoration (if appropriate).

9.1.4 Essential Systems and Services

Certain of the facility systems and services must remain in place until all radioactive and/or contaminated material have been removed from the site, to assure that no significant quantities of radioactive or hazardous materials are released to the environs. Also, certain of these systems are needed to facilitate the cleanup and disassembly efforts. As areas within the facility are readied for demolition, the extensions of these services into those areas are deactivated and removed, while maintaining continuity of the services to the remaining work areas. The required support systems are listed in Table 9.1-2, together with the justification for retaining each system.

The principal cost item listed in Table 9.1-2 is electrical power. The operating plant load is about 45 MW, and the cold shutdown plant load is about 22 MW. By maximum load reduction efforts, the cold shutdown load can probably be reduced initially to about 11 MW. Use of the RCS pumps during chemical decontamination would add about 18 MW to that base load while the pumps are running. A 0.057 m^3/min (15 gpm) evaporator contributes about 2 1/2 MW of the base load, and a minimum water supply from the river requires about 1 1/2 MW. Making reasonable assumptions about how the electrical load is distributed across the various buildings at the site, and following the schedule of dismantlement given in Section 9.1.2, the power usage by year after shutdown is estimated to be 1) 96,400 MW-hr, 2) 93,600 MW-hr, 3) 42,500 MW-hr, and 4) 1000 MW-hr, for a total of 233,500 MW-hr during the dismantlement operations.

TABLE 9 1-2. Systems and Services Required During Immediate Dismantlement

System or Components	Justification
Electrical Power, including emergency diesel backup system	Required for HVAC, lighting and radiation monitoring
HVAC Systems	Required for ventilation and contamination confinement
Component Cooling Water	Required for secondary cooling water supply to spent fuel pool and boric acid and radwaste evaporators
Water Supply (service and domestic systems)	Required for decontamination, cleanup, fire protection, and general potable water usage
Fire Protection System (direction and suppression)	Required for health and safety.
Compressed Air Systems (control and suppression)	Required for operation of pneumatic controls, for operation of pneumatically operated tools
In-plant Communications Systems (telephones and intercoms)	Required to facilitate and coordinate activities
Radiation Monitoring Systems	Required for protection of personnel
Solid, Liquid and Gaseous Radwaste Systems (DRW and CRW)	Required for treatment and disposal of potentially contaminated liquids, gases and solids
Demineralizer System	Required for cleanup of water in Fuel Pool and refueling Cavity during vessel cutting operations
Sewer Plant	Required for sewage treatment

9.2 DECOMMISSIONING BY SAFE STORAGE WITH DEFERRED DISMANTLEMENT

The goal in the Safe Storage Mode for the decommissioning of the reference pressurized water reactor (PWR) is to achieve a condition that ensures that the safety of the public is not endangered by the residual radioactivity at the site with a modest expenditure of effort. Modifications to the facilities are limited to those that ensure the security of the buildings against intruders and those required to assure containment of radioactive or toxic material. It is not intended that the facility would be reactivated. To achieve this goal, the facilities are left structurally sound. There is a minimum removal of loose contamination and fixation and sealing of all remaining contamination. All systems and equipment that are not required to be in operation during the surveillance and maintenance period are deactivated. The Safe Storage preparations and the continuing care period that follows should be recognized as only an interim stage in the total decommissioning process. Current NRC philosophy encourages a decommissioning approach that ends in the termination of the facility's nuclear license and the release of the property for unrestricted use within a finite period of time. Thus, dismantlement is required eventually.

The Safe Storage period may vary from a few years to 100 years or longer, depending on such variables as:

- societal concerns
- the risk to the safety of the public
- the occupational radiation dose commitment
- the cost of further decommissioning efforts
- the needs and desires of the facility owner

The preparations for Safe Storage, when complete, essentially immobilize all remaining contamination and place the facilities in a condition amenable to routine surveillance and maintenance but generally unavailable for any other uses. Work sequences and procedures are presented in this section to achieve this goal. These sequences and procedures were developed under the assumption that the preparations commence immediately following reactor shutdown.

This assumption has the effect of maximizing occupational radiation exposure but generally does not affect the time and cost associated with placing the facility in Safe Storage.

It is estimated, based on detailed work plans, that approximately 34 months are required from the time of initiation of preparations for Safe Storage until the facilities enter the period of continuing care, which consists primarily of security, surveillance and maintenance. Eighteen months prior to final reactor shutdown are devoted to planning and preparation and 16 months following final shutdown are required to complete the preparations for Safe Storage. The time and work estimates assume reasonable success with a minimum of delays and/or major unanticipated problems.

9.2.1 Program Plan

A minimum of activities is used to place the reference PWR in Safe Storage. These activities include:

- chemical decontamination of the RCS, CVCS, and related inter-tied systems and removal and/or immobilization of accessible contamination in the rest of the facility--this includes the draining and isolation of all systems directly connected to the Reactor Coolant System (RCS).
- Offsite disposal of radioactive and nonradioactive wastes in approved burial facilities
- erection of physical barriers to prevent entry into radioactive and/or contaminated areas
- operation and maintenance of protective systems such as radiation, intrusion, and fire detection systems as well as electrical power distribution systems
- periodic surveys and inspections of the facility and site

The final plant condition is one with most of the transportable radioactivity either removed or immobilized, but with significant quantities of fixed radioactivity (millions of curies) remaining in the Reactor Building. The overall work schedule estimates developed for this study are given in Section 9.2.2.

All buildings and areas not specifically listed in the Safe Storage decommissioning plan are restored to unrestricted use conditions as defined in Section 8 such that either unrestricted access is possible for maintenance and routine inspections by authorized personnel in accordance with the requirements of Section 20.105 of 10 CFR 20 or the structures can be demolished and the areas released for public use, as discussed in Section 9.4 and Appendix H.5.2. The buildings and/or areas located outside of the Exclusion Area shown in Figure 9.2-1 are assumed to be in this category. The surveillance, maintenance, and security of the buildings that make up the Exclusion Area are subject to the provisions detailed in Section 9.2.1.4, Final Preparations for Safe Storage, Section 9.2.1.5, Environmental Surveillance Program, and Section 9.2.1.6, Security

The major benefits from placing the facility in Safe Storage are: 1) reduced radiation levels at a future date when further decommissioning activities are undertaken, 2) postponement of irrevocable decommissioning activities, 3) reuse of the accessible uncontaminated portions of the facility by the utility if so desired, and 4) low initial outlay of funds.

Activities at the site during the continuing care period that follows placing the facility in Safe Storage are limited primarily to building maintenance and radiation monitoring of the facility, and environmental radiation surveillance The facility is not manned on a continuous basis after being placed in Safe Storage but periodic surveillance and maintenance of the facility structures and of passive safety and security related systems are required. The outer perimeter site fence is maintained and no unauthorized entry is permitted. Detailed accounts of the decommissioning operations are stored at the facility and made a part of the public record. These accounts are required for use when final dismantlement of the facility is performed.

The facility is divided into five major sections for the purposes of this plan. These are:

- Reactor Containment Building
- Fuel Building
- Auxiliary Building
- Support Buildings

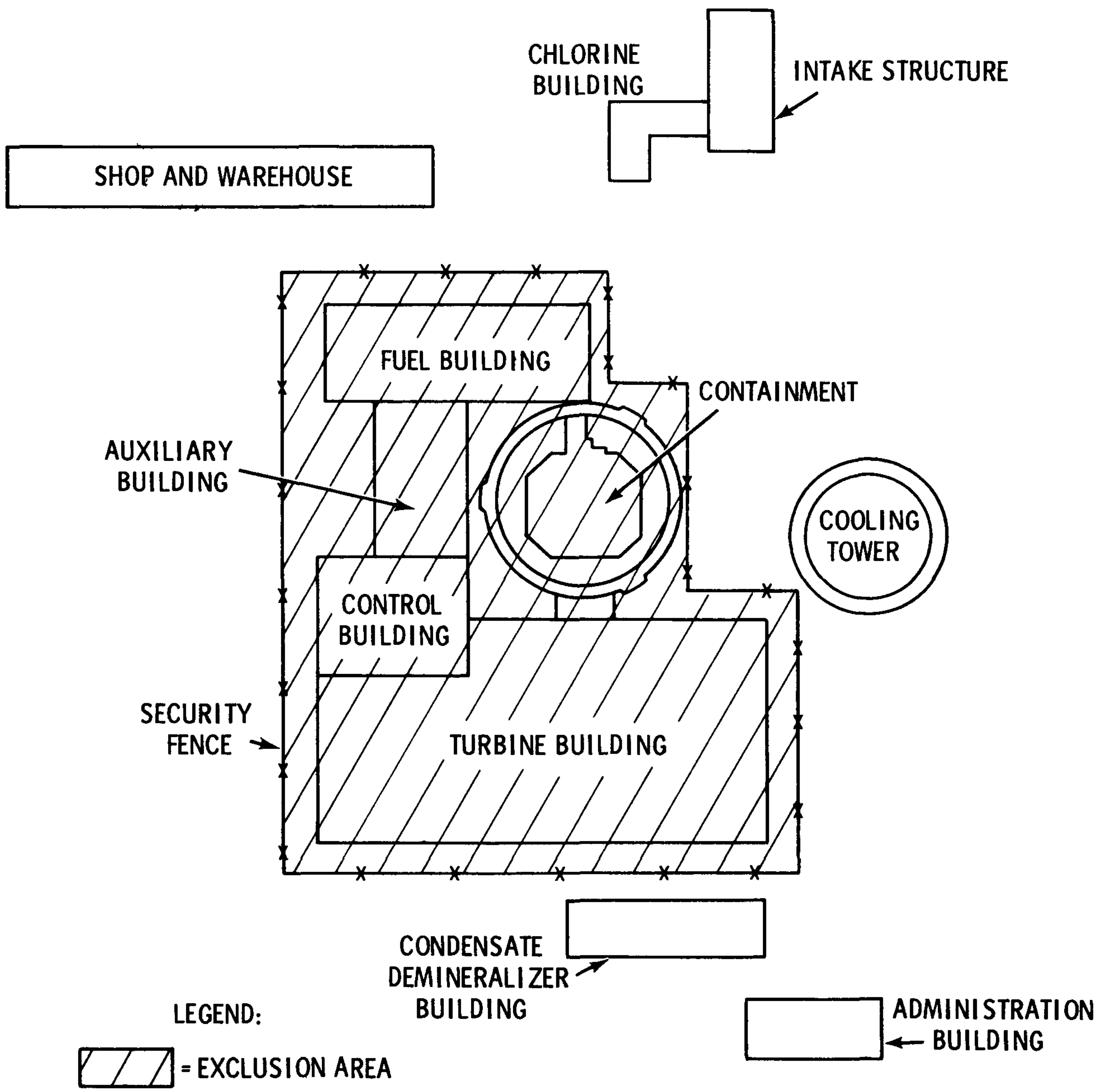

FIGURE 9.2-1. Plot Plan of the Exclusion Area and the Surrounding Area

- Facility Site - The site is surveyed and any areas outside the exclusion area found to have contamination levels greater than are permitted for unrestricted use as defined in Section 8 are assumed to be decontaminated by appropriate means. Within the exclusion area, contamination levels may exceed unrestricted use levels but the radioactivity is made non-dispersible.

The Safe Storage activities in each section of the facility are generally divided into the following six major phases:

- Planning and Preparation
- Chemical Decontamination
- Mechanical Decontamination and Fixing of Residual Contamination
- Equipment Deactivation
- Isolation of Contaminated Areas
- Final Preparations for Surveillance and Maintenance (Continuing Care).

Portions of these phases in various locations of the facility may overlap or proceed concurrently. When these activities are completed, the facility is placed under continuing surveillance and maintenance.

9.2.1.1 Planning and Preparation

The planning and preparation activities for Safe Storage are carried out concurrently with the final 18 months of facility operation. Work begins in the engineering and operations departments of the utility organization to prepare the paperwork needed to convert the operating license to a possession-only license following final reactor shutdown.

This phase includes all the activities required to prepare for Safe Storage. Current regulations and guides in place are:

- Regulatory Guide 1.86, Termination of Operating License
- Regulatory Guide 1.88, Collection and Storage of Nuclear Power Plant Quality Assurance Records
- 10 CFR 50, Rules of General Applicability to Licensing Byproduct Material

- 10 CFR 50.33, Financial Qualifications
- 10 CFR 50.59, Authorization of Changes, Tests and Experiments
- 10 CFR 50.82, Application for Termination of License
- 10 CFR 50.90, Application for Amendment to License
- 10 CFR 51.5, Environmental Impact Assessment of Decommissioning
- 10 CFR 50, Appendix B Quality Assurance Criteria for Nuclear Power Plants
- 29 CFR 1910, Occupational Safety and Health Act Standards--OSHA.

Included in the above regulations are the requirements for preparation of changes in Technical Specifications, (taking into consideration monitoring requirements and specifications for equipment that must be maintained, and deleting those related to reactor operation), and guidance for the preparation and submittal of a Master Decommissioning Plan for NRC review and approval. A detailed compilation and review of regulations and guides applicable to decommissioning is given in Section 5, Regulatory Considerations For Decommissioning. In addition, design procurement and testing of any special devices and equipment must be initiated during the 18-month period prior to final reactor shutdown, to assure that work can proceed without undue delay after shutdown.

Creation of a decommissioning organization within the utility is initiated, with the structure and staffing requirements identified, and commitments obtained from key engineering and operating personnel to fill critical positions. Orientation and training of personnel identified as members of the decommissioning staff are carried on during the final 10 months of reactor operation. The decommissioning staff draws on their own experience and on the experience of the operations staff to assist in the planning activities. A postulated organizational structure, with staffing requirements is given in Section 9.2.3.

Before the plant is shut down, a decommissioning plan and safety analysis report is prepared, detailed working procedures and safety requirements are developed, and cost estimates are made. A variety of predecommissioning activities are carried out as part of plant shutdown operations. These activities include removal from the site of bulk quantities of process chemicals, radioactive materials, nonessential equipment, and engineering review of effluent control systems that are necessary for decommissioning.

The Master Decommissioning Plan (MDP), including overall decommissioning criteria and work scope, is divided into manageable tasks and available decommissioning techniques are carefully reviewed and decisions are made on the general techniques to be used to accomplish each task. Detailed procedures are developed, including those for predecommissioning cleanup of the facility. Equipment and materials requirements and manpower, schedule and time estimates are prepared. The plan is documented in detail, necessary safety analysis reports are prepared, and all appropriate documents are submitted for approval of plant management and regulatory agencies.

It is expected that most of the planning and the actual decommissioning activities will be performed by plant operating and maintenance personnel. The various specialty contractors required for the decommissioning effort are selected during the final six months of reactor operation. The types of specialty contractors required to place the facility in Safe Storage are listed in Section 9.2.3.

A final preparatory step is the comprehensive survey of radiation dose rates and contaminated areas within the facility, taken after final shutdown, following defueling of the reactor. This survey provides the baseline data for decisions about which systems need chemical decontamination, which areas need surface decontamination, and which components may need temporary shielding to reduce personnel exposure. In addition, a comprehensive radiation survey of the site will be performed.

The decommissioning activities following reactor shutdown are analogous in many ways to a normal, well run reactor outage. For example, it is of paramount importance to recognize early any areas where additional manpower is required and to arrange for that manpower. Such arrangements should include contractual negotiations, including incentive pay, support facilities such as shop areas and eating facilities, training, indoctrination, and any special qualifications for individuals retained.

9.2.1.2 Facility Decontamination

Facility decontamination begins after the reactor is defueled, the comprehensive plant decommissioning radiation survey is completed and facility housekeeping is accomplished. Unnecessary noncontaminated equipment and materials are removed and inventories of flammable articles are reduced to minimum levels. Contamination that cannot be readily removed is fixed in place by painting over the contaminated area with a water-based coating. Radiation warning signs are affixed near the painted areas. The location and characteristics of each such area is noted in the permanent records of the decommissioning operation. Chemical decontamination of the RCS, CVCS and related inter-tied systems is accomplished as outlined in Appendix F.1.

Scheduled decontamination activities may proceed concurrently. The primary concern is to assure that no recontamination of clean areas occurs and that air leaving a given area flows through the existing facility filter system or, in the case of liquid effluents, through the existing contaminated waste system. The equipment/systems are isolated as indicated in Section 9.2.1.3, Equipment Deactivation and Isolation of Contaminated Areas.

All liquid radioactive wastes generated during decommissioning operations are sent to the plant liquid waste storage system or to other tanks that are designated for temporary storage of these solutions. The wastes are then processed through the waste concentration and solidification system. All systems designed to control the release of hazardous material to the environment or to noncontaminated portions of the facility are in operation during the chemical decontamination activities and subsequent waste processing. The radiation dose rates and contamination levels expected to be present in the facility prior to decontamination operations are described in Section 7 and Appendix C.

In the complex systems of the PWR, the Decontamination Factors (DFs) may vary up to factors of 50 or greater. For this study, an "average" conservative DF of 10 has been assumed. The effectiveness of the chemical decontamination as measured by the DF is expressed here as the ratio of the original level of radioactivity to the level present after chemical decontamination.

Decontamination of floors, walls, and other building surfaces is accomplished using direct worker contact decontamination techniques, where possible. Many areas are cleaned using simple "janitorial" techniques such as vacuuming, sweeping or scrubbing with cleaning agents compatible with the waste treatment system. Acid-proof sponges soaked in decontamination solution are used on small areas where chemicals are effective. Larger areas are decontaminated using a portable high pressure decontamination solution sprayer.

The decontamination and/or shielding of systems and components in a portion of the facility gives priority to radiation "hot-spots" on components and piping segments to reduce background levels and minimize personnel radiation exposure. Radiation dose level, accessibility, and radiation exposure during decontamination influences the selection of "hot-spots" to be decontaminated first. In general, these selections are field decisions based on the comprehensive radiation survey results.

External Decontamination of Ventilation Systems - The exhaust ductwork from the Reactor, Fuel Auxiliary Buildings is decontaminated as required. Decontamination procedures used during plant operations are generally followed. It is expected that the decontamination effort will consist primarily of hot water flushes to remove dirt and grease. Chemical solutions may be used if there is significant buildup of contamination. The first stage of the HEPA filters is replaced during these operations. Subsequent stages of HEPA filters are replaced only if replacement is required due to damage or high pressure drop.

Decontamination and Covering of the Spent Fuel Pool (SFP) - The Spent Fuel Pool is drained, decontaminated and covered after the last fuel shipment has left the site. Encapsulating the SFP and the storage racks contained therein allows for more economical long-range surveillance, maintenance and security of the Fuel Building.

During the planning and preparation stage, procedures and results from previous decontamination efforts are reviewed to obtain maximum benefit from previous experience. Current as-built drawings are reviewed to identify system deadlegs and to facilitate planning of system modifications required to achieve decontamination. Existing procedures are reviewed for applicabilit to the present effort and modified as necessary. A detailed step-by-step procedure is developed for the decontamination campaign, with check lists for valve settings, etc. to assure proper operation of the systems.

The SFP is decontaminated and secured by a specialty contractor. This includes draining, decontamination, stabilizing any surface contamination left on the fuel storage racks and other areas, equipment deactivation, and providing a carbon steel cover with HEPA filter over the SFP as shown in Figure 9.2-2.

Major decontamination is done using portable high pressure spray units. Manual techniques of decontamination such as swabbing and scrubbing with acid-proof sponges are used where necessary, and inaccessible locations such as portions of the fuel storage racks are sealed by painting.

As the SFP is drained, the walls and floor are sprayed with high pressure steam or water jets and painted as necessary to immobilize and/or fix residual contamination. The Fuel Transfer Tube opening is sealed after the Fuel Transfer Canal is drained. Radioactive particulates that have settled out on the SFP floor are removed with an underwater vacuum cleaner.

Final preparations for Safe Storage of the SFP include installation of the 0.635 cm (1/4-in.) thick carbon steel, welded cover with HEPA filtered vent plus the following items that are integral parts of the overall Fuel Building decommissioning activities:

- installation and/or upgrading of radiation alarms
- installing or relocating intrusion alarms
- installing high-security locks on all exterior and selected interior doors and
- performing a comprehensive radiation survey.

Final disposal of the SFP water and the smaller quantities of chemical solutions generated during decontamination is accomplished via the existing plant evaporation and solidification systems. A typical PWR radiation waste evaporation and solidification system is shown functionally in Figure 9.2-3.

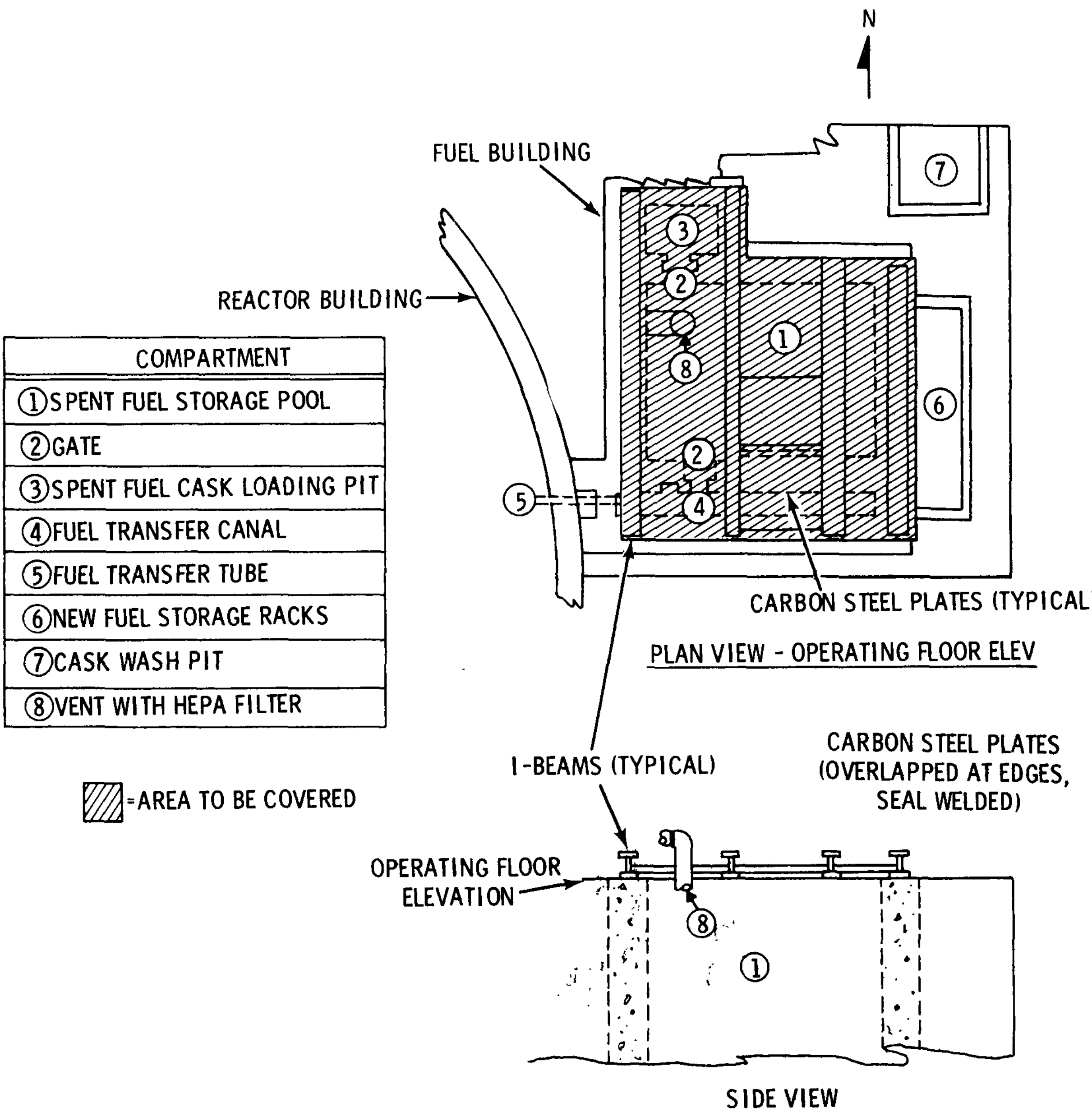

FIGURE 9.2-2. Proposed Carbon Steel Cover for the SFP

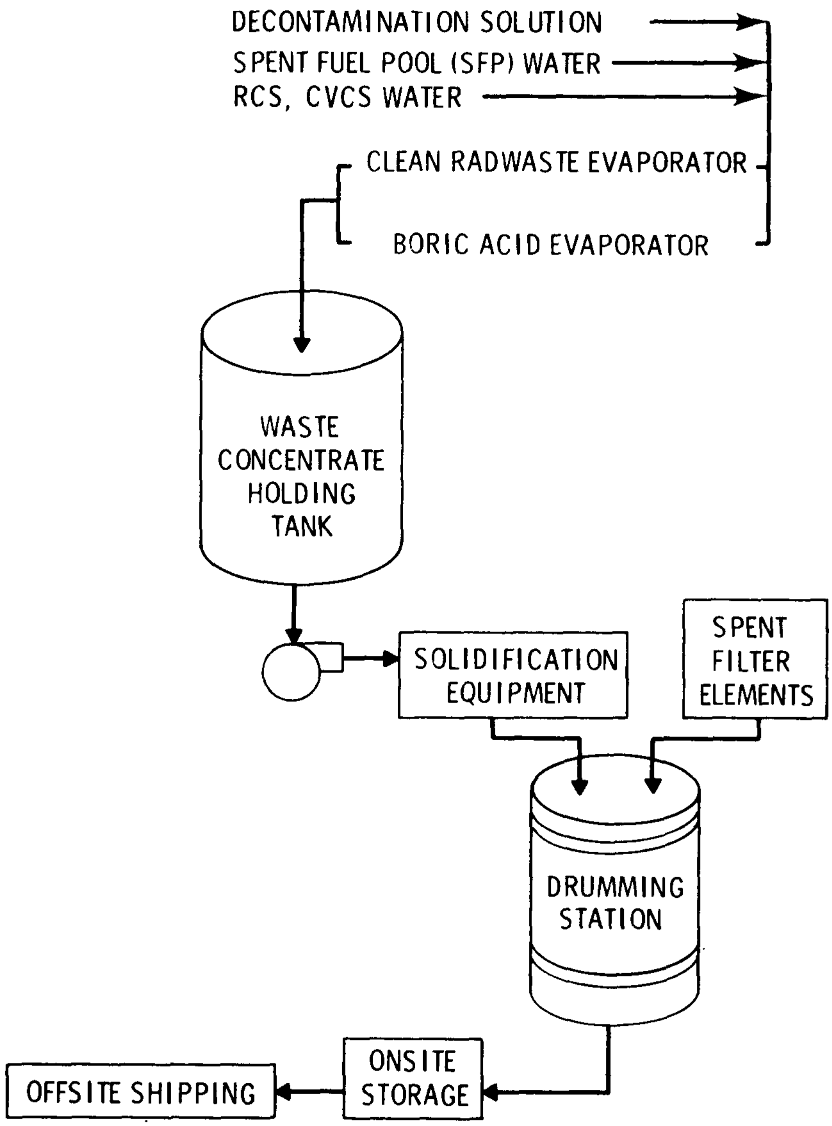

FIGURE 9.2-3. Typical PWR Radiation Waste Evaporation and Solidification

Mechanical Decontamination and Fixing of Residual Contamination - Mechanical decontamination of structures is carried out only in areas that contribute significantly (i.e. $\geq$5 mR/hr)[1] to the radiation exposure of surveillance and maintenance personnel such as hallways and corridors. Drilling and rock-splitting or jackhammering are some of the methods used to decontaminate

[1] U.S. Nuclear Regulatory Commission, Termination of Operating License for Reactors. Regulatory Guide 1.86, p. 1.86-2, June 1974.

these areas. The contaminated materials that are removed are packaged and either shipped to a burial site or placed in one of the areas that is isolated during the Continuing Care period. Combustible materials are packaged and shipped offsite for disposal. Typical mechanical removal techniques are described more fully in Appendices F.1.2 and F.1.3.

Some residual amounts of low level contamination may be present in areas outside the isolated areas. These areas typically contain amounts of radioactivity that do not contribute significantly to occupational radiation exposure levels in the facility (i.e. $\leq$5 mR/hr.). This contamination is immobilized by covering it with paint or other protective coatings to prevent the contamination from becoming airborne.

9.2.1.3 Equipment Deactivation and Isolation of Contaminated Areas - Only essential safety systems such as radiation detection alarms, security monitors, and fire detection and portable fire fighting equipment remain in operation during the surveillance and maintenance period. All other equipment and systems are placed in a condition that provides maximum safety with minimum maintenance. Whenever possible, equipment is left in a condition that permits salvage at a later date. Deactivation and isolation techniques include closing and securing installed valves, installing blank flanges and disconnecting electrical power and other utilities. Equipment deactivation procedures are coordinated with facility decontamination operations. In some areas decontamination must be carried out before equipment deactivation, while in other areas the opposite approach may be necessary. A safety audit of all systems is performed to assure that all flammable and other potentially hazardous materials have been removed. All deactivated equipment and systems are tagged for identification and status.

The particular method used to deactivate 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. (See Section 9.2.5 for systems retained). All equipment, valves, circuit breakers, etc., are tagged when deactivated. These tags identify the piece of equipment, the system it is in and its condition.

The first step in equipment deactivation is a safety audit of all pumps and pipes used for radioactive materials or chemicals to ensure that all hazardous or corrosive materials have been removed. Electrical service is disconnected from all pumps not required to be in operation during the surveillance and maintenance period.

Systems inside the Reactor Building, the Auxiliary Building, and the Fuel 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. All cranes are disabled by removal of their circuit breakers to prevent their unauthorized use during the surveillance and maintenance period. Other electrical equipment that should not be operated during the surveillance and maintenance period is disabled in a similar manner. Electrical service is disconnected from instrumentation not required to be in operation during the surveillance period.

Portions of the facility sections containing significant amounts of radioactivity are isolated by the installation of tamper proof barriers. Indirect access routes, however unlikely, are investigated from as-built drawings and sealed. Such routes may include, but are not limited to, access through large vessels, tanks, or large diameter pipes that could allow such trespass, willful or otherwise. Barriers are constructed by welding or by bolting and sealing steel plates to block potential pathways for unauthorized entry or migration of contamination. Because of its durability and flexibility, extensive use is made of polysulfide rubber as a sealant. Vents with HEPA filters are installed in isolated areas to allow for changes in air pressure and temperature; however, the ventilation systems are deactivated. A possible method of isolation is shown in Figure 9.2-4.

<u>Techniques for Sealing Contaminated Areas</u> - Portions of the facility that contain significant amounts of radioactive contamination during the surveillance period are 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. Possible methods for design and/or construction of these barriers are presented in this section.

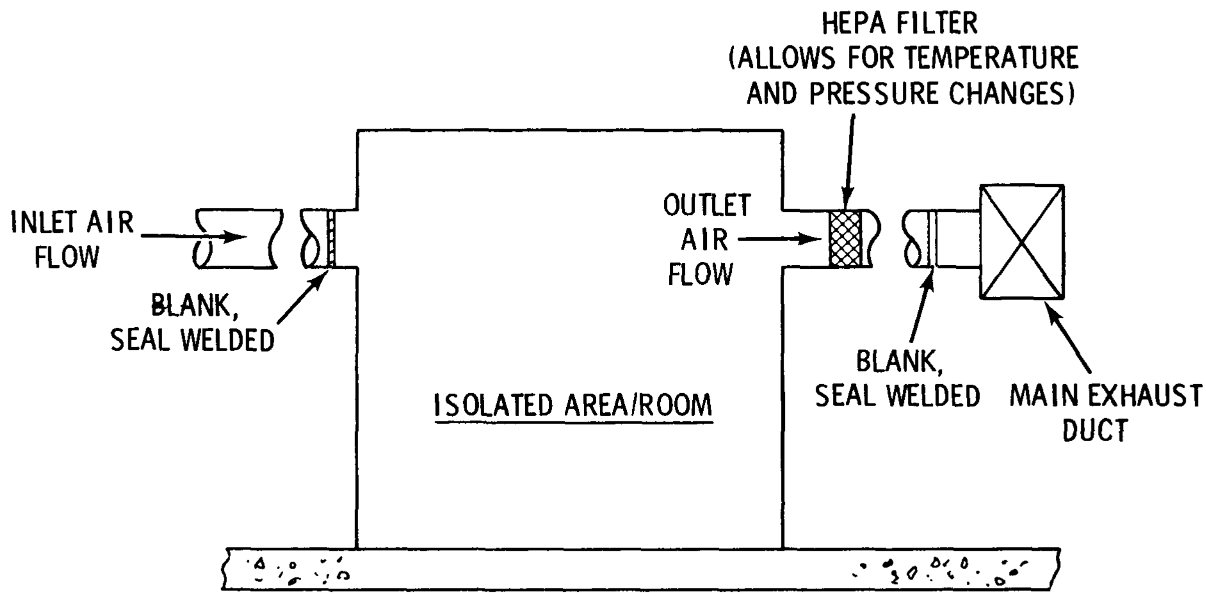

FIGURE 9.2-4. Possible Area Isolation Method

Besides acting as a contamination control barrier, the barriers are also designed to discourage unauthorized personnel entry into contaminated areas. Structurally substantial barriers are used, and extensive use is made of steel in constructing the barriers.

The preferred method of constructing the barriers 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 so as not to interfere with any fire alarm sensors that may be present in the ducts.

The polysulfide adhesive is 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 different rates of thermal expansion between two bonded surfaces without breaking the seal. This adhesive can be applied with a spatula or extrusion gun and will not run after application.

Doorways provided with removable concrete panels as shown in Figure A.2-15 of Appendix A.2.2 are protected by covering them with a steel plate sealed to the surrounding concrete with polysulfide adhesive. The plate is mounted on studs placed in the concrete. After the adhesive has been applied to the underlying concrete the plate is put in place and the nuts are tightened to produce an even seal. The nuts are welded to the studs to discourage unauthorized removal.

Metal doors are secured by welding them to the metal door frame. Spot welds are used where the intent is only to make the door inoperative. Continuous welds are made or sealant is used where an airtight seal is required. The perimeter of the door frame where it meets the surrounding concrete and other surfaces that can't be welded is sealed with the polysulfide adhesive.

A pressure equalization line is provided between the outside environment and the interior of each of the following buildings: Containment, Turbine, Control, Auxiliary, and Fuel. The pipes used for this purpose are provided with replaceable absolute filters. The lines prevent pressure differentials from changes in temperature and atmospheric pressure from developing between the inside of a building and the outside atmosphere.

9.2.1.4 Final Preparations for Safe Storage

Surveillance activities for each portion of the facility begin as soon as it is placed in Safe Storage. These activities include routine inspection, preventive and corrective maintenance on safety systems and a regular program of radiation and environmental monitoring. In addition to these routine tasks, a comprehensive inspection of the facility is performed annually by qualified professional third party inspectors. Any unusual or potentially unsafe condition detected during the surveillance program is corrected immediately.

The surveillance and maintenance programs are structured so that personnel inspect various portions of the facility on a routine basis. Radiation monitoring will be done at each pre-established surveillance point at least quarterly. These checks will be staggered so that the monitoring actually takes place over several days, distributed throughout the quarter. Preventive maintenance activities and routine equipment inspections are also distributed throughout the quarter. HEPA-filtered vents and physical barriers are inspected routinely and repaired as necessary. The fire alarms, radiation alarms and intrusion alarms operating in the facility during the surveillance period are monitored continuously at an off-site location. Routine inspections of these systems that were performed by outside experts during plant operation continue Safe Storage on a reduced frequency.

9.2.1.5 Environmental Surveillance Program

An abbreviated version of the environmental monitoring program conducted during plant operation will be carried out during the continuing care period. The purpose of the program is to identify and quantify any 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 will be useful in evaluating lapses of control. For situations involving releases from events such as fire or malicious acts that may require prompt emergency actions to minimize public risk, special surveillance requirements would apply. This program is discussed in more detail in Appendix F.5.

9.2.1.6 Security

The protection of the public, principally against the consequences of their own actions, is an important dimension of the security program used for the continuing care period of Safe Storage. Conventional security detection and notification systems normally used to protect the utility against loss or damage are augmented by audible alarms. These alarms, strategically located outside secured radiation zones, loudly warn an intruder of his potential danger. Silent sensors simultaneously elicit an appropriate predetermined response from offsite security personnel.

Routine patrol checks by onsite guards are not considered to be cost-effective. By contracting for the services of a reputable private security agency, the facility owner is assured of adequate surveillance and prompt response to alarms without overloading the local law enforcement unit. Liaison with local law enforcement agencies is maintained and their assistance called for only when necessary.

Security is provided during the continuing care period by several methods. Locks on the gates in the fence around the decommissioned facility provide the first line of defense. The fence is maintained in good condition throughout the surveillance and maintenance period. Facility security is maintained at all times by installed intrusion alarms and high security locks on exterior doors. Intrusion, fire and radiation alarms are monitored continuously by a commercial security agency. Security agency personnel respond immediately or summon assistance as required, depending on the situation indicated by the alarms.

Physical security to prevent inadvertent radiation exposure of surveillance and maintenance personnel is provided by multiple locked barriers. The presence of these barriers makes it extremely difficult for an unauthorized person to gain access to areas where radiation or contamination is present. Discussion of the security plan for the continuing care period is given in more detail in Appendix H, Section H.4.

A representative, who is responsible for controlling authorized access into and movement within the facility, is designated by the utility. He is further charged with the responsibilities of appropriate actions and noticiations regarding breaches of security, upkeep of plant surveillance and maintenance programs, and administrative reporting of these events as required by state and federal regulations.

9.2.1.7 Quality Assurance Program

An extensive quality assurance (QA) program is carried on throughout the decommissioning effort to assure that the work does not endanger public safety, to assure the safety of the decommissioning staff, and to assure that all applicable QA and quality control (QC) regulations are met.

During the 18-month period prior to shutdown, QA personnel are active in the following areas:

- Review decommissioning plans for quality assurance involvement.
- Prepare inspection/test procedures as work plans are developed.
- Review designs of test equipment for quality input.

- Order inspection/test equipment required to perform the QC/QA function.
- Receive procured equipment, verify acceptance.
- Qualify suppliers for fabrication of radioactive shipping containers.
- Prepare inspection/test procedures to be imposed on subcontractors.
- Prepare inspection plans for shipment of radioactive materials, containers, truck, etc.
- Finalize formal Quality Assurance Plan.

The QA efforts during the actual decommissioning period include the following:

- Perform QA functions for procurements.
- Qualify suppliers
- Audit all program activities
- Monitor performance of Specialists, Utility Operators, Laborers, Craftsmen, and H. P. Technicians, for compliance with work procedures
- Verify compliance of radioactive shipments
- Perform dimensional, visual, nondestructive examination or other required inspection service to assure compliance with work plans
- Maintain auditable files on QA audits

Prepare final report on overall performance of the decommissioning program with regard to the QA function.

More details on the Quality Assurance Plan are given in Appendix F.4. A nominal level of effort, on an annual basis, consisting of audit functions and records checks is required during the continuing care period.

9.2.2 Work Schedule Estimates

A proposed overall schedule and sequence of events for the preparations for Safe Storage is presented in Figure 9.2-5. Initial work begins about 18 months prior to final reactor shutdown and includes the following activities:

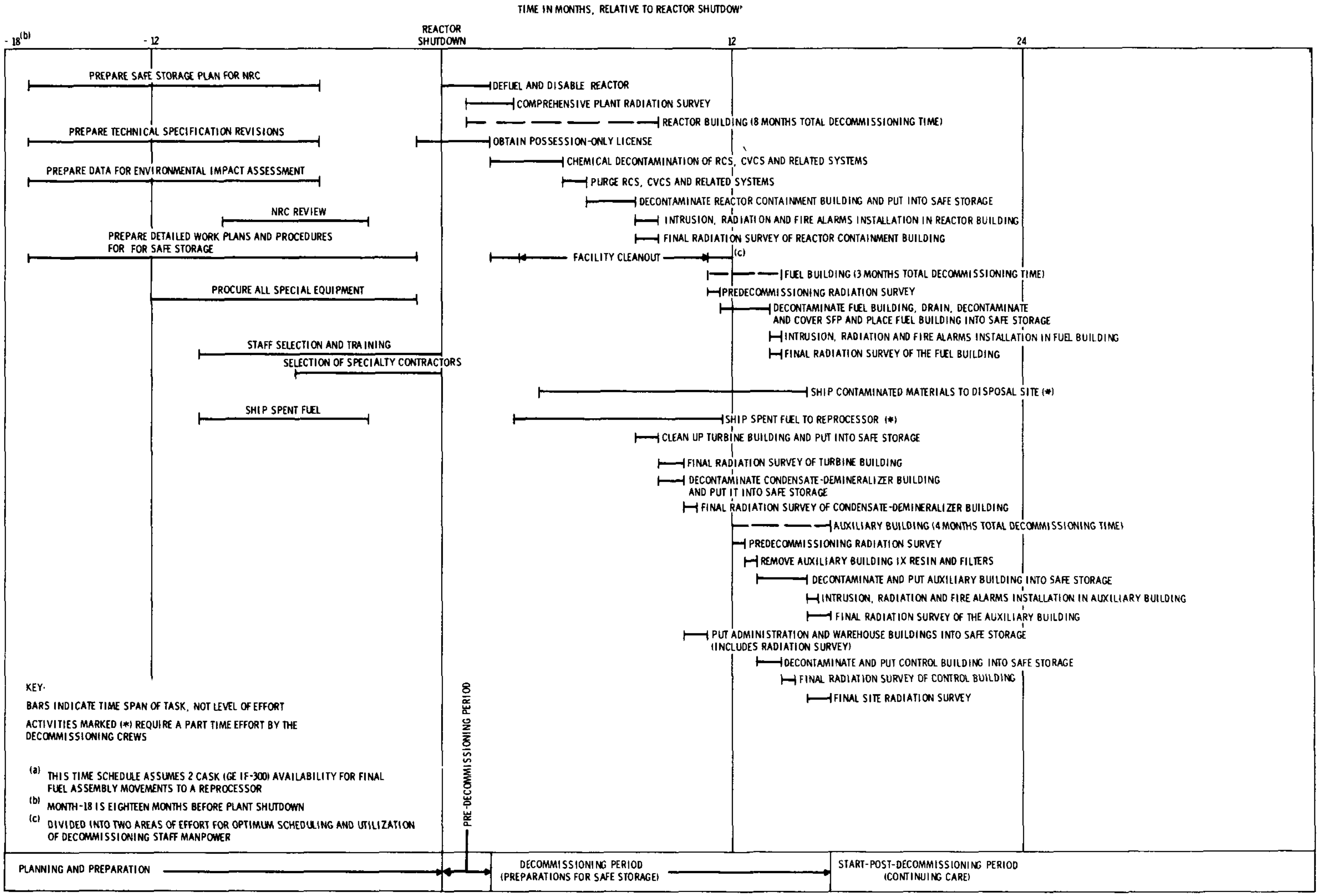

FIGURE 9.2-5. Sequence and Schedule of Preparations for Safe Storage[a]

- preparation of a decommissioning plan for NRC approval,
- preparation of the revisions to the Facility Technical Specifications necessary to change from an operating license to a possession-only license,
- preparation of the data needed to make an assessment of the environmental impact of the decommissioning activities,
- preparation of detailed work plans and procedures for placing the Facility in safe storage,
- design, procurement, and testing of any special equipment,
- selection of specialty contractors, and
- selection and training of personnel for the decommissioning staff.

Following shutdown, the reactor is defueled and disabled as required to obtain the possession-only license. The Reactor Coolant System, the Chemical Volume Control System, and related systems are chemically decontaminated. Decommissioning begins with a general facility cleanout and progresses through the various structures such as the Turbine Building, the Condensate Demineralizer Building, and the Reactor Containment Building. Then, supporting structures such as the warehouses, the Fuel Building, the Auxiliary Building, and the Control Building are decommissioned and prepared for the continuing care period. The draining of approximately 19,700 m^3 (5.2 million gal) of water from the Cooling Tower Reservoir and subsequent sealing of the reservoir progresses in parallel with the decontamination and decommissioning of the other structures. Decommissioning of all remaining nonessential systems, including purging all systems containing gases, completes the program.

The time required to ship the last core load of fuel to a reprocessor or storage site is estimated to be 8 1/2 months, assuming the availability of two General Electric Company IF-300 casks or their equivalent. Two casks, each capable of transporting seven PWR fuel assemblies by rail, are assumed to have a complete turnaround time of 18 days per shipment including two days for loading/unloading operations. Since fuel shipment is on the critical path for placing the facility in Safe Storage, the availability of more than two shipping casks could reduce the total time required to complete preparations for Safe Storage. It is further assumed that the fuel shipment operations include a final fuel inventory for accountability purposes.

As can be seen in Figure 9.2-5, about 16 months of effort is estimated to be required after reactor shutdown to complete preparations for Safe Storage.

9.2.3 Decommissioning Staff Requirements

The staffing requirements set forth in this section are for the decommissioning staff (those persons whose preparations for Safe Storage lead to the start of the continuing care period) and for the Safe Storage staff (those persons who provide the surveillance and maintenance services during the continuing care period). Those utility staff personnel who contribute to the planning and preparation during the final 18 months of plant operation as part of their normal work assignments are also included.

9.2.3.1 Staff For Preparations For Safe Storage

The staff is sized and structured on a two-shift, 5-day week. Certain operations such as chemical decontamination and security[a] are carried out on three shifts. A postulated organizational structure is shown in Figure 9.2-6. The personnel are drawn from the operations and maintenance staff at the facility, to the maximum extent possible, to capitalize on their detailed knowledge of and familiarity with the facility. Specialists and consultants are brought in as required to assist the dismantling staff.

The decommissioning staff organization for the reference PWR shown in Figure 9.2-6 consists basically of two parallel branches reporting to the Decommissioning Superintendent. The operational branch, under the Decommissioning Engineer, plans and carries out the actual decommissioning activities.

The safety branch, under the Health and Safety Supervisor, plans and conducts the industrial and radiological safety programs. The duties of key individuals on the decommissioning staff are given in Section 9.1.3 and are not repeated here.

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[a] At the start of the continuing care period, the guard force is disbanded and replaced by the private security agency forces and electronic intrusion detection systems as described in Appendix H.4.

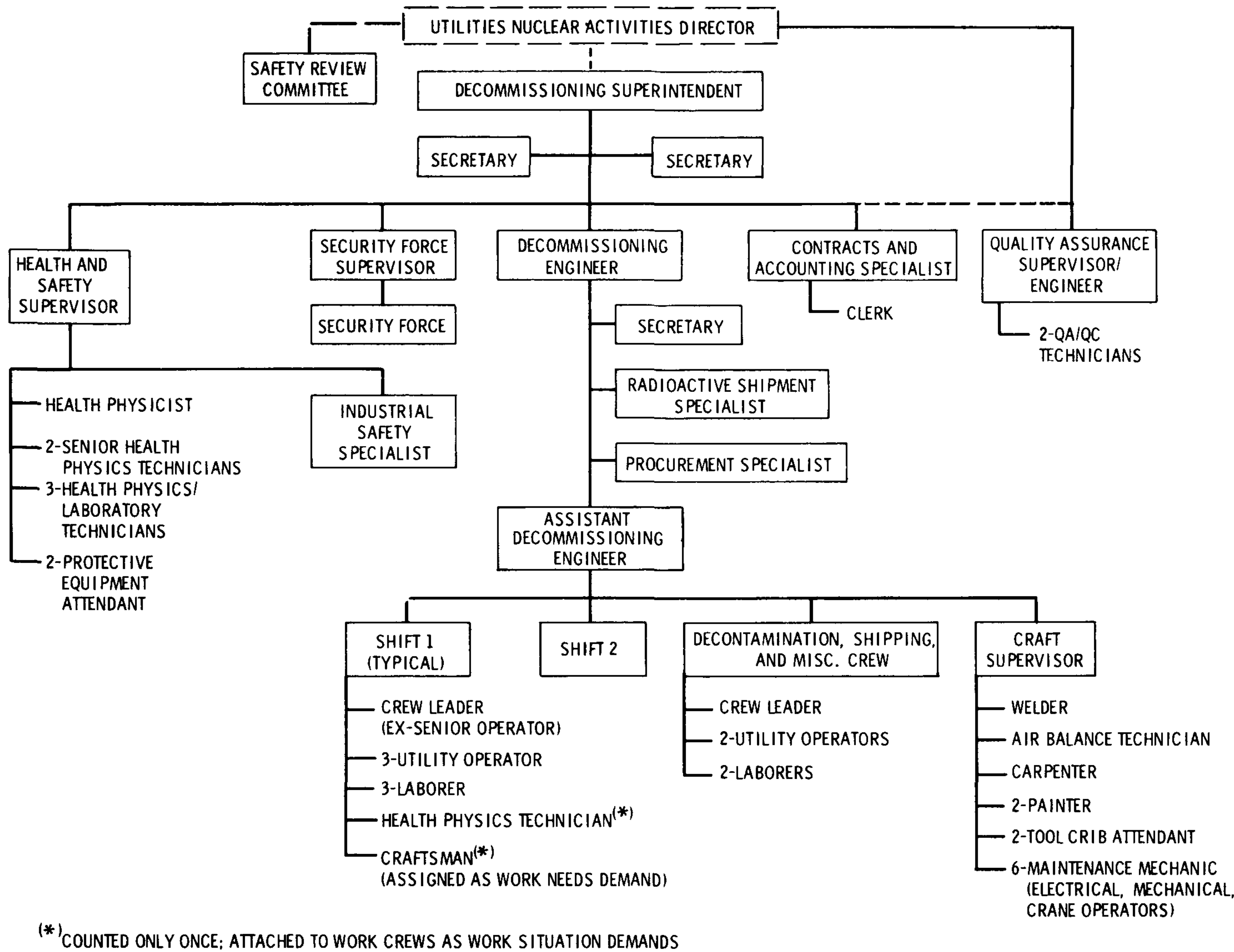


FIGURE 9.2-6. Postulated Decommissioning Staff Organization for Preparations for Safe Storage

The basic working unit is the crew, consisting of the Crew Leader (typically a Senior Operator), three Utility Operators, and three laborers, plus health physics technicians and craftsmen who are assigned to a crew as the work situation demands. There are three crews. All crews are supervised by the Assistant Decommissioning Engineer. Two crews handle the decommissioning of the facilities while the third crew's primary duties are decontamination, shipping functions, and assistance to the other two crews as needed. The projected utilization of manpower as a function of time is shown in Table 9.2-1. This summary is based on the detailed manpower assignments presented in Appendix H.1. The manpower shown in Table 9.2-1 exceeds the total manpower shown in Figures H.1-1 through H.1-4 in Appendix H. This additional manpower is utilized in planning and preparation, training activities, preparation of reports, and numerous, small unspecified work items.

It should be recognized that the completion of decommissioning activities occasionally takes longer than anticipated. Increased costs can often be offset by savings made from the rapid reduction of decommissioning personnel as soon as it is recognized that they can no longer be effectively utilized. The final cleanup items, for example, can be accomplished by a relatively small group. Savings in property tax, insurance and other expenses might also be realized by retiring buildings and equipment as soon as they are no longer needed.

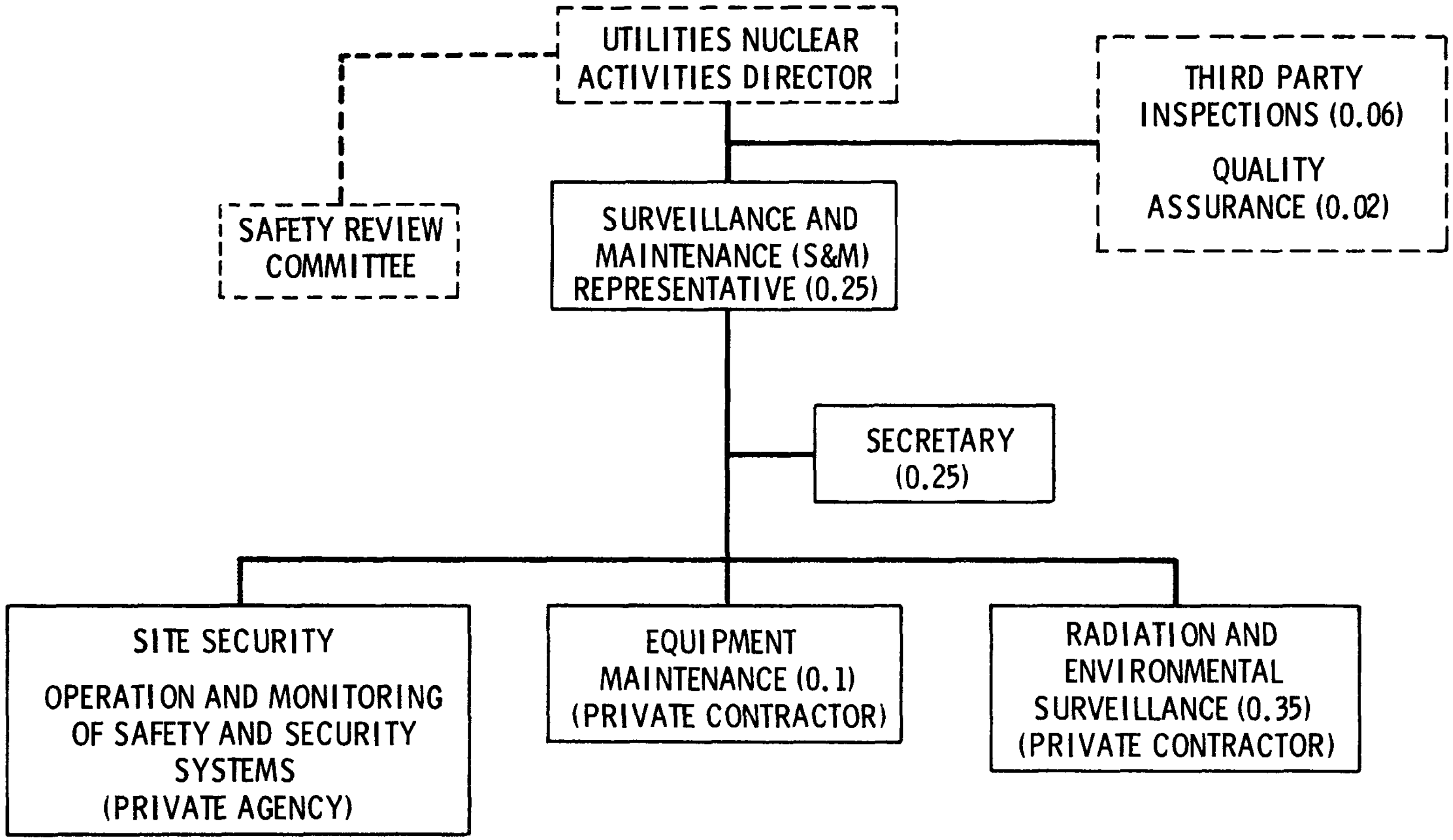
9.2.3.2 Staff For Safe Storage

The staff organization shown in Figure 9.2-7 takes over the surveillance maintenance, and security for the duration of the continuing care period. The surveillance and maintenance is supervised by one part-time employee known as the Surveillance and Maintenance (S&M) Representative. In addition to 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.

TABLE 9.2-1. Projected Utilization of Manpower During Preparations for Safe Storage

Title or Function	Time in Years Relative to Reactor Shutdown -2.0	-1.0	1.0	2.0	Total Man Years
Decommissioning Superintendent	0.5	1	1	0.33	2.83
Decommissioning Engineer	0.5	1	1	0.33	2.83
Asst. Decommissioning Engineer	0.5	1	1	0.25	2.75
Health and Safety Supervisor	0.25	1	1	0.33	2.58
Industrial Safety Specialist	0.25	1	1	0.33	2.58
Health Physicist		0.5	1	0.33	1.83
Sr. Health Physics Technician		1	2	0.66	3.66
Health Physics/Laboratory Technician		1.5	3	1	5.5
Protective Clothing Attendant			2	0.66	2.66
Quality Assurance Supervisor/Engineer	0.2	1	1	0.33	2.53
Quality Assurance Technician		0.5	2	0.33	2.83
Contracts and Accounting Specialist	0.2	1	1	0.33	2.53
Accounting Clerk		0.3	1	0.33	1.63
Security Supervisor			1	0.33	1.33
Security Patrolman			10	1.65	11.65
Radioactive Shipment Specialist		1	1	0.33	2.33
Procurement Specialist	0.2	1	1		2.2
Craft Supervisor			1	0.33	1.33
Craftsman			8	2.64	10.64
Tool Crib Attendant			2	0.66	2.66
Crew Leader		1.5	3	1	5.5
Utility Operator		4	10[a]	2.64	16.64[a]
Laborer		4	8	2.64	14.64
Secretary	0.5	2	3	1	6.5
Safety Review Committee Consultants	0.25	0.25	0.5	0.05	1.05
MAN YEARS/YEAR	3.35	23.55	66.5	18.81	113.21

(a) Defueling and disabling operations, occurring the first two months after the final reactor shutdown, requires the services of 20 utility operators (former reactor operators); this group is then reduced in number to a total of 8 for the remainder of the preparations for safe storage. The 12 extra utility operators required for the 2 months of defueling and disabling operations are equivalent to an additional 2 man years that are subsequently reflected in the Total Man Years column for the Utility Operators.



NOTE: EQUIVALENT MAN YEARS/YEAR INDICATED IN PARENTHESES ( ).

FIGURE 9.2-7. Postulated Staff Organization for Safe Storage (Continuing Care)

9.2.3.3 Contractors

Specialty contractors are employed as required to perform unique services during both the active decommissioning phase and the continuing care period. Table 9.2-2 shows the principal contractors for both periods.

The criteria for services are established during bid solicitation and remain as requirements during the job. Although the nature of the work dictates varying requirements, contractors should be required, as a minimum, to:

- scope the job ahead of time to enable adequate preparation
- Provide their own equipment, tools, etc.
- If working on a critical path item, be capable of completion of that item on schedule.

9.2.4 Essential Systems and Services

Certain of the facility systems and services must remain in place until radioactive and/or contaminated materials are either chemically decontaminated, fixed in place, or removed and packaged to assure that no radioactive or hazardous materials are released to the environs. Also, certain of these systems are needed to facilitate the cleanup and Safe Storage preparation efforts. These systems are identical to those required for Immediate Dismantlement as described in Section 9.1.4 and are not repeated here. As areas within the facility sections are readied for the continuing care period, the extensions of these services into those areas are deactivated and secured as required by Safe Storage procedures, while maintaining continuity of the service to the remaining work areas.

The support systems requiring surveillance and maintenance during the continuing care period are listed in Table 9.2-3. These systems remain in operation throughout the continuing care period. These systems, in combination with inherent facility structural integrity, provide the primary means for minimizing the release of hazardous material to the environment during the continuing care period. The equipment in these systems is inspected and renovated to assure adequate equipment reliability before the surveillance and maintenance period begins. Additionally, the intrusion alarm system within the facility as well as on the perimeter fence is modified to provide offshi surveillance capability by a commercial security agency.

TABLE 9.2-2. Postulated Specialty Contractors Utilized During Decommissioning by Safe Storage

Preparations for Safe Storage	Continuing Care Period
Temporary Waste Solidification Support, a transportable evaporator-solidifier system and operating personnel, to provide additional waste handling capacity and final cleanup capability after the installed waste handling systems have been deactivated, drained, and secured. This equipment is described further in Appendix F.3.	Environmental surveillance as described in Appendix F.5.2.
Hauling contractors, for transport of packaged radioactive waste materials from the facility to an authorized disposal site, Appendix I.	Security as described in Appendix H.
Decontamination specialists for fuel transfer, reactor cavity, and spent fuel pool decontamination utilizing portable high pressure spray units as described in Appendix F.3.	Quality assurance as described in Appendix F.4.

NOTE: On specific jobs that entail high radiation exposures, contracting out specific jobs has proven beneficial to completing the job in timely fashion and has reduced overall occupational radiation exposure through the use of trained specialists.

TABLE 9.2-3. Systems and Services Required During The Continuing Care Period of Safe Storage

Systems or Components	Justification
Electric power	Normal and emergency power are maintained for radiation monitoring systems and alarms, for lighting circuits, for fire protection systems and alarms, and for surveillance monitoring systems and alarms. Switchboards are aligned so that no electrical power is fed to deactivated systems.
Fire Protection System (detection and suppression)	Protable fire extinguishers remain at selected locations and fire detection systems remain in operation as required for safety.
In-plant Communication Systems (telephone)	Required for emergency communication (main gate only).
Radiation Monitoring Systems	Radiation monitors and alarms remain in operation at strategic locations throughout the facility sections. The locations of some devices may be installed to ensure that important areas are adequately covered. Selected monitoring programs are also continued.
HVAC Systems	The ventilation systems remain intact; however, the ventilation systems are drained, decontaminated, deactivated, and secured. Unrequired valves and dampers are closed as each system is secured. The system usage is limited to providing pathways via existing ductwork and through HEPA filters to relieve any pressure differentials caused primarily by temperature changes between contaminated/radiation areas and the outside world. See description in Figure 9.2-4.
Security Systems	Onsite security devices and alarms (provided with both normal and backup emergency power) are maintained by the security agency subcontractor described in Section 9.2. In addition to intrusion system monitoring and maintenance, it is postulated that the security agency also monitors and responds appropriately to the Radiation Monitoring System and Fire Protection System alarms described above.

The principal cost item listed in Table 9.2-3 is electrical power. The operating plant load is about 45 MW and the shutdown plant load is about 22 MW. By maximum load reduction efforts, it is estimated that shutdown load can be reduced to about 11 MW. Use of the RCS pumps during decontamination would add about 18 MW to that base load while the pumps are running. A 0.057 m^3/min (15 gal/min) evaporator contributes approximately 2-1/2 MW of the base load, and a minimum water supply from the river requires about 1-1/2 MW for pumping. The estimated electricity usage during the 16 months required to place the facility in Safe Storage is about 124,000 MWh. Details on electrical costs are presented in Section 10.2.

9.3 DEFERRED DISMANTLEMENT

Deferred Dismantlement, as defined in Section 4, is the final stage of decommissioning when Safe Storage is utilized. The facility and the site must be shown to have residual radioactivity levels sufficiently low to permit unrestricted use, as defined in Section 8, when decommissioning is complete.

The same basic operations are performed during Deferred Dismantlement as were performed during Immediate Dismantlement. The internal components of the reactor vessel will have sufficiently high radiation dose rates to require disassembly and sectioning underwater, even after a 100-year decay period, due to the presence of ^{94}Nb. Thus, the same semi-remote cutting techniques would be employed. Similarly, portions of the reactor vessel may be sufficiently radioactive to require sectioning using semi-remote equipment, especially for decay periods of 50 years or less. Portions of the concrete in the biological shield will remain radioactive for long periods of time, due to the presence of activated trace elements such as ^{152}Eu, and ^{154}Eu, and will have to be removed for packaging and burial. The radioactive corrosion products on the inner surfaces of the piping, tanks, etc., consist mostly of ^{60}Co. Even though these systems were chemically decontaminated during the preparations for Safe Storage, it is unlikely that the residual radioactivity will have decayed to levels that permit unrestricted use before 50 years have elapsed. All of the systems will have to be disassembled in order to make measurements on the interior surfaces of the systems to determine whether the material can be released or must be buried, regardless of the length of the Safe Storage period.

Operations such as reactor defueling and shipment of spent fuel, chemical decontamination of the fluid systems, and removal of radioactive wastes such as cartridge filters, ion exchange resins and evaporator bottoms liquids were performed during Immediate Dismantlement and are not required during Deferred Dismantlement. These activities are replaced by extensive training and familiarization of the decommissioning staff with the facility, since the staff cannot be made up of personnel from the operations staff after an extended period of Safe Storage. Additional effort will be required to restore the services needed for dismantlement throughout the station, and to remove the various locks, welded closures, and barricades that were installed to secure the station during the preparations for Safe Storage.

In view of the considerations given above, it is reasonable to assume that a work force of the same size as was utilized in Immediate Dismantlement will be required for Deferred Dismantlement, and over approximately the same period of time. Other assumptions made in this study with regard to Deferred Dismantlement are:

- If dismantlement is performed sooner than 50 years after reactor shutdown, all of the systems and materials are assumed to be still too radioactive to be released for unrestricted use. The same volumes of material must be removed and transported to a burial site.
- After 50 years of Safe Storage, the only contamination remaining in the facility is assumed to be the accumulation of fission products on the surfaces of isolated shielded cells (ion exchange vaults). The amount of contaminated material for disposal is reduced to 150 m^3 or less. The activated corrosion products in the piping systems and on the nonactivated components have decayed sufficiently to permit unrestricted use of those materials.

9.3.1 Work Schedule Estimates

As discussed in the previous sections, since the same basic efforts are required to dismantle a plant regardless of when the dismantlement takes place, the work schedules presented in Figure 9.1-1 are assumed to be valid. Items such as a reactor defueling, fuel shipment, and chemical decontamination are replaced by familiarization and orientation of the work force with the facility, by training, restoring essential services and unsecuring the facility.

9.3.2 Staff For Deferred Dismantlement

Based on the discussion and assumptions given in Section 9.3, the total manpower required during Deferred Dismantlement is the same as for Immediate Dismantlement. The time formerly devoted to chemical decontamiantion, reactor defueling and shipment of spent fuel, packaging and shipment of filters, resins, evaporator bottoms liquids, and combustible wastes is spent in familiarization, orientation and training, and unsecuring the facility and restoring essential services throughout the facility. Thus, the basic staff structure outlines in Section 9.1.3 is again employed. Similiarly, the same types of Specialty Contractors are employed.

One possibility that was not considered in this study was the formation of companies specializing in decommissioning of nuclear facilities, especially for facilities that have been held in Safe Storage for some time. In this way, personnel experienced in decommissioning operations could be assembled, rather than having to orient and train a staff that is initially unfamiliar with decommissioning work. It is anticipated that use of such specialized organizations could result in reducing the costs of deferred dismantlement significantly, but no attempt was made in this study to evaluate the magnitude of these potential cost reductions.

9.4 PARTIAL DISMANTLEMENT WITH SAFE STORAGE

A variation of Safe Storage that is of potential interest is one in which the structures and surfaces in the area outside of the Exclusion Area but within the site perimeter fence are surveyed and decontaminated as required to permit unrestricted use. Demolition of the structures could be accomplished if desired, or the structures could be utilized for non-nuclear purposes. In any event, the outer region of the site, about 4.6 km^2, could be released for productive use while the Exclusion Area was held in Safe Storage. Thus, nearly 98% of the land area at the site could be made available for other uses. Removal of some structures such as the Intake Structure for pumping water from the river could require a temporary intake installation for use during deferred dismantlement if other sources were not readily available.

Decontamination of the outer area of the site is not anticipated to be a significant effort. Radiation surveys and analysis of samples from the soil surface and from paved areas would be required to demonstrate that the site is releasable. Removal of a few square meters of soil surface to a depth of a few centimeters and removal of sections of paving material from areas that were found to be "hot spots" might be required, thus increasing radioactive waste disposal costs slightly.

If demolition of the structures in the outer area were selected, these costs would be incurred during the preparations for Safe Storage rather than during Deferred Dismantlement. In terms of constant dollars, the total decommissioning costs would be essentially the same, since the costs associated with decontamination of the outer area are insignificant in comparison with the cost for preparation for Safe Storage.

REFERENCES

1. U.S. Nuclear Regulatory Commission, Termination of Operating License for Reactors. Regulatory Guide 1.86, p. 1.86-2, June 1974.

10.0 DECOMMISSIONING COSTS

The cost estimates for accomplishing the decommissioning of the reference PWR by Immediate Dismantlement and by Safe Storage with Deferred Dismantlement are developed in detail in Appendices G and H and are summarized in this Section.

10.1 COST ESTIMATES FOR IMMEDIATE DISMANTLEMENT

The principal assumptions made in the generation of cost estimates for the dismantlement of a PWR are the following:

- The staff is drawn from the Technical and Operations staffs of the electric utility. Thus, all support services and part-time assistance of many staff members can be utilized during the planning and preparation period with only nominal costs to the dismantlement program.
- The possession-only license is in place by the time the reactor fuel has been unloaded, thus permitting decontamination and dismantlement activities to begin promptly.
- Chemical decontamination of the various systems is sufficiently successful to permit the decommissioning staff to work in direct contact with the systems in all areas except the highly-activated reactor vessel, vessel internals and biological shield.
- All pumps, piping, pool liners, tanks, heat exchangers, etc., associated with the Containment, Fuel, and Auxiliary buildings are assumed to be contaminated and must be packaged and shipped to a burial site.

The detailed cost estimates for the activities needed to dismantle a PWR are developed in Appendix G, with the results summarized in Table 10.1-1. The costs have been collected into groupings that facilitate their use in consideration of other possible decommissioning modes. The costs related to removal and disposal of the reactor vessel, internals, and biological shield are grouped as Activated Materials (8.1% of the total decommissioning costs). The steam generators, pressurizer, associated tanks, heat exchangers, piping and all other contaminated materials within the reactor containment are grouped as Containment Internals (2.9% of the total cost). The tanks, piping, heat exchangers, pool liners, and all other contaminated materials within the Fuel, Auxiliary, and Control buildings are grouped as Other Building Internals (12.5 % of the total cost).

TABLE 10.1-1. Summary of Estimated Dismantlement Costs for the Reference PWR Facility

Category	Cost in Millions[c] of 1978 Dollars	Percent of Total
Spent Fuel Disposal	2.467	7.3
Activated Materials Disposal	2.734[a]	
Containment Internals Disposal	0.961	
Other Building Internals Disposal	4.222	25.6
Waste Disposal	0.693	
Staff Labor	8.986	26.7
Electrical Power	3.500	10.4
Special Equipment	0.822[b]	2.4
Miscellaneous Supplies	1.559	4.6
Facility Demolition (non-radioactive)	6.410	19.0
Specialty Contractors	0.390	1.2
Nuclear Insurance	0.800	2.4
Environmental Surveillance	0.154	0.5
SUBTOTAL	33.698	
25% Contingency	8.425	
TOTAL DISMANTLING COSTS (ROUNDED)	42.1	

(a) Cost differential for deep geologic disposal of highly activated reactor vessel components = $1.8 million. (See Appendix G.4.2.1)

(b) See Table F.3-1, Appendix F, or Table I.6-1, Appendix I.

(c) Number of figures shown is for computational accuracy and does not imply precision to the nearest thousand dollars.

Disposal of liquid wastes, ion exchanger resins, filter cartridges, and miscellaneous dry wastes are grouped as Waste Disposal (2% of the total cost). The liquid wastes are reduced in volume by evaporation as much as possible, compatible with the characteristics of the solidification process.

All staff labor costs (26.7% of the total cost) are given as Staff Labor, except for the labor costs of the Demolition Contractor and the Specialty Contractors which are included in their respective totals.

Costs for shipment of spent reactor fuel to an unspecified off-site spent fuel repository (7.3% of the total cost) are listed explicitly. Handling or processing costs at the repository are not included.

Development, procurement, and testing of the equipment items described in Appendix F.3 are grouped as Special Equipment (2.4% of the total cost). Significant development costs are anticipated for the underwater manipulator system and for the cutting devices to be used with it, the plasma torch or the arc saw.

Items such as disposable protective clothing, decontamination chemicals, assorted cleaning agents, rags, mops, plastic bags and sheeting, glass-fiber and HEPA filters, ion exchange resins and fluid filter cartridges, expendable tools, etc. are grouped together as Miscellaneous Supplies (4.6% of the total cost).

The costs associated with demolition of the decontaminated and dismantled structures, including special equipment, explosive specialists, rubble removal and backfilling to grade levels are listed as Facility Demolition (19% of the total cost). Credits for salvage of noncontaminated materials are included in arriving at the net demolition cost. Demolition of the decontaminated structures is not an NRC requirement for decommissioning but is included here to give a complete picture of the potential total cost should the station owner choose to completely clear the site.

Special services such as a mobile radwaste treatment system, explosives specialists working on dismantling and decontamination, and the final landscaping of the site are grouped together as Specialty Contractors (1.2% of the total cost).

A significant savings can be made if the electropolishing decontamination system is successful in cleaning stainless steel components to unrestricted use levels, as defined in Section 8, permitting salvage and sale of essentially all nonactivated stainless steel. Over 0.91 million kilograms ($\sim$2 million pounds) of potentially salvageable stainless steel is removed from the facility during dismantling, which at $\sim$60¢/kg, is worth about $544,000. Since the material is salvaged, it does not need to be packaged and shipped to a disposal site for burial, resulting in an additional savings of about $670,000. Thus, the net value of decontaminating the stainless steel components to unrestricted use levels is about $1.2 million, and a valuable nonrenewable resource is returned to the commercial stream.

Additional savings of over a quarter million dollars could be made by decontaminating to unrestricted use levels all of the carbon steel piping from the facility, thus eliminating the packaging, shipping, and burial costs.

10.1.1 Estimated Dismantlement Staff Costs

The utilization of manpower together with the accumulated man-years and staff labor costs are given in Table 10.1-2. This summary is derived from the detailed manpower per job item schedules developed in Appendix G, and the listing of salary data for typical staff positions given in Table I.1-1, Appendix I. From Table 10.1-2, it can be seen that a total of about 300 man-years is estimated to be required to decontaminate and to remove activated and contaminated materials from the facility, at a labor cost of nearly $9 million. Not included are costs for specialty contractors, which are discussed in the next section.

10.1.2 Estimated Costs for Specialty Contractors

A variety of specialized services are required to accomplish the decommissioning of the reference PWR. These services are supplied by the specialty contractors listed below.

- Temporary Waste Solidification System - a transportable evaporator-solidification system with operating personnel, to provide additional liquid radioactive waste handling capability and final cleanup capability after the installed facility waste handling systems are removed. The estimated cost for such a system is $10,000 per month, for an estimated total of $280,000 over the decommissioning period.

- Explosives specialists, for removal of the activated portions of the biological shield, and for cutting of selected piping, etc., using shaped charges, for an estimated total of $110,000 during the decommissioning period. This estimate does not include explosives work during the demolition of the decontaminated structures.

- Hauling contractors, for transport of packaged radioactive materials from the facility to a licensed burial site. These costs are included in the estimates for Material Disposal. The estimated total costs for truck transport of radioactive materials from the facility to

TABLE 10.1-2. Projected Manpower Utilization and Staff Costs during Immediate Dismantlement

Title or Function	Time in Years Relative To Reactor Shutdown -2	-1	1	2	3	4	Total Man Years	Total Cost ($ millions)
Decommissioning Superintendent	0.3	1	1	1	1	1	5.3	0.3948
Decommissioning Engineer	1	1	1	1	1		5	0.3175
Asst. Decommissioning Engineer	1	1	1	1	0.5		4.5	0.1971
Health & Safety Supervisor	0.3	1	1	1	1		4.3	0.2167
Industrial Safety Specialist	0.3	1	1	1	1		4.3	0.1883
Health Physicist		0.5	1	1	1		3.5	0.1379
Senior Health Physics Technician		1	2	2	1		6	0.1740
Health Physics Technician		3	7	7	5		22	0.5522
Protective Equipment Attendant			2	2	1.5		5.5	0.1276
Quality Assurance Supervisor	0.3	1	1	1	1	1	5.3	0.2321
Quality Assurance Engineer	0.5	2	1	1	0.6		5.1	0.2009
Quality Assurance Technicians		0.5	4	4	3	1	12.5	0.2900
Contract and Accounting Specialist	0.3	1	1	1	1	1	5.3	0.2035
Accounting Clerk		0.3	1	1	1		3.3	0.0670
Security Supervisor			1	1	1	1	4	0.1316
Security Patrolman			15	10	8	8	41	0.8733
Radioactive Shipment Specialist		1	1	1	1		4	0.1316
Procurement Specialist	0.3	1	1	1			3.3	0.1013
Craft Supervisor			2	2	2		6	0.2364
Craftsman			16	16	7		39	1.0569
Tool Crib Attendant			2	2	1.5		5.5	0.1276
Shift Engineer			2	2	2		6	0.2364
Crew Leader			7	7	5		19	0.6251
Utility Operator			10	7	3.4		20.4	0.5528
Laborer			18	18	11.2		47.2	1.2225
Secretary	1	2	3	3	2	1	12	0.2436
Safety Review Committee Consultants	0.5	0.5	0.5	0.5	0.2		2.2	0.1474
Man Years/Year	5.8	18.8	103.5	95.5	63.9	14		
Cost (millions of dollars)	0.2653	0.7164	2.9456	2.7578	1.8975	0.4035		8.9861
Total Man Years	------	------	--------	-------	-------	-------	301.5	

NOTE: The number of significant figures shown is for computational accuracy and does not imply precision to the nearest one hundred dollars.

a licensed burial site are $2.518 million. This estimate does not include shipment of spent fuel, which is accomplished by train with an estimated cost of $702,800.

- Demolition contractor, for demolition of the decontaminated structures, disposal of excess clean rubble, back-filling of below-grade areas to grade level, and final grading and restoration of the site. The estimated costs for the demolition of the decontaminated and noncontaminated structures are summarized in Table 10.1-3. The value of noncontaminated salvage material from the structures is reflected in lowered demolition costs, since the demolition estimates take the potential value of salvageable materials into account. No separate estimates of salvage value are made. The amount of demolition worker labor needed to accomplish the demolition and site restoration is estimated to be approximately 82 man-years. The estimated demolition costs, summarized in Table 10.1-3 in 1978 dollars, include all job costs, supplies, overhead and profit. Any extraordinary costs for bonding, insurance, or state sales taxes are not included.

TABLE 10.1-3. Summary of Estimated Demolition Costs for the Reference PWR[a,b]

Cooling Tower	$2,468,000
Reactor Containment Vessel	1,327,000
Auxiliary Building	665,800
Fuel Building	378,200
Control Building	427,400
Turbine Building	869,900
Turbine Auxiliary Building	107,400
Condensate Demineralizer Building	28,400
Administration Building	18,700
Shop and Warehouse	2,200
Chlorine Building	6,400
Intake Structure (no quantity estimate)	
Barge Loading Facility (no quantity estimate)	
Intake Piping (no quantity estimate)	
Contingency allowance for No Quantity Items	55,600
Area Landfilling (excludes roadways)	55,600
Estimated Total Cost to Remove Structures (rounded)	$6,410,000

(a) J. M. McFarland, Report on Representative Cost Estimates for Demolition of Structures at a Pressurized Water Reactor Site, PNL-2450, prepared for Battelle Pacific Northwest Laboratories by McFarland Wrecking Corporation, Seattle, WA 98108, September 30, 1976.

(b) Estimates in (a) have been increased by 11.3% to account for inflation and pay increases between September 1976 and January 1978, based on data supplied by the Seattle area chapter of the Association of General Contractors.

10.1.3 Estimated Costs for Disposal of Radioactive Materials

The radioactive materials that must be removed from the facility during dismantlement are grouped into five categories: spent fuel, activated material, contaminated material from the containment building, contaminated material from all other buildings, and radioactive waste (resins, filters, evaporator bottoms, combustibles). The estimated costs for disposing of these materials are summarized in Table 10.1-4, based on detailed analyses given in Appendices G and I. No final disposal costs are assigned for the spent fuel assemblies since the ultimate disposition of spent fuel is as yet undefined.

TABLE 10.1-4. Estimated Costs for Disposal of Radioactive Material

	Millions of Dollars				
Category	Container[a]	Transportation[b]	Burial[c]	Total	Burial Volume (m^3)
Spent Fuel	None	2.467[d]	Disposition Undefined	2.467	37
Activated Material	1.740	0.485	0.509	2.734[e]	1,191
Containment Internals	0.370	0.311	0.280	0.961	2,568
Other Building Internals	1.488	1.470	1.264	4.222	13,510
Waste	0.137	0.425	0.131	0.693	618
TOTALS	3.735	5.158	2.184	11.077	17,924

(a) Container costs as given in Table I.2-1, Appendix I.
(b) Includes cask rental for 5 days/shipment, plus trucking costs (except for spent fuel), as given in Tables I.3-1 and I.4-4, Appendix I.
(c) Handling, burial, and surchanges as given in Table I.5-1, Appendix I.
(d) Includes rental on GE IF-300 cask, and rail transport costs as given in Appendix I.4.2.
(e) Cost differential for deep geologic disposal of highly activated material is +$1.8 million.

NOTE: The total of truck shipments is 1363.

10.1.4 Nuclear Insurance Costs

The premium costs for nuclear insurance during Immediate Dismantlement have been estimated.[1] Based on an assumed policy limit of $125 million carried through the decommissioning period, the annual premiums were estimated to be in the ranges shown in Table 10.1-5.

TABLE 10.1-5. Estimated Nuclear Liability Insurance Premium During Immediate Dismantlement

Year After Shutdown	Range of Estimated Annual Premium ($)
1	200,000 - 234,000
2	180,000 - 210,000
3	180,000 - 210,000
4	120,000 - 140,000

Based on these estimates, the total cost for nuclear liability insurance during immediate dismantlement has been postulated to be $800,000.

10.1.5 Estimated Costs for Miscellaneous Supplies

Many expendable supplies are used during the decommissioning work. These include decontamination chemicals, ion exchange resins, glass-fiber and HEPA filters, cartridge-type fluid filters, disposable protective clothing, assorted cleaning agents, rags, mops, plastic bags and sheeting, and expendable tools. The estimated costs for the major items are given in Table 10.1-6.

10.1.6 Estimated Environmental Surveillance Costs

The costs for the program of environmental surveillance during the dismantlement period (described in Appendix F.5) are developed in Appendix G.4 and summarized here. The program will continue at the same level until shipment of radioactive materials from the site is completed, in approximately four years, for a total cost of $154,000.

[1] Personal Communication: Letter from C. R. Bardes, Nuclear Energy Liability Property Insurance Association (NEL-PIA), to R. I. Smith, Battelle-Northwest, November 18, 1976.

TABLE 10.1-6. Estimated Costs for Miscellaneous Supplies

Item	Quantity	Total Cost($)
Decontamination chemicals: (EDTA/Oxalic Acid/Citric Acid)	3,600 to 18,100 kg[a]	23,000 to 115,000
Ion Exchange Resins	30 m^3	96,000
Filters	Unspecified	380,000
Protective Clothing	Unspecified	578,000
Cleaning Supplies	Unspecified	290,000
Expendable Tools	Unspecified	100,000
		1,467,000 to 1,559,000

[a] One coolant system volume of 1 to 5% solution by weight.

10.1.7 Estimated Costs for Special Tools and Equipment

The special tools and equipment anticipated to be utilized during Immediate Dismantlement are listed in Table 10.1-7, together with the estimated cost of each item and the probable use of the item during dismantlement.

10.1.8 Electrical Costs

Electricity is the primary cost item associated with providing essential systems and services (described in Section 9.1.4) that must remain in place until all radioactive and/or contaminated material have been removed from the site during Immediate Dismantlement. As areas within the facility are readied for demolition, the extensions of electrical services into those areas are deactivated and removed, while maintaining continuity of electrical services to the remaining work areas.

The operating plant load is about 45 MW, and the cold shutdown plant load is about 22 MW. By maximum load reduction efforts, the cold shutdown load can probably be reduced initially to about 11 MW. Use of the RCS pumps during chemical decontamination would add about 18 MW to that base load while the pumps are running. A 0.057 m^3/min (15 gpm) evaporator contributes about 2 1/2 MW of the base load, and a minimum water supply from the river requires about 1 1/2 MW. Making reasonable assumptions about how the electrical load is distributed across

TABLE 10.1-7. Special Tools and Equipment for Dismantlement

Item	No. Required	Estimated Unit Cost ($000)	Functions
Underwater manipulator[a]	1	300	Movement and positioning of underwater cutting devices used in sectioning and reactor pressure vessel and vessel internals.
Underwater plasma cutting torch[a] Underwater oxyacetylene torch Portable plasma cutting torch Arc saw[b]	2 2 4 1	20 5 20 100	Used for sectioning vessel and internals. (Ability to cut 230 mm, SS clad steel vessel needs to demonstrated.) Used for cutting stainless steel piping, fuel racks, etc. Used for general cutting of piping, support structures, etc. Used for sectioning vessel internals, possible reactor vessel steam generators, tanks and pool liners.
Guillotine pipe saws[c]	9	4	Portable power pipe saws for removal of piping
Closed circuits, high resolution television systems, underwater and in air	2	(plant equip.)	For observation and control of remote or underwater operations in high radiation fields.
Shielded vehicle with manipulators[d] Exchangeable tools, scoop loader, etc.	1	50	For remote handling of highly radioactive materials, packaging of activated concrete rubble, etc.
Underwater lights and viewing windows	as req.	5 total	For illuminating and observing underwater operations.
Submersible pumps with disposable filter cartridges	5	1.5	For rapid cleanup of pool water during underwater cutting operations, collection of chips and particles.
Assorted underwater tools such as impact wrenches, grapples, bolt cutters, etc.	as req.	25 total	For underwater disassembly, handling and packaging of highly activated materials.
High velocity water jets[e]	—	—	For surface decontamination of piping, tanks and equipment.
Hydraulic concrete surface spalling device	4	5	For removal of contaminated concrete surfaces. See subsequent description in Section F.3.3.
Concrete drills Electric/pneumatic hammers	4	0.5	For drilling holes in concrete as required for surface spalling and volume blasting.
Portable filtered ventilation enclosures	4	1.5	For collecting and filtering air, smoke, fumes, etc., from cutting operations on contaminated materials. See subsequent description in Section F.3.5.
Supplied airbubble suit	200	0.05	Provide personnel with maximum respiratory and surface protection against contamination.
Safety nets[f]	5	25 total	For use under personnel working on elevated systems and structures.
Blasting mats	10	0.5	For use over areas being blasted for removal of concrete, prevention of flying missiles, etc.
Shipping casks and liners	as req.	0.1/day	For transport and burial of radioactive materials at a disposal site. See Section F.2 for details.
Electropolishing decontamination	1	100	For final decontamination of valuable materials. See subsequent description in Section F.3.6.
Mobile radwaste processing unit[g]	1	10/month	For additional radwaste processing capacity during dismantling, and use after inplant radwaste system has been dismantled.
Primary piping jumper	—	—	For bypassing the reactor vessel volume during chemical decontamination, if a multiple cycle decontamination program is followed, to reduce the volume of solutions and rinse water to be processed. See subsequent description in Section F.3.7.

(a)See COO-651-93 Final Program Report on dismantling of Elk River Reactor, September 1974.
(b)Schlienger, M. P. (Schlienger, Inc., San Rafael, California) and G. A. Beitel (Atlantic Richfield Hanford Co., Richland, Washington), Topical Report: ARC SAW TESTING, February 18, 1976.
(c)Brochures from E. H. Wachs, Co., Wheeling, Illinois.
(d)Bainbridge, G. R. Decommission of Nuclear Facilities, A Review of Status, IAEA, October 1973.
(e)Carver, J. Barton, New Uses for Cleaning with Water, POWER, May 1976.
(f)Brochures from SINCO, Safety and Industrial Net Division, East Hampton, Connecticut.
(g)Nucleonics Week, June 24, 1976.

the various buildings at the site, and following the schedule of dismantlement given in Section 9.1.2, the power usage by year after shutdown is estimated to be: 1) 96,400 MW-hr, 2) 93,600 MW-hr, 3) 42,500 MW-hr, and 4) 1000 MW-hr, for a total of 233,500 MW-hr during the dismantlement operations. At a cost of 15 mils per kWh, the power costs are: 1) $1,446,000, 2) $1,404,000, 3) $637,500, and 4) $15,000, for a total of $3.5 million over the dismantlement period.

10.2 COST ESTIMATES FOR PREPARATIONS FOR SAFE STORAGE

The principal assumptions made in the generation of cost estimates for the Safe Storage decommissioning activities are:

- Most of the decommissioning staff is drawn from the Technical and Operations staffs of the electric utility. Thus, many support services and much part-time assistance of these onboard staff members are utilized during the Planning and Preparation period.
- The possession-only license is in place by the time the reactor fuel has been unloaded, thus permitting initial decontamination and decommissioning activities to begin promptly.
- The RCS and CVC Systems are assumed to be drained, dried, and isolated after chemical decontamination as described in Appendix F.1.1.
- Chemical decontamination of the various systems is sufficiently successful to permit the remaining decommissioning work to be accomplished by direct contact of the staff.
- Specialty contractors are assumed to expeditiously handle critical path items such as decommissioning of the Reactor and Refueling Cavity and the SFP and associated isolation cover, hauling radioactive wastes, and supplemental waste solidification support activities, etc.

The costs to place the reference pressurized water reactor facility in Safe Storage immediately after plant shutdown are estimated to be approximately 14 million dollars. The annual cost of continuing surveillance and maintenance at the facility for the Safe Storage (Continuing Care) Period is estimated to be about $80,000. The detailed cost estimates for the preparations for Safe Storage of a PWR are developed in Appendix H, with the results summarized in Table 10.2-1. Manpower cost estimates are given in Section 10.2.1. Special tools and equipment costs are derived from items described in Section 10.2.8.

10.2.1 Estimated Staff Labor Costs for the Preparations for the Safe Storage

The bases for the estimates of the staff labor costs are given in Appendix I.1, where salary data for typical staff positions are presented.

TABLE 10.2-1. Time Distribution of Expenditures for Preparations for Safe Storage

	Year of Expenditure Relative to Start of Decommissioning (Thousands of Dollars)				
Cost Item	-2	-1[a]	1	2	Total[b]
Staff Labor	163 9	844 4	2,033 6	599 4	3,641 3
• QA Travel		2	3		5
• QA Materials and Equipment		2	3		5
Specialty Contractors					
• Waste Solidification System			50	10	60
• Decontamination Specialists			54		54
• Environmental Surveillance			35 6	8 9	44 5
• Reactor and Refueling Cavity Decontamination			8 4		8 4
• SFP Decontamination and Covering				52 6	52 6
Disposal of Radioactive Materials					
• Spent Fuel, Transportation Only[e]			2,467		2,467
• Radioactive Waste[c,d]			363	181	544
Nuclear Insurance Costs			234	60	294
Miscellaneous Supplies[e]		669	148 5	74	891 5
Special Tools and Equipment		74 5			74 5
Utilities			1,655	210	1,865
Security[f]				85 8	85 8
SUB-TOTAL	163 9	1,591 9	7,055 1	1,281 7	10,092 6
25% Contingency					2,523 1
TOTAL					12,616

(a) Year -1 is one year before plant shutdown

(b) Number of significant figures shown is for computational accuracy and does not imply precision to the nearest hundred dollars

(c) Refer to Table 10 2-3 for details

(d) Refer to Table H 3-2 of Volume 2 for more details

(e) Derived from Table 10 2-5

(f) Refer to Section 10 3 for estimated security costs for Safe Storage (Continuing Care) and Appendix H 4 2 of Volume 2 for itemized security costs breakdown

The projected utilization of manpower as a function of time, the accumulative man-years for each staff labor position, and the total cost of decommissioning staff labor is presented in Table 10.2-2.

Staff labor represents about 33% of the cost for preparations for Safe Storage. Delays in the postulated Safe Storage schedule would tend to increase the costs and schedule acceleration would tend to decrease the costs. The cost of utilities is an important factor in determining total costs.

10.2.2 Estimated Costs for Specialty Contractors

A variety of specialized services are required to accomplish the preparations for Safe Storage of the reference PWR. These services are supplied by the specialty contractors presented in Section 9.2.3.3, with a summary of each specialty contractor and estimated costs listed below.

- Temporary Waste Solidification System - a transportable evaporator-solidification system with operating personnel, to provide additional liquid radioactive waste handling capability and final cleanup capability after the installed facility waste handling systems are deactivated. The estimated cost for such a system is $10,000 per month, for an estimated total of $60,000 over the decommissioning period.

- Decontamination specialists for decontamination of the fuel transfer system and the reactor cavity, and the spent fuel pool (SFP), utilizing portable high pressure spray units as described in Appendix F.3. These estimated costs total about $8,400 and $66,000, respectively. The SFP cost also includes the carbon steel cover described in Section 9.2.1.2.

- Hauling contractors, for transport of packaged radioactive materials from the facility to a licensed burial site. These costs are included in the estimates for Radioactive Materials Disposal. The estimated total costs for truck transport of radioactive materials from the facility to a licensed burial site are $328,000 (includes cask rental as given in Table 10.2-3). The estimated total costs for shipment of spent fuel, which is accomplished by train, are $702,800.

TABLE 10.2-2. Projected Utilization of Manpower and Staff Labor Costs During Preparations for Safe Storage

Title or Function	Time in Years Relative to Reactor Shutdown -2 0	-1 0	1 0	2 0	Total Man Years	Total Cost, $
Decommissioning Superintendent	0 5	1	1	0 33	2 83	210,835
Decommissioning Engineer	0 5	1	1	0 33	2 83	179,705
Assistant Decommissioning Engineer	0 5	1	1	0 25	2 75	120,450
Health and Safety Supervisor	0 25	1	1	0 33	2 58	130,032
Industrial Safety Specialist	0 25	1	1	0 33	2 58	113,004
Health Physicist		0 5	1	0 33	1 83	72,102
Senior Health Physics Technician		1	2	0 66	3 66	106,140
Health Physics/Laboratory Technician		1 5	3	1	5 5	137,500
Protective Clothing Attendant			2	0 66	2 66	61,712
Quality Assurance Supervisor/Engineer	0 2	1	1	0 33	2 53	110,814
Quality Assurance Technician		0 5	2	0 33	2 83	65,656
Contracts and Accounting Specialist	0 2	1	1	0 33	2 53	97,152
Accounting Clerk		0 3	1	0 33	1 63	33,089
Security Supervisor			1	0 33	1 33	43,757
Security Patrolman			15	3 75	18 75	399,375
Radioactive Shipment Specialist		1	1	0 33	2 33	76,657
Procurement Specialist	0 2	1	1		2 2	67,540
Craft Supervisor			1	0 33	1 33	52,402
Craftsman			8	2 64	10 64	288,344
Tool Crib Attendant			2	0 66	2 66	61,712
Crew Leader		1 5	3	1	5 5	180,950
Utility Operator		4	10	2 64	16 64[(b)]	450,944
Laborer		4	8	2 64	14 64	379,176
Secretary	0 5	2	3	1	6 6	131,950
Safety Review Committee Consultants	0 25	0 25	0 5	0 05	1 05	70,350
Man Years/Year	3 35	23 55	66 5	18 81		
Cost, Million of Dollars	0 1639	0 8444	2 0336	0 5994		3 6413
Total Man Years	--	--	--	--	113 2	

Reference (Also, see Appendix I 1, Salary Data for Typical Staff Positions)

(a) The detailed numbers are preserved for calculational purposes and do not imply precision to the nearest one hundred dollars

(b) Defueling and disabling operations, occurring the first two months after the final reactor shutdown, requires the services of 20 utility operators (former reactor operators), this group is then reduced in number to a total of 8 for the remainder of the Preparations for Safe Storage The 12 extra utility operators required for the 2 months of defueling and and disabling operations are equivalent to an additional 2 man years manpower utilization The 2 man years are subsequently reflected in the Total Man Years column for the utility operators

10.2.3 Estimated Costs for Disposal of Radioactive Materials

The radioactive materials that are postulated to be removed from the facility during the preparations for Safe Storage are grouped into two categories: spent fuel and radioactive waste materials (resins, filters, evaporator bottoms, combustibles). The estimated costs for disposing of these materials are summarized in Table 10.2-3, based on detailed analysis given in Appendices H and I. No final disposal costs are assigned for the spent fuel assemblies since the ultimate disposition of spent fuel is as yet undefined, and the fuel could have a positive value if recycled.

TABLE 10.2-3. Estimated Costs for Disposal of Radioactive Material

Category	Millions of Dollars			
	Container[a]	Transportation	Burial[c]	Total
Spent Fuel	None	2.467[d]	Disposition Undefined	2.467[e]
Waste Disposal	0.115	0.328[b]	0.101	0.544[f]
TOTALS	0.115	2.795	0.101	3.011

[a] Container costs as given in Table I.2-1, Appendix I.

[b] Includes cask rental for 5 days/shipment, plus trucking costs (except for spent fuel), as given in Tables I.3-1 and I.4-4, Appendix I.

[c] Handling, burial, and surchanges as given in Table I.5-1, Appendix I.

[d] Includes rental on GE IF-300 casks, and rail transport costs as given in Appendix I.4.2.

[e] The total does not include estimated staff labor costs of about $93,000 for removal of the fuel from the reactor.

[f] The total does not include estimated staff labor costs for waste disposal of about $70,000.

NOTE: Total volume of all radioactive material buried (including containers) is estimated to be 388 m^3. The total number of truck shipments is 139.

10.2.4 Nuclear Insurance Costs

The premium costs for nuclear insurance during the decommissioning period have been estimated.[1] Based on an assumed policy limit of $125 million carried through the preparations for Safe Storage until the last fuel is shipped from the site (i.e., about one year after final reactor shut down), the premiums were estimated to be in the ranges shown in Table 10.2-4.

TABLE 10.2-4. Estimated Nuclear Liability Insurance Premium During the Preparations for Safe Storage and for the Safe Storage Period

Year After Shutdown	Range of Estimated Premium ($)
1	200,000 - 234,000
2	50,000 - 60,000
3 & thereafter	1,000 - 5,000 Annually

Based on these estimates, the total cost for nuclear liability insurance during the sixteen months of preparations for Safe Storage is postulated to be $300,000.

10.2.5 Estimated Costs for Miscellaneous Supplies

Many expendable supplies are used during the decommissioning work. These include decontamination chemicals, ion exchange resins, glass-fiber and HEPA filters, cartridge-type fluid filters, disposable protective clothing, assorted cleaning agents, rags, mops, plastic bags and sheeting, and expendable tools. The estimated costs for the major items are given in Table 10.2-5.

10.2.6 Estimated Environmental Surveillance Costs

The costs for the program of environmental surveillance during the preparations for Safe Storage (described in Appendix F.5) are developed in Appendix H.3 and summarized here. The program will continue at the same level until all shipping of radioactive materials from the site is completed, approximately 16 months, for a total cost of $44,500.

[1] Personal Communication: Letter from C. R. Bardes, Nuclear Energy Liability Property Insurance Association (NEL-PIA), to R. I. Smith, Battelle-Northwest, November 18, 1976.

TABLE 10.2-5. Estimated Costs for Miscellaneous Supplies

Item	Quantity	Total Cost ($)
Decontamination Chemicals (EDTA/Oxalic Acid/Citric Acid)	3,600 to 18,000 kg[a]	23,000 to 115,000
Ion Exchange Resins	30 m^3	96,000
Filters	Unspecified	380,000
Protective Clothing (Including Laundry Service)	Unspecified	171,000
Cleaning Supplies	Unspecified	94,500
Expendable Tools	Unspecified	35,000
		799,500 to 891,500

(a) One coolant system volume of 1 to 5% solution.

10.2.7 Estimated Quality Assurance Program Costs

The costs for the program of quality assurance during the preparations for Safe Storage (described in Appendix F.4) are developed in Appendix H.3 and summarized here. The labor costs represent about 95% of all QA program costs and are included in the staff labor total costs of Table 10.2-2 for all decommissioning personnel. Materials and equipment and travel expenses total $10,000.

10.2.8 Costs of Postulated Special Tools and Equipment for Preparations for Safe Storage

A lesser number of activities is involved in meeting the need for specific tools and equipment in the preparations for Safe Storage than for Immediate Dismantlement. Essentially no dismantlement of highly activated material or equipment is necessary, thus eliminating the need for and expense of special remote handling equipment.

A listing of special tools and equipment to be utilized in preparing the reference PWR for Safe Storage, together with their functions, is given in Table 10.2-6. This listing is not intended to be all inclusive, but is

TABLE 10.2-6. Special Tools and Equipment Postulated for Preparations for Safe Storage

Item	Number Required	Unit Cost ($000)	Functions
Portable oxy-acetylene cutting torch	2	1	Used for general cutting of piping.
Guillotine pipe saws[a]	2	4	Portable power pipe saws for removal of piping.
Closed circuits, high resolution television systems	2	(Plant equipment)	For observation and control of remote operations in high radiation fields.
Submersible pumps with disposable filter cartridges	2	1.5	For rapid cleanup of pool water, collection of chips and particles.
High velocity water jets[b]	-	-	For surface decontamination of piping, tanks and equipment.
Hydraulic concrete surface spalling device	1	5	For removal of contaminated concrete surfaces. See subsequent description in Appendix F.3.3.
Concrete drills Electric/pneumatic hammers	1	0.5	For drilling holes in concrete as required for surface spalling and blasting.
Portable filter ventilation enclosures	1	1.5	For collecting and filtering air, smoke, fumes, etc., from cutting operations on contaminated materials. See subsequent description in Appendix F.3.5.
Supplied-air bubble suit	50	0.05	Provide personnel with maximum respiratory and surface protection against contamination.
Safety nets[c]	1	5 total	For use under personnel working on elevated systems and structures.
Shipping casks and liners	As required	0.1/day	For transport and burial of radioactive materials at a disposal site. See Appendix F.2 for details.
Mobile radwaste reprocessing unit[d]	1	10/month	For additional radwaste processing capacity during protective storage activities, and use after in-plant radwaste system has been deactivated.
Primary piping jumper	-	-	For bypassing the reactor vessel volume during chemical decontamination, if a multiple cycle decontamination program is followed, to reduce the volume of solutions and rinse water to be processed. See subsequent description in Appendix F.3.7.

[a] Brochures from E. H. Wachs Co., Wheeling, IL.

[b] Carver, J. Barton, "New Uses for Cleaning with Water," POWER, May 1976.

[c] Brochures from SINCO, Safety and Industrial Net Division, East Hampton, CT.

[d] Nucleonics Week, June 24, 1976.

intended to be illustrative of those major items deemed necessary for completion of the preparations for Safe Storage. Descriptions of some of the more unique nonstandard devices listed in Table 10.2-6 are presented in Appendix F.3, and are not repeated here. Estimated costs for the special tools and equipment given in Table 10.6-6 total $74,500.

10.2.9 Electrical Costs

The primary cost associated with providing essential systems and services (described in Section 9.2.5) that must remain in place during the preparations for Safe Storage is electricity. Development of the 11 MW base load assumed during the preparations for Safe Storage is given in Section 9.2.4 and is not repeated here.

The postulated distribution of the base load across the various buildings at the site during the decommissioning period is shown in Table 10.2-7.

TABLE 10.2-7. Postulated Distribution of Electrical Power Consumption During the Preparations for Safe Storage

Job Item	Assumed % of Base	MW Load
Auxiliary Building	50	5.5
Reactor Building	15	1.65
Fuel Building	15	1.65
Control Building	10	1.1
Turbine Building	5	0.55
Condensate-Demineralizer Building	3	0.33
Warehouse Building	2	0.22
Totals	100	11.00

Applying these estimated loads over the entire 16 months of decommissioning activities, as shown in Figure 10.2-1, results in an estimated total power consumption of 1.17×10^8 kWh. At a cost of 15 mils per kWh, the power costs are about $1,760,000 over the decommissioning period. Total power consumption is increased by 0.07×10^8 kWh by the chemical decontamination of the RCS and CVCS, and costs are increased by $105,000, for a total electrical power cost of $1,865,000.

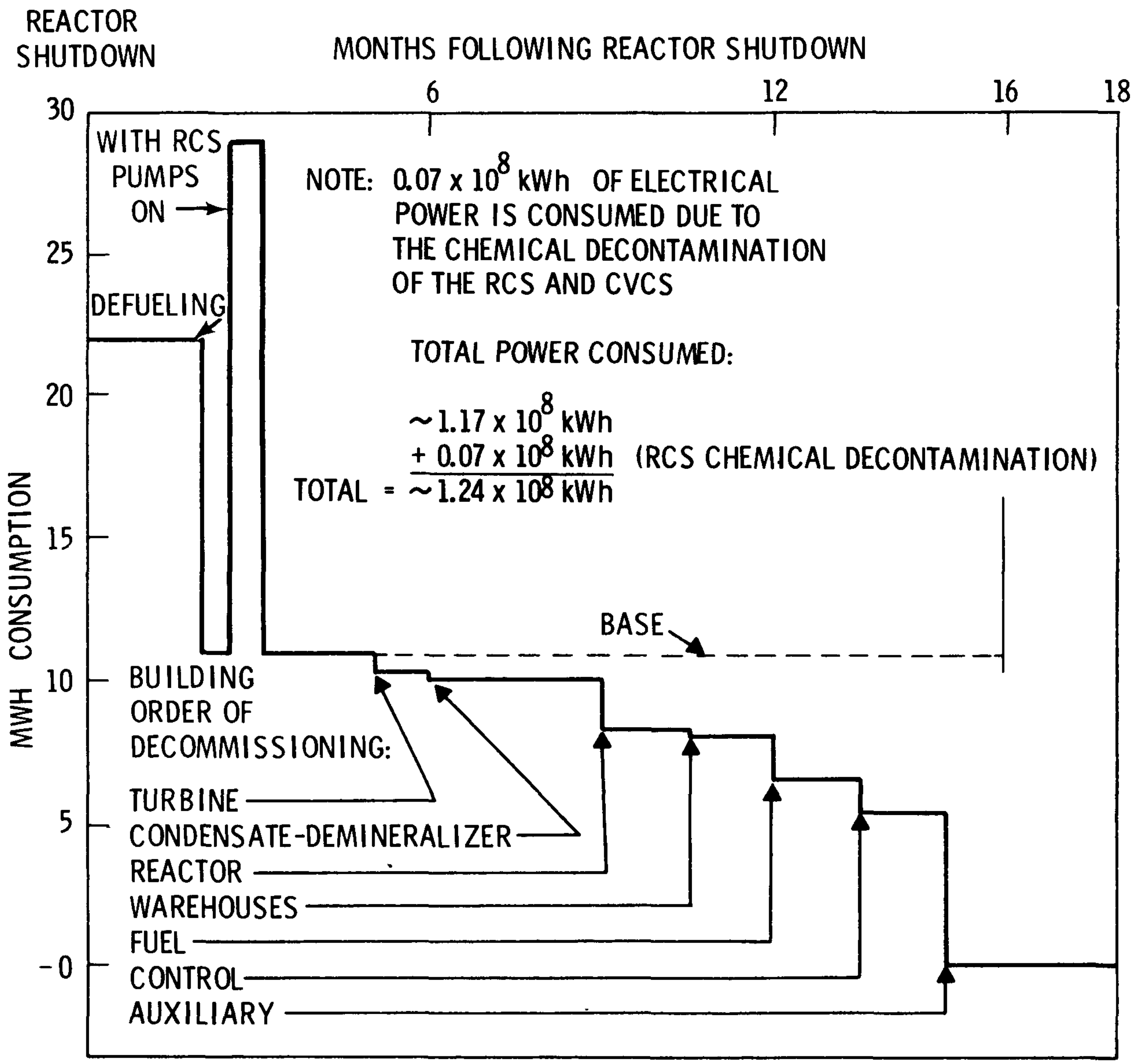


FIGURE 10.2-1. Estimated Power Consumption During the Preparations for Safe Storage Period

Factors such as time of year (effects of peak load demands), schedule slippage and/or schedule changes affecting time, and conservation measures, among others, are job specific and can grossly affect total electricity costs.

10.2.10 Estimated Costs for Decontamination and Covering of the SFP

The activities required to drain, decontaminate, and cover the Spent Fuel Pool (SFP) are described in detail in Section 9.2.1.2 and are not repeated here. The work is assumed to be done by a specialty contractor at a cost of about $66,000, and to take approximately two months to complete. The estimated costs are summarized in Table 10.2-8.

TABLE 10.2-8. Summary of Costs Postulated for Decontamination and Covering of the SFP

Item	Cost[a]
Materials	$ 7,300
Equipment and Supplies	12,500
Manpower	29,000
Subtotal	$47,800
10% General Contractor Overhead and Profit	4,780
Total	$52,580

[a] The number of figures is for computational accuracy and does not imply precision to the nearest dollar.

10.3 COST ESTIMATES FOR SAFE STORAGE (CONTINUING CARE)

The detailed cost estimates for the continuing care period are developed in Appendix H.4, with the results summarized in Table 10.3-1.

10.4 ESTIMATED COSTS FOR DEFERRED DISMANTLEMENT

The cost of deferred dismantlement of the reference PWR has been estimated assuming that dismantlement takes place starting at intervals of 10, 30, 50, and 100 years after reactor shutdown. These estimates are developed in Appendix H.5 and are summarized in Table 10.4-1, together with the costs for continuing care and for preparations for Safe Storage.

TABLE 10.3-1. Estimated Annual Manpower Utilization and Surveillance, Maintenance, and Security Costs During the Continuing Care Period of Safe Storage

Labor	Cost, $[d]
Surveillance and Maintenance (S&M) Representative (0.25 Man-Years)	6,500
Health Physicist (0.1 MY)	3,940
Secretary (0.25 MY)	5,075
Repairman (0.1 MY)	2,710
Security (0.3 MY) (see Table H.4-3, Option C for details)	8,800[a]
Third Party Inspection (0.06 MY)[b]	7,500
Environmental Radiological Monitoring Program Personnel (see Table H.4-2 for details) (0.35 MY)	16,434[c]
Quality Assurance Specialist	900
Other Costs	
Equipment and Supplies	1,000
Annual Allowance for Repairs	5,000
Utilities and Services	5,000
NEL-PIA Insurance	1,000
Subtotal	63,859
25% Contingency	15,965
TOTAL	$79,824

(a) Refer to Section H.4.2 for initial costs.

(b) Third party inspection costs are based on an assumed cost of $500 per man day.

(c) Refer to Section H.3.5 for initial costs during the first year after year after reactor shutdown.

(d) The number of figures carried is for computational accuracy and does not imply precision to the nearest dollar.

TABLE 10.4-1. Estimated Costs for Deferred Dismantlement[a]

Decommissioning Costs ($ Millions)				Number of Years
Preparations for Safe Storage	Continuing Care	Deferred Dismantlement	Total	Dismantlement Deferred
12.6	0	0		
--	0.6	37.0	50.2	10
--	2.2	37.0	51.8	30
--	3.7	30.5	46.8	50
--	7.8	30.4	50.8	100

(a) Includes 25% contingency.
(b) Values are in constant 1978 dollars.

The values in Table 10.4-1 are to be compared with the estimated cost of Intermediate Dismantlement of $42.1 million. In terms of constant dollars, it is least costly to dismantle the facility immediately. The decrease in cost for dismantlement at 50 years or later results from lower disposal costs due to lesser quantities of contaminated material for burial, due to decay of the radionuclides.

10.5 Estimated Costs for Partial Dismantlement with Safe Storage

In the event that the station owner wishes to use the outer portion of the site for non-nuclear purposes while the Exclusion Area is held in Safe Storage, a dismantlement of the outer portion of the site during preparations for Safe Storage could be performed. The structures in the region outside of the Exclusion Area are nominally uncontaminated. However, to obtain their release, an in-depth radiation survey of the structures and their internal systems and of the surrounding surface areas is required.

The accumulation of radioactive releases on the site surface during the 40 years of station operation that was calculated in Section 7 is less than the level that would cause a 1 mrem annual radiation dose to the maximally-exposed individual, as presented in Table 8.2-5 and 8.2-6. Therefore, no major decontamination effort is expected to be required. Removal of a few square meters of soil surface to a depth of a few centimeters and the removal of portions of paving material that have "hot spots" due to operational spills should be essentially all that is required to obtain release of the property. It is estimated that the cost of removal, packaging, shipment and burial at a disposal site would cost less than $50,000. These costs would be incurred during the preparations for Safe Storage rather than during Deferred Dismantlement.

Should the owner choose to demolish the structures outside the Exclusion Area, the costs would be about $2.5 million. These costs would also be incurred during the preparations for Safe Storage rather than during Deferred Dismantlement.

The net cost for the total decommissioning program, in constant dollars, is essentially the same for the case of Safe Storage with initial partial dismantlement as for the base case of Safe Storage with Deferred Dismantlement.

10.6 PRESENT VALUE ANALYSIS OF DECOMMISSIONING COSTS

The time distributions of cumulative expenditures, from Tables H.5-2 and -3 of Appendix H are presented graphically in Figure 10.6-1. In constant dollars the least costly approach to decommissioning is Immediate Dismantlement. The reduction in cumulative costs seen at 50 years is the result of lower costs for the disposal of the greatly reduced volumes of contaminated material at that time. The radioactive corrosion product contamination will have decayed to levels sufficiently low to permit unrestricted use of the bulk of the previously contaminated material. For dismantlement deferred 100 years, the decreased costs of radioactive material disposal are overshadowed by the accumulated costs of continuing care.

Assuming an annual inflation rate of 6% and an annual discount rate of 10%, the present values (at the time of reactor shutdown) of the decommissioning costs shown in Figure 10.6-1 have been computed and are presented graphically in Figure 10.6-2.

As is always the case with computing the present value of a future expenditure, the longer the payment is deferred, the smaller is its present value, providing the inflation rate is smaller than the discount rate. Thus, by deferring dismantlement for 100 years, the present value of the cost of the total decommissioning program is reduced to only about $3 million more than the initial cost of placing the facility in Safe Storage. However, the validity of present value costing over time spans greater than 10 to 20 years is rather speculative since the projections depend upon stable values of the inflation and discount rates.

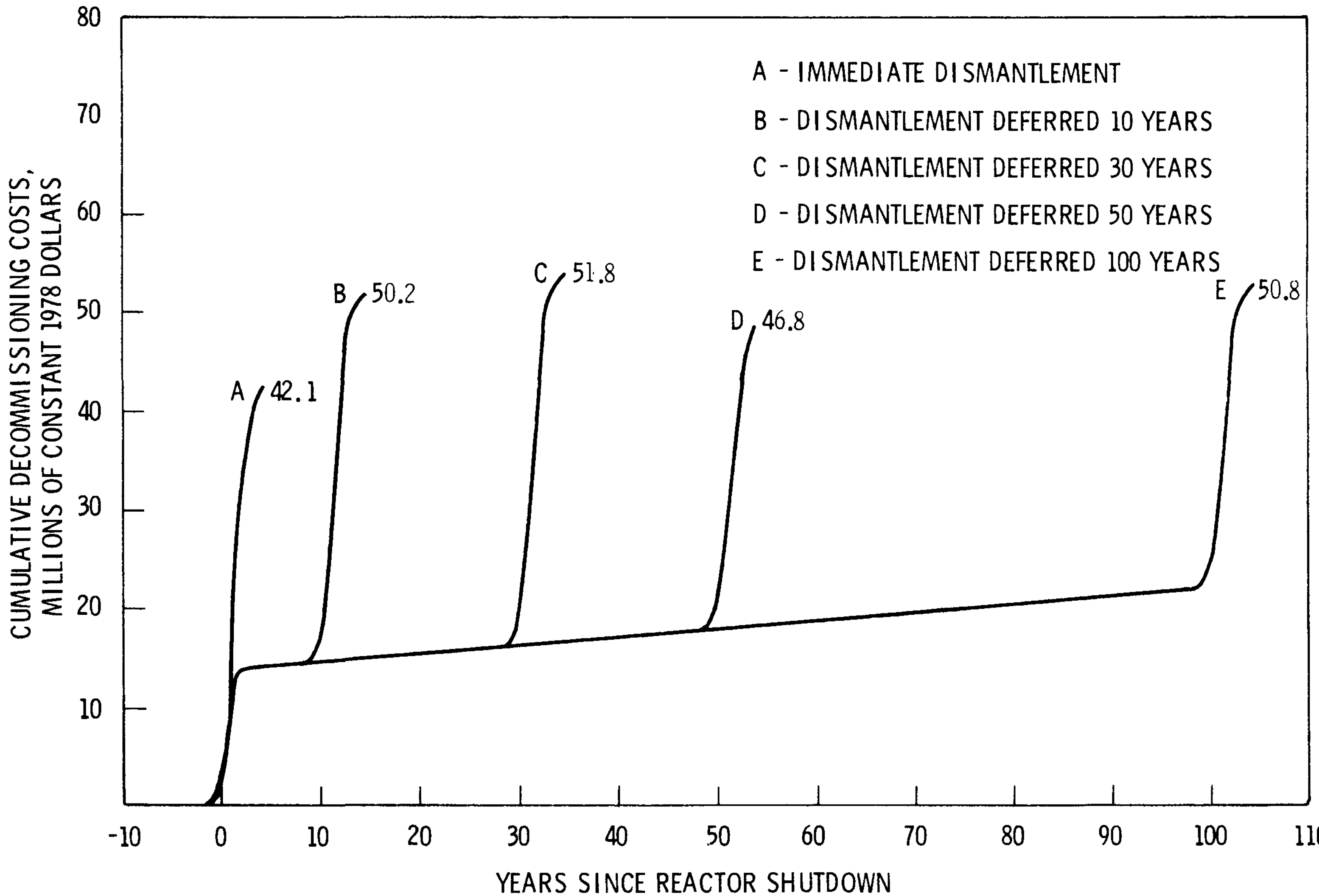


FIGURE 10.6-1. Time Distribution of Cumulative Decommissioning Costs in Constant 1978 Dollars

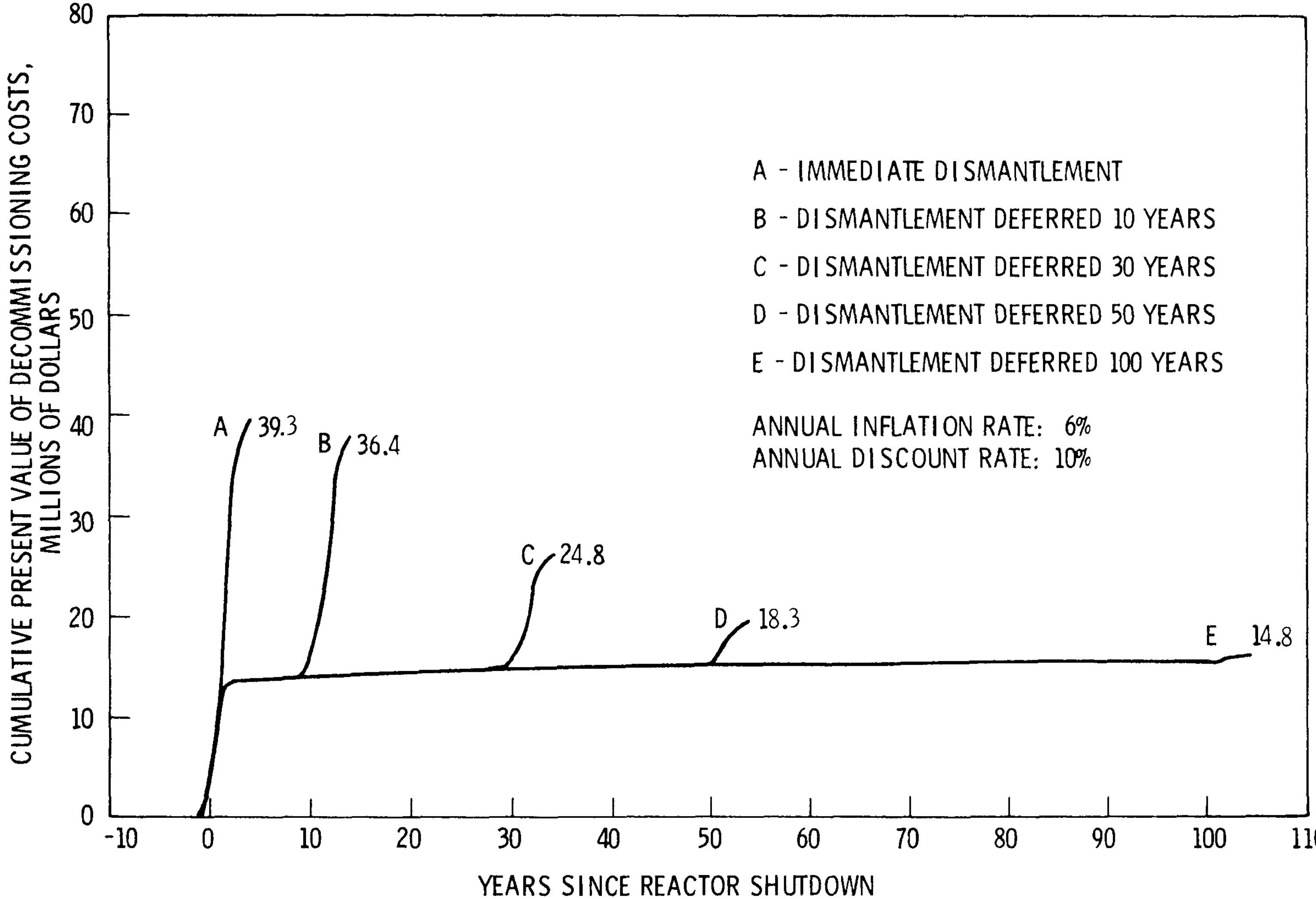


FIGURE 10.6-2. Time Distribution of Cumulative Present Values of Decommissioning Costs

REFERENCES

1. Personal Communication: Letter from C. R. Bardes, Nuclear Energy Liability Property Insurance Association (NEL-PIA), to R. I. Smith, Battelle Northwest, November 18, 1976.

11.0 PUBLIC AND OCCUPATIONAL SAFETY OF DECOMMISSIONING A REFERENCE PWR

Public and occupational safety impacts from decommissioning operations at the reference PWR nuclear power station have been identified and evaluated, and are summarized in this section. The safety evaluation includes a consideration of the impacts of public radiation exposure, occupational radiation exposure, industrial accidents, and chemical pollutants. This evaluation utilizes current data and state-of-the-art methodology to estimate the information required. A conservative approach, using parameters that tend to maximize the consequences, is used to evaluate the safety impacts of each decommissioning operation.

The safety evaluation is divided into three major parts: 1) Public Safety, 2) Occupational Safety and 3) Transportation Safety. Within each of these major parts are discussions of the radiological and nonradiological impacts of both routine and accident situations. Public radiological considerations are determined by using the atmospheric release scenarios in Appendix J and the radiation dose methodology presented in Appendix E. Occupational radiation doses are estimated using information about expected dose rates and man-hour job requirements discussed in Appendices C, G, and H.

The radiological safety evaluation is accomplished by calculating radiation doses to the public from airborne radionuclide releases, and to the decommissioning workers from external exposure to surface contamination. For Immediate Dismantlement, the fifty-year population dose commitment to the total body from routine airborne releases is calculated to be about 1×10^{-4} man-rem, and the fifty-year dose commitment to the maximum exposed individual from the worst postulated accident is calculated to be about 4×10^{-2} millirem to the total body. Radiation doses to the public resulting from preparations for Safe Storage are about a factor of 100 less than those calculated for Immediate Dismantlement. The estimated occupational radiation dose from Immediate Dismantlement is calculated to be about 1,200 man-rem, while the values for Safe Storage vary between 430 and 760 man-rem depending upon the duration of the Continuing Care period. Radioactive material transportation activities associated with decommissioning are calculated to give radiation doses to the total population along the transport route of: about 0.9 man-rem for transporting spent reactor fuel by rail, about 21 man-rem for the transportation of wastes generated during Immediate Dismantlement, and about 2.1 man-rem for the transportation of wastes generated during preparations for Safe Storage. Occupational doses from the transportation of radioactive materials are calculated to be: about 3.5 man-rem to railway workers during spent fuel shipments, about 99 man-rem from Immediate Dismantlement waste shipments, and about 10 man-rem from preparation for Safe Storage waste shipments. These radiation doses are comparable with or less than radiation doses calculated for similar activities at an operating PWR power station because of the reduced radionuclide inventories during decommissioning, carefully designed procedures that minimize atmospheric releases, and the utilization of existing process and building ventilation filtration systems.

11.1 TECHNICAL APPROACH

The results of the safety evaluation, which are summarized in this section, are based on the following key assumptions:

1) The quantities and mixtures of radionuclides in the radioactive contamination present at plant shutdown are assumed to be typical of those found at an operating PWR and are shown in the reference radionuclide inventories in Appendix J.

2) The plant process areas are assumed to have been kept relatively clean during the operating period to allow for easier operational maintenance. As a result, expected contamination levels are generally modest, but should be reasonably consistent with the quality of operation expected in modern commercial facilities.

3) Accidents that may have occurred during plant operation are assumed to be relatively minor with respect to contamination of normally clean surfaces. Any major contamination episodes are assumed to have been reasonably well cleaned up immediately following the event.

4) Decommissioning and radiation protection philosophies and techniques applied are assumed to conform to the principle of keeping occupational radiation does as low as reasonably achievable (ALARA).

5) Radioactive wastes are assumed to be sent to a regulated burial site.

6) The largest potential radiological consequence of a given decommissioning operation is assumed to be associated with performing that operation in the area with the highest radionuclide inventory.

7) The maximum release for a specific type of decommissioning operation is assumed to apply to that operation whenever it is used in the facility. In performing the dose calculations for releases of radionuclides from normal operations, the estimated releases for the entire decommissioning period are summed and assumed to be released during a one-year period.

8) All decommissioning operations are assumed to release the radionuclide mixtures that are present at plant shutdown, with no credit taken for radioactive decay. The radionuclide releases from normal operations and postulated accidents for the case of Deferred Dismantlement after a period of Continuing Care are not calculated in this analysis. The evaluation of public safety for dismantlement was done at shutdown only.

9) The atmospheric release of radionuclides is assumed to be the only source of radiation to the public from routine decommissioning operations and postulated accidents. All liquid radioactive wastes generated during decommissioning operations are assumed to be sent to the plant liquid waste storage system or to other tanks that are designated for temporary storage of these solutions. The wastes are then assumed to be processed through the waste concentration and solidification system. All systems designed to control the release of hazardous material to the environment or to noncontaminated

portions of the facility are assumed to be in operation during the chemical decontamination activities and subsequent waste processing. All liquid releases are assumed to be within the limits established for an operating PWR, and are not further analyzed in this study.

The public radiological safety evaluation is based on airborne radionuclide release scenarios for both routine and accident situations. These airborne release scenarios are discussed in Appendix J and listed here in Tables 11.1-1 and 11.1-2. These tables refer to reference radionuclide inventories that relate the release scenarios to the specific radionuclide mixtures listed in Appendix J in Tables J.3-1 through J.3-5. The reference radionuclide inventories are residual radionuclide mixtures that characterize: 1) stainless steel activation products, 2) carbon steel activation products, 3) concrete activation products, 4) activated corrosion products, and 5) general surface contamination.

A complete discussion of the occupational radiation dose calculations is contained in Appendix G and Appendix H. The occupational radiation doses are based on the estimated radiation levels in the reference PWR, given in Section 7 and Appendix C, and the man-hour job estimates for the decommissioning operations considered.

Transportation operations are examined to evaluate the safety impact of routine and accident situations. Radiation and nonradiation transportation safety impacts are evaluated for both the public and transportation workers.

11.2 PUBLIC SAFETY EVALUATION OF DECOMMISSIONING THE REFERENCE PWR

The impacts on public safety of decommissioning the reference PWR by Immediate Dismantlement, preparation for Safe Storage, and Continuing Care are evaluated for both radiological and nonradiological events. This analysis includes consideration of routine operations and of postulated accidents.

The consequences of the airborne radionuclide releases from routine decommissioning operations are calculated in terms of the radiation dose to

TABLE 11.1-1. Anticipated Airbor Radioactive Releas During Routine Decommissioning Operations(a)

Operation	Reference Radionuclide Inventory Number[b]	Immediate Dismantlement: Airborne Radioactivity Generation Rate	Immediate Dismantlement: Total Airborne Radioactivity in Building during Entire Operation μCi	Immediate Dismantlement: Estimated Atmospheric Radioactive Release μCi	Preparations for Safe Storage: Airborne Radioactivity Generation Rate	Preparations for Safe Storage: Total Airborne Radioactivity in Building during Entire Operation μCi	Preparations for Safe Storage: Estimated Atmospheric Radioactive Release μCi
Segmenting of Nonactivated Stainless							
Reactor Coolant Pumps and Primary Piping	4	8 8 μCi/min	1 5 x 10^{5}	75	–[c]		
Steam Generators	4	55 μCi/min	2 5 x 10^{1}	0 4	–		
Segmenting of Activated Reactor Vessel and Internals							
Internals	1	1150 μCi/min	3 x 10^{4}	0 5	–		
Vessel	2	20 μCi/min	4 15 x 10^{2}	0 2	–		
Waste Handling of Bioshield Concrete	3	0 04 μCi/min	3 5 x 10^{3}	1 75	–		
Surface Cleaning Operations							
Hand Held Lance							
HX Room (Auxiliary Bldg)	5	3 7 x 10^{1} μCi/min	7 8 x 10^{3}	3 9	–		
Boric Acid Evaporator Room (Fuel Building)	5	1 9 x 10^{1} μCi/min	2 0 x 10^{3}	1 0	S[d]		
Primary System	5	–			3 7 x 10^{2} μCi/min	1 8 x 10^{1}	1
Reactor Cavity							
Spent Fuel Pool	5	2 6 μCi/min	3 4 x 10^{2}	0 2	S		
Steam Generator Area (Containment)	5	7 5 x 10^{2} μCi/min	3 x 10^{2}	0 2	S		
Ion Exchanger Vault	5	3 1 x 10^{2} μCi/min	1 0	5 0 x 10^{-4}	–		
Laundry Room (Control Bldg)	5	7 4 x 10^{-4} μCi/min	4 x 10^{-4}	S			
Final Chemical Decontamination	4	4 x 10^{3} μCi/min	10^{3}	7 8 x 10^{1}	–		
Sweeping	5	–			2 8 x 10^{2} μCi/m^{3}	75	4 x 10^{2}
(Vacuuming Alternative)	5	–				1	(5 x 10^{-4})
Household Detergents	5	Insig[e]			–		
Aerosol Type Foam	5	Insig			–		
Acid Soaked Sponges	5	–			Insig		
In Situ Chemical Decontamintion							
Spray Leak	5	136 μCi/min	1 36 x 10^{3}	0 7	S		
Liquid Leak	5	0 45 μCi/min	4 5	2 3 x 10^{3}	S		
Removal of Bioshield							
Drilling	3	7 5 x 10^{-5} μCi/min (to containment)	1 2 x 10^{-1}	6 0 x 10^{-5}	–		
Explosion	3	5 3 μCi/explosion (in cavity)	10 6	5 3 x 10^{-3}	–		
Radiation Survey							
HX Room (Auxiliary Bldg)	5	9 x 10^{2} μCi/m^{3}	3 79 x 10^{2}	0 2	S		
Boric Acid Evaporator Room (Fuel Bldg)	5	2 6 x 10^{-2} μCi/m^{3}	92	0 05	S		
Steam Generator Area (Containment)	5	1 8 x 10^{-2} μCi/m^{3}	38	1 9 x 10^{2}	S		
Laundry Room (Control Bldg.)	5	1 8 x 10^{-4} μCi/m^{3}	2 x 10^{-4}	1 x 10^{-7}	S		
Removal of Concrete							
Steam Generator Enclosure							
Explosives	5	2 3 x 10^{3} μCi/ explosion	2 x 10^{2}	1 x 10^{-5}	–		
Drilling	5	1 x 10^{-4} μCi/min	5 x 10^{-3}	2 x 10^{-4}	Insig		
Pneumatic Jackhammer	5	Insig			S		
Rock Splitters	5	Insig			S		
Interfacility Transport	5	Insig			S		
Fixing Residual Contamination	5	–			Insig		
Equipment Deactivation and Isolation	5	–			Insig		
Surveillance and Maintenance (53 Years)	5	–			3 μCi/ inspection	621	3 8 x 10^{3}
On Site Retrievable Waste Storage	5	–				2 9 x 10^{4}	15

(a) The first year dose and fifty year dose commitments calculated for the maximum exposed individual and the population are listed in Tables J 4-1 through J 4-4
(b) These numbers refer to the radionuclide inventories shown in Tables J 3-1 through J 3-5
(c) A dash indicates that the operation does not apply to this decommissioning mode
(d) If the release is the same for the second mode it is marked with an S
(e) Insignificant means a building release of less than 2 x 10^{-4} μCi, and radiation doses are not calculated.

TABLE 11.1-2. Postulated Accidental Airborne Radioactive Releases During Decommissioning

Incident	Reference Radionuclide Inventory Number[b]	Immediate Dismantlement: Airborne Radioactive Release in Building (μCi)	Immediate Dismantlement: Estimated Radioactive Release to the Atmosphere (μCi)	Preparations for Safe Storage: Airborne Radioactive Release in Building (μCi)	Preparations for Safe Storage: Esimated Radioactive Release to the Atmosphere (μCi)	Estimated Frequency of Occurrence[f]
Accident during Offsite Transportation	5		$3\ 20 \times 10^{6}$	S[c]		–
Explosion of LPG Leaked from Front End Loader	5		$3\ 6 \times 10^{3}$	–[d]		Low
Explosion of Oxyacetylene during Segmenting of Vessel Shell	2	$7\ 2 \times 10^{5}$	$3\ 6 \times 10^{2}$	–		Medium
Explosion and/or Fire of Ion Exchange Resin	5	$7\ 6 \times 10^{4}$	$3\ 8 \times 10^{1}$	–		Medium
Gross Leak during In Situ Decontamination						
Spray Leak	5	$4\ 1 \times 10^{4}$	$2\ 07 \times 10^{1}$	S		Medium
Liquid Leak	5	$1\ 38 \times 10^{2}$	7×10^{-2}	S		Medium
Segmentation of RCS Piping with Unremoved Contamination	4	2.1×10^{4}	$1\ 05 \times 10^{1}$	–		High
Loss of Contamination Control Envelope during Oxyacetylene Cutting of Vessel Shell	2	$4\ 7 \times 10^{3}$	2 30	–		Medium
Vacuum Bag Rupture	5	–		2×10^{3}	1	Medium
Pressure Surge Damage to Filters during Blasting of Activated Concrete Bioshield	3	504	0 30			Low
Accidental Cutting of Contaminated Piping	4	–		$3\ 65 \times 10^{2}$	$1\ 78 \times 10^{-1}$	High
Accidental Spraying of Concentrated Contamination with High Pressure Spray	5	–		$2\ 44 \times 10^{2}$	$1\ 22 \times 10^{-1}$	High
Accidental Break of Contaminated Piping during Inspection	4	–		$2\ 2 \times 10^{2}$	<0 11	Low
Loss of Integrity of Portable Filtered Ventilation Enclosure	5	60	3×10^{-2}	–		Medium
Fire Involving Contaminated Clothing or Combustible Waste	5	12	$6\ 0 \times 10^{-3}$	S		Medium
Loss of Blasting Mat during Removal of Activated Concrete	3	5 3	2.7×10^{-3}	–		Medium
Detonation of Unused Explosives in Reactor Cavity	3	5 3	$2\ 7 \times 10^{-3}$	–		Medium
Fire in Contaminated Sweeping Compound	5	–		0 15	$7\ 5 \times 10^{-5}$	Medium
Temporary Loss of Local Airborne Contamination Control						
During Blasting	3	$2\ 7 \times 10^{-3}$	$1\ 35 \times 10^{-4}$	–		Low
During Scarfing of Contaminated Concrete Surfaces with Jackhammer	5	Insig[e]		S		–
Temporary Loss of Services	3, 4, or 5	Insig		S		–
Dropping of Concrete Rubble	5	Insig		–		–
Natural Phenomena	5	Insig		S		–
Aircraft Crashes	5	Insig		S		–
Drop of Concrete Slab during Placement	5	–		Insig		–

(a) The first year dose and fifty year dose commitments calculated for the maximum exposed individual and the population are listed in Tables J 4-5 and J 4-6
(b) These numbers refer to the radionuclide inventories shown in Tables J 3 1 through J 3-5
(c) If the release is the same for the second mode it is marked with an S
(d) A dash indicates that the accident does not apply to this decommissioning mode
(e) Insignificant means a building release of less than 2×10^{-4} μCi, and radiation doses are not calculated
(f) Frequency of Occurrence High >1×10^{-2} Medium 1×10^{-2} to 1×10^{-5} Low <1×10^{-5} per year A dash in this column means that no estimate was made for the specific incident listed

TABLE 11.1-2. Postulated Accident Airborne Radioactiv Releases During Decommissioning(a)

the maximum exposed member of the public and to the population residing within a circle of 80 km radius centered at the reference PWR. The consequences of postulated accidents are calculated in terms of the radiation dose to the maximum exposed individual. Both dose calculations use the radiation dose models and data discussed in Appendix E. An estimate of the frequency of occurrence for the accidents is given in Appendix J as being high (greater than 10^{-2} per year), medium (between 10^{-2} and 10^{-5} per year), or low (less than 10^{-5} per year) based on published values or engineering judgment or experience, since a rigorous probabilistic risk assessment is beyond the scope of this study. Inhalation of airborne radionuclides is found to be the dominant radiation exposure pathway to members of the public for most radionuclide mixtures and releases. It is felt that remedial action can be taken to mitigate other important terrestrial pathways should they appear to be significant.

Nonradiological safety areas considered include the effects of chemical residues from plant operations, chemicals used during decommissioning, and nonradioactive fission products resulting from radioactive decay.

11.2.1 Radiological Safety Evaluation of Routine Decommissioning Operations

As in an operating PWR, the primary radiological safety concern to the public during decommissioning is the loss of confinement of radioactive material resulting in public exposure to abnormally high levels of radiation. The estimated levels of radiation during the majority decommissioning are lower than for an operating PWR because of removal of the fuel elements, decay of short-lived neutron activation products, and chemical decontamination. Thus, the majority of the operations performed during PWR decommissioning offer a potential for smaller airborne releases of radionuclides than that of an operating PWR.

The primary sources of radioactive effluents from routine decommissioning operations are the release of contaminated liquid aerosols during chemical decontamination, the release of contaminated vaporized metal during equipment removal, and the release of contaminated concrete dust during decontamination or removal of concrete structures. Equipment and concrete removal operations are held to a minimum during the preparations for Safe Storage.

A complete listing of the radiation doses calculated for the airborne releases from routine decommissioning operations, shown in Table 11.1-1, is found in Section J.4 of Appendix J. Tables 11.2-1 and 11.2-2 contain summaries of the calculated radiation doses for Immediate Dismantlement and for preparation for Safe Storage. These tables show that the operations result in extremely small radiation doses as compared to the average annual radiation dose to an individual from natural sources in the United States of 80 to 170 millirem per year.[1]

Radiation doses from decommissioning operations are also lower than the allowable doses for operating LWR's as set forth in Appendix I of 10 CFR 50.[2] Estimates of the radiation doses to the public from Immediate Dismantlement operations are about 100 times larger than similar estimates for the preparations for Safe Storage, as shown in Tables 11.2-1 and 11.2-2. Since the calculated radiation doses for Immediate Dismantlement using the radionuclides inventories present at shutdown are small, and the expected radiation levels are significantly reduced by radioactive decay with time after shutdown as shown in Tables J.3-1 through J.3-5 in Appendix J, public radiation doses for Deferred Dismantlement are not calculated.

The release of radionuclides from normal Continuing Care operations is expected to be negligible compared to those during the preparations for Safe Storage. Because of the rugged construction of the PWR plant areas that contain radioactivity, the erection of rigid barriers, and the surveillance and maintenance activities, airborne radionuclide releases from normal

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

(2) Code of Federal Regulations. Title 10, Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Practicable' for Radioactive Material in Light-water-Cooled Nuclear Power Reactor Effluents." Superintendent of Documents, GPO, Washington, DC, 1977.

TABLE 11.2-1. Summary of Calculated Radiation Doses to the Maximum Exposed Individual from Airborne Radionuclides Released During Normal Decommissioning Operations

Operation	Reference Radionuclide Inventory[a]	Immediate Dismantlement: Airborne Release (μCi)	Immediate Dismantlement: First Year Dose (mrem) Total Body	Immediate Dismantlement: First Year Dose (mrem) Lung	Immediate Dismantlement: Fifty Year Dose Commitment (mrem) Total Body	Immediate Dismantlement: Fifty Year Dose Commitment (mrem) Lung	Preparations for Safe Storage: Airborne Release (μCi)	Preparations for Safe Storage: First Year Dose (mrem) Total Body	Preparations for Safe Storage: First Year Dose (mrem) Lung	Preparations for Safe Storage: Fifty Year Dose Commitment (mrem) Total Body	Preparations for Safe Storage: Fifty Year Dose Commitment (mrem) Lung
Segmenting Nonactivated Stainless Steel Coolant Pumps and Primary Piping	4	7.5×10^{1}	1.7×10^{-5}	4.3×10^{-5}	1.7×10^{-5}	5.8×10^{-5}	-[b]				
Waste Handling-Concrete Bioshield	3	1.8×10^{0}	--[c]	--	1.1×10^{-6}	1.6×10^{-6}	-				
Surface Cleaning											
Hand Held Lance											
HX Room	5	3.9×10^{0}	3.0×10^{-6}	1.3×10^{-6}	4.5×10^{-6}	1.7×10^{-6}	-				
Evaporator Room	5	1.0×10^{0}	7.7×10^{-7}	3.3×10^{-7}	1.2×10^{-6}	4.4×10^{-7}	1.0×10^{0}	7.7×10^{-7}	--	1.2×10^{-6}	--
Primary System	5						1.0×10^{0}	7.7×10^{-7}	--	1.2×10^{-6}	--
In-Situ Decontamination											
Spray Leak	5	7.0×10^{-1}	5.4×10^{-7}	--	8.1×10^{-7}	8.1×10^{-7}	7.0×10^{-1}	5.4×10^{-7}	--	8.1×10^{-7}	8.1×10^{-7}
Totals[d]		8.6×10^{2}	2.0×10^{-4}	4.9×10^{-4}	2.0×10^{-4}	6.6×10^{-4}	2.7×10^{0}	2.1×10^{-6}	8.9×10^{-7}	3.2×10^{-6}	1.7×10^{-6}

(a) These numbers refer to the radionuclide inventories listed in Appendix J, Tables J.3-1 through J.3-5.

(b) A dash indicates that this operation does not apply for the Decommissioning Mode shown.

(c) Two dashes indicates doses less than 5×10^{-7} mrem.

(d) The average annual total body dose to an individual in the U.S. from natural sources ranges from 80 to 170 mrem.[1]

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

TABLE 11.2-2. Summary of Radiation Doses to the Population from Airborne Radionuclides Released During Normal Decommissioning Operations

Operation	Reference Radionuclide Inventory[a]	Immediate Dismantlement: Airborne Release (μ Ci)	Immediate Dismantlement: First Year Dose (man-rem) Total Body	Immediate Dismantlement: First Year Dose (man-rem) Lung	Immediate Dismantlement: Fifty Year Dose Commitment (man-rem) Total Body	Immediate Dismantlement: Fifty Year Dose Commitment (man-rem) Lung	Preparations for Safe Storage: Airborne Release (μ Ci)	Preparations for Safe Storage: First Year Dose (man-rem) Total Body	Preparations for Safe Storage: First Year Dose (man-rem) Lung	Preparations for Safe Storage: Fifty Year Dose Commitment (man-rem) Total Body	Preparations for Safe Storage: Fifty Year Dose Commitment (man-rem) Lung
Segmenting Nonactivated Stainless Steel Coolant Pumps and Primary Piping	4	7.5×10^{1}	1×10^{-5}	4×10^{-5}	1×10^{-5}	6×10^{-5}	-[b]				
Waste Handling-Concrete Bioshield	3	1.8×10^{0}	--[c]	--	1×10^{-6}	2×10^{-6}	-				
Surface Cleaning											
Hand Held Lance											
HX Room	5	3.9×10^{0}	2×10^{-6}	9×10^{-7}	3×10^{-6}	1×10^{-6}	-				
Evaporator Room	5	1.0×10^{0}	6×10^{-7}	--	9×10^{-7}	--	1.0×10^{0}	6×10^{-6}	--	9×10^{-7}	--
Primary System	5	-					1.0×10^{0}	6×10^{-6}	--	9×10^{-7}	--
In-Situ Decontamination											
Spray Leak	5	7.0×10^{-1}	--	--	6×10^{-7}	--	7.0×10^{-1}	--	--	6×10^{-7}	--
Totals[d]		8.6×10^{2}	1×10^{-4}	1×10^{-4}	1×10^{-4}	7×10^{-4}	2.7×10^{0}	1×10^{-6}	--	2×10^{-6}	--

(a) These numbers refer to the radionuclide inventories listed in Appendix J, Tables J.3-1 through J.3-5.

(b) A dash indicates that this operation does not apply for the decommissioning mode shown.

(c) Two dashes indicates doses less than 5×10^{-7} man-rem.[1]

(d) The total body first year doses to an average individual are 3×10^{-8} and 2×10^{-10} mrem for Immediate Dismantlement and Safe Storage, respectively. For comparison, the average annual dose to an individual in the U.S. from natural sources ranges from 80 to 170 mrem.[2]

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

Continuing Care operations are expected to be extremely small and the radiation dose to the public is expected to be negligible. The decontamination level at the end of Continuing Care is assumed to be consistent with the 1 and 25 millirem per year maximum annual dose range discussed in Section 8.

11.2.2 Radiological Safety Evaluation of Postulated Decommissioning Accidents

The primary impact of decommissioning accidents is the release of radioactive materials into the environs and the resulting public radiation exposure. Decommissioning procedures are analyzed and accidents are postulated that result in airborne radionuclide releases in Appendix J. A summary of the accidents and releases is found in Table 11.1-2. A wide spectrum of accidents is considered in Appendix J along with the calculations, basis, assumptions, and resulting radiation doses to the maximum exposed individual. Table 11.2-3 contains a summary of the higher consequence accidents postulated for Immediate Dismantlement and for preparations for Safe Storage and the associated calculated radiation doses. A complete listing of the radiation doses considered is found in Appendix J, Tables J.4-5 and J.4-6.

The major accident postulated for Immediate Dismantlement is an explosion of Liquid Petroleum Gas (LPG) leaked from a front end loader. It is postulated that loose contaminated dust resulting from previous operations is made airborne by the heat, mechanical, and aerodynamic disturbances created by the explosion. In addition, the pressure resulting from the explosion is calculated to be sufficient to cause failure of the HEPA filter in the control enevlope and to suspend in air the loose contaminated dust on the inside surface of the control envelope. The final bank of HEPA filters on the Reactor Contaminent Building Ventilation System is assumed to remain intact. The radionuclides released directly to the atmosphere are calculated to be 3.6×10^3 μCi. The frequency of occurrence of this accident with the severity is estimated to be in the low range.

In preparations for Safe Storage, the dominant accident identified is a spray leak of contaminated solution during decontamination of the Reactor

TABLE 11.2-3. Summary of Radiation Doses to the Maximum Exposed Individual from Accidental Airborne Radionuclide Releases During Decommissioning Operations

Incident	Reference Radionuclide Inventory[a]	Immediate Dismantlement: Airborne Release (µCi)	Immediate Dismantlement: First Year Dose (mrem) Total Body[b]	Immediate Dismantlement: First Year Dose (mrem) Lung	Immediate Dismantlement: Fifty Year Dose Commitment (mrem) Total Body	Immediate Dismantlement: Fifty Year Dose Commitment (mrem) Lung	Preparations for Safe Storage: Airborne Release (µCi)	Preparations for Safe Storage: First Year Dose (mrem) Total Body[b]	Preparations for Safe Storage: First Year Dose (mrem) Lung	Preparations for Safe Storage: Fifty Year Dose Commitment (mrem) Total Body	Preparations for Safe Storage: Fifty Year Dose Commitment (mrem) Lungs	Estimated Frequency of Occurrence[c]
Explosion of LPG Leased from a Front End Loader	5	3.6×10^{3}	3.6×10^{-2}	4.7×10^{-2}	4.4×10^{-2}	5.4×10^{-2}	--[d]					Low
Explosion of Oxyacetylene During Segmenting of the Reactor Vessel Shell	2	3.6×10^{2}	4.3×10^{-5}	6.1×10^{-3}	6.9×10^{-3}	6.9×10^{-3}	--					Medium
Explosion and/or Fire in the Ion Exchange Resin	5	3.8×10^{1}	3.8×10^{-4}	5.0×10^{-4}	4.6×10^{-4}	5.7×10^{-4}	--					Medium
Gross Leak During In Situ Decontamination	5	2.1×10^{1}	2.1×10^{-4}	2.8×10^{-4}	2.5×10^{-4}	3.2×10^{-4}	2.1×10^{1}	2.1×10^{-4}	2.8×10^{-4}	2.5×10^{-4}	3.2×10^{-4}	Medium
Segmentation of RCS Piping with Unremoved Contamination	4	1.1×10^{1}	4.6×10^{-6}	7.3×10^{-4}	4.8×10^{-6}	7.9×10^{-4}	--					High
Loss of Contamination Control Envelope During Oxyacetylene Cutting of the Reactor Vessel Shell	2	2.3×10^{0}	--[e]	--	--	4.4×10^{-4}	--					Medium
Vacuum Bag Rupture	5	--					1.0×10^{0}	1.1×10^{-6}	1.3×10^{-5}	1.2×10^{-5}	1.5×10^{-5}	Medium
Accidental Cutting of Contaminated Piping	4	--					1.8×10^{-1}	--	1.2×10^{-5}	--	1.3×10^{-5}	High
Accidental Spraying of Concentrated Contamination with the High Pressure Spray	5	--					1.2×10^{-1}	--	1.6×10^{-6}	1.5×10^{-6}	1.8×10^{-6}	High

(a) These numbers refer to the radionuclide inventories listed in Appendix J, Tables J 3-1 through J 3-5

(b) The average annual total body dose to an individual in the U S from Natural Sorces Ranges from 80 to 170 mrem [1]

(c) Frequently of occurrence High 1×10^{-2}, medium 1×10^{-2} to 1×10^{-5}, Low $< 1 \times 10^{-5}$ per year

(d) A dash indicates that this accident does not apply to the decommissioning mode shown

(e) Two dashes indicate doses less than 1×10^{-6} *mrem*

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

Coolant System (RCS) or the Chemical and Volume Control System (CVCS). The leak is postulated to result from either a leak that developed during previous reactor operations or from human error during decommissioning rather than from corrosion or abrasion by decontamination solutions. The leak is assumed to occur for 30 minutes before corrective action stops the flow and 4×10^4 μCi are assumed to be made airborne. The leak can occur in the Reactor, Fuel or Auxiliary building. The total leak to the atmosphere through the building HEPA filter is about 20 μCi. The frequency of occurrence of this accident is estimated to be in the medium range.

Radiation doses to the population are not calculated for decommissioning accidents. The segment of population exposed under accident conditions is different for each site sector. A conservative upper limit estimate can be made of the radiation dose to the population by comparing Table 11.2-3 with Tables 11.2-1 and 11,2-2. For a release with the same spectrum and quantity of radionuclides, the ratio of the radiation dose to the maximum individual for an accident to that from routine operations can be calculated. This ratio times the population radiation dose for the release from routine operations gives an upper limit for radiation dose to the populace from an accidental release. The actual radiation dose to the populace for an accidental release of radioactivity is expected to be below this upper value.

Contamination remaining during the Continuing Care period is fixed firmly in place and is not readily available for airborne release. The PWR during Continuing Care is almost entirely nonoperational with no operating components except for automatic monitoring systems. Only low probability events with causes external to the plant, such as earthquakes, or certain man-related events, such as intrusion into the facility, appear to have the potential to release radioactive material to the environs. The combination of the low probability of the initiating events and the immobility of the radionuclide inventories reduces the impact of the postulated accidents during Continuing Care to levels far below those postulated for the other decommissioning operations.

11.2.3 Nonradiological Safety Evaluation

Chemical pollutants that could be released during decommissioning operations were examined and the quantities were found to be insignificant.

Potentially hazardous chemicals were found to come from three sources: 1) residuals from PWR operations, 2) chemicals employed to chemically decontaminate, and 3) nonradioactive fission products resulting from radioactive decay. The small quantities of the hazardous chemicals used and the low likelihood of their dispersal into the environs suggest that chemical pollutants from decommissioning operations do not pose a significant public hazard.

11.3 OCCUPATIONAL SAFETY EVALUATION OF DECOMMISSIONING THE REFERENCE PWR

Occupational safety impacts for Immediate Dismantlement and Safe Storage are evaluated for both radiological and nonradiological events. The analysis considers routine radiological events and postulated nonradiological accidents.

Radiation doses to workers are calculated based on the estimated radiation levels in various areas of the reference PWR, and the estimated labor requirements to perform the decommissioning work. Summaries of the detailed information in Appendices C, G, and H are given in this section. An estimate of worker injuries and fatalities resulting from decommissioning operations is made and presented, based on nuclear industry experience.

11.3.1 Radiological Safety Evaluation of Routine Decommissioning Operations

Summaries of the estimated external occupational radiation exposure for Immediate Dismantlement, and preparations for Safe Storage are given in Tables 11.3-1 and 11.3-2. These tables contain a listing of the work tasks in each of the reference PWR buildings, man-hour estimates of each task, and man-rem estimates of the accumulated external radiation exposure to all workers doing each task.

The radiation doses to decommissioning workers are calculated using the man power requirements estimated for each job and estimates of the average radiation dose rates associated with each job. The dose rate estimtes are based on the data given in Appendix C that were measured at operating reactors during refueling outages. The measured data have been adjusted to reflect the dose rate reduction achieved from a chemical decontamination of the fluid systems. The radiation doses computed in Appendices G and H are based on constant values of the adjusted dose rate data, regardless of how long after reactor shutdown the specific job was performed. Those dose estimates are

TABLE 11.3-1. Summary of the Estimated External Occupational Radiation Exposure for the Immediate Dismantlement of the Reference PWR[a,b]

Description	Estimated Total (man hours)[c,d]	Event Total Dose (man-rem)[c]	Decay[e] Factor	Corrected Dose, man-rem
Reactor Building				
Move internals to refueling cavity	240	8.0	0.98	7.8
Comprehensive predismantling radiation survey	300	9.0	0.96	8.6
Segment vessel internals and load containers	18,000	92	0.94	86.5
Segment reactor pressure vessel and load containers	16,000	78	0.86	67.1
Segment steam generators (4) and prepare for shipment	14,000	140	0.82	118.1
Remove RCS pumps (4) and RCS piping	6,700	67	0.80	53.6
Remove pressurizer, relief tank, and safety injection system	4,300	65	0.78	50.7
Remove HX's and assorted pumps	2,900	29	0.77	22.3
Remove contaminated internal structures	16,000	81	0.76	61.6
Remove spray piping and ventilation systems	3,400	12	0.76	8.6
Decontaminate remaining internal surfaces	5,800	6.0	0.71	4.3
Reactor Building Total	89,000	590		489
Auxiliary Building				
Remove electrical equipment	570	1.5	0.70	4.0
Decontamination (contact work and internal flushes)	1,400	13	0.94	12.2
Remove selected building internal (plugs, walls, platforms) and package	680	1.5	0.85	1.3
IX resin and filters removal	280	23	0.78	17.9
Remove IX and filters system piping	1,300	96	0.77	73.9
Remove tanks, pumps, and HX's	3,000	150	0.75	115.5
Remove HVAC and fire sprinkler systems and monorails	2,300	7.0	0.73	5.1
Auxiliary Building Total	9,600	300		227
Fuel Building				
Remove CVC system	1,400	85	0.93	79.1
Remove condensate holding tank system	210	7.0	0.88	6.2
Remove new fuel storage racks and install electropolishing system	190	1.0	0.98	1.0
Remove boric acid system, tanks, and piping	660	34	0.91	30.9
Remove CCW system	660	4.0	0.81	3.2
Remove spent fuel racks, fuel transfer system, and peripheral equipment	410	8.0	0.84	6.7
Remove SFP recirculation system, liners, and contaminated concrete	570	8.0	0.83	6.6
Fuel Building Total	4,100	150		134
Ancillaries				
Comprehensive plant radiation survey	400	2.0	1.0	2.0
Special radiation surveys	1,300	6.3	1.0	6.3
RCS and CVCS chemical decontamination	130	26	0.97	25.6
Decontamination (electropolishing) and disposal (packaging and shipping of contaminated equipment and debris)	930	4.6	0.85	3.9
Packaging (combustible wastes)	970	29	0.85	24.8
Fuel handling (defuel reactor and ship fuel offsite)	14,000	140	1.0	140
Miscellaneous	17,000	170	1.0	170
Ancillaries Total	35,000	380		373
Total for Immediate Dismantlement	140,000	1,400		1,223

(a) These estimates are based on the assumption that the RCS, CVCS, and other related systems have been chemically decontaminated internally.
(b) For a detailed discussion of these estimates, consult Appendix G, Table G.3-1.
(c) Estimates of man hours of labor and man-rem of radiation dose have been rounded to two significant figures.
(d) Man hours of labor listed are estimated hours spent in radiation zones.
(e) Fractional reduction in ^{60}Co radioactivity over the period from reactor shutdown to the time midpoint of the decommissioning task.

TABLE 11.3-2. Summary of the Estimated External Occupational Radiation Exposure for Preparations for Safe Storage[a,b]

Description	Estimated Total (man hours)[c,d]	Event Total Dose (man-rem)[c]	Decay[e] Factor	Corrected Dose, man-rem
Reactor Building				
Comprehensive predecommissioning survey	220	6.6	0.98	6.47
Decontaminate reactor and refueling cavities (contractor)		4.0	0.92	3.69
Chemical decontamination (RCS, CVS, and related systems)				
Valving, readings, etc	65	1.6	0.96	1.54
Evaporators maintenance and repair	62	25	0.96	24.0
Purge RCS and CVC systems	350	1.1	0.94	1.03
Decontaminate remaining internal surfaces and place reactor building in Safe Storage	5,800	17	0.92	15.92
Intrusion, radiation, and fire alarm installation (contractor)	88	4.5	0.91	4.10
Final Reactor Building radiation survey	350	1.1	0.91	1.00
Reactor Building Total	6,900	61		58
Auxiliary Building				
Comprehensive predecommissioning survey	720	3.6	0.87	3.13
IX resin and filters removal	280	23	0.87	20.01
Decontamination (contact work and internal flushes)	700	6.3	0.86	5.42
Place auxiliary building in Safe Storage	160	0.16	0.85	0.14
Intrusion, radiation, and fire alarm installation (contractor)	30	0.67	0.85	0.57
Final Auxiliary Building radiation survey	790	0.79	0.84	0.66
Auxiliary Building Total	2,700	34		30
Fuel Building				
Comprehensive predecommissioning survey	200	1.2	0.88	1.06
Decontamination (contact work and internal flushes)	480	9.6	0.87	8.35
Miscellaneous work	130	0.13	0.87	0.11
Place the spent fuel pool in Safe Storage (contractor)		5.0	0.87	4.35
Intrusion, radiation, and fire alarm installation (contractor)	30	0.67	0.86	0.58
Final Fuel Building radiation survey	360	0.36	0.86	0.31
Fuel Building Total	1,200	17		15
Ancillaries				
Radiation surveys (weekly)				
Reactor Building	160	0.78		
Auxiliary Building	54	0.27	1.0	1.32
Fuel Building	54	0.27		
Packaging (combustible wastes)				
Reactor Building	110	3.3		
Auxiliary Building	43	1.3	0.85	5.10
Fuel Building	47	1.4		
Fuel handling				
Defuel the reactor	8,000	72	1.0	72
Ship fuel offsite	6,500	65	1.0	65
Miscellaneous work	-	170	1.0	170
Ancillaries Total	15,000	310		313
Total for Preparations for Safe Storage	26,000	430		416

(a) These estimates are based on the assumption that the RCS, CVCS, and related systems have been chemically decontaminated internally

(b) For a detailed discussion of these estimates, consult Appendix H, Table H.2.1

(c) Estimates of man hours of labor and man-rem of radiation dose have been rounded to two significant figures

(d) Man hours of labor are estimated hours spent in radiation zones

(e) Fractional reduction in ^{60}Co radioactivity over the period from reactor shutdown to the time midpoint of the decommissioning task presented in the second column of Tables 11.3-1 and 11.3-2. In fact, the radioactive materials that produce the radiation dose rates are decaying throughout the decommissioning period. Evidence from operating reactor facilities supports the assumption that the majority of the radiation dose from activated corrosion products that are deposited throughout the interior of the piping systems is produced by ^{60}Co. Trends in data measured at operating reactors show the proportion of ^{60}Co increasing with years of operation. Therefore, to correct the estimated occupational radiation doses for radioactive decay since reactor shutdown, the fractional reduction in ^{60}Co radioactivity occurring over the time from reactor shutdown to the time midpoint of each of the decommissioning tasks is computed and listed in the third column of Tables 11.3-1 and 11.3-2. Multiplying the estimated radiation dose in column 2 by the decay factor in column 3 yields the corrected radiation dose listed in column 4. Several major tasks are not decay corrected since the radiation dose estimates for those tasks are based directly on measured data. These corrections reduce the cumulative radiation exposure estimates for Immediate Dismantlement by about 12% and for preparations for Safe Storage by about 3%. These corrections have no impact on the radiation doses estimated for continuing care and for deferred dismantlement.

The total occupational radiation dose for Immediate Dismantlement is estimated to be about 1200 man-rem. The dismantlement activities in the PWR Reactor Building are the main contributors to this total. Specific dismantlement activities that result in the highest worker radiation doses are: 1) segmentation and packaging operations for the reactor pressure vessel and vessel internals, 2) equipment removal in the Auxiliary Building, 3) segmentation and packaging operations for the four steam generators, and 4) fuel removal and subsequent shipping operations. The estimated occupational exposure for preparations for Safe Storage is about 420 man-rem.

The average quarterly radiation doses to decommissioning workers for Immediate Dismantlement and for preparations for Safe Storage are listed in Table 11.3-3. These quarterly radiation doses are conservatively high since the radioactive contamination levels present at plant shutdown are used in these calculations, with credit taken only for the radioactive decay of ^{60}Co and no credit taken for the decay of the short-lived activation products during the dismantlement period.

TABLE 11.3-3. Estimated Quarterly Occupational Radiation Dose From Decommissioning Operations

Mode	Total Man-rem Per Mode	Decommissioning Technicians[a] Man-yr	Decommissioning Technicians[a] Average Dose rem/quarter	Estimated Average for all Decommissioning Staff[b] Man-Yr	Estimated Average for all Decommissioning Staff[b] rem/quarter
Immediate Dismantlement	1200	140[c]	2.2	300[c]	1.0
Preparation for Safe Storage	420	55[d]	1.9	110[d]	0.95

[a] All personnel are assumed to be trained radiation zone workers.
[b] Values are rounded to two significant figures.
[c] Based on Table 10.1-2.
[d] Based on Table 10.2-2.

The surveillance and maintenance staff will be exposed to the residual radiation levels present in the reference PWR during the Continuing Care period. During this period, the radiation levels will be continuously declining by radioactive decay. The dominant isotope during Continuing Care is assumed to be ^{60}Co. Table 11.3-4 lists a summary of the man-hours of labor and man-rem of occupational dose accumulated for Continuing Care periods of 1, 10, 30, 50, and 100 years. The majority of the occupational dose is accumulated during the first 30 years of Continuing Care.

The estimated external occupational radiation doses for decommissioning the reference PWR are summarized in Table 11.3-5. The total occupational dose for Immediate Dismantlement is given, and a break-down of Safe Storage into preparations for Safe Storage, Continuing Care, and Deferred Dismantlement is presented. Occupational radiation doses for Deferred Dismantlement are calculated by reducing the Immediate Dismantlement doses in proportion to the decay of ^{60}Co over the time period of interest. Thus, if a given task performed immediately after shutdown caused a radiation exposure proportional to the amount of radioactive material present, N_o, that same task performed t years later during Deferred Dismantlement would cause an exposure proportional to the amount of radioactive material present, then $N(t) = N_o e^{-\lambda t}$, where λ is the decay constant for ^{60}Co in years. This is a conservative assumption as

TABLE 11.3-4. Summary of the Estimated External Occupational Radiation Exposure During Continuing Care[a,b]

Time After Reference PWR Final Shutdown (years)	Estimated Accumulated Man Hours[c]	Accumulated Radiation Dose (man-rem)[c]
1	1.5×10^4	1.9
10	1.5×10^5	9.9
30	4.5×10^5	14
50	7.5×10^5	14
100	1.5×10^6	14

(a) The facility radiation levels are assumed to decline at a rate governed by the half-life of ^{60}Co

(b) For a more detailed discussion, see Section 9 and Appendix H

(c) Estimates of man hours of labor and man-rem of radiation dose have been rounded to two significant figures

TABLE 11.3-5. Summary of the Estimated External Occupational Radiation Doses for Decommissioning the Reference PWR

Decommissioning Mode	Time After Reactor Shutdown (Years)	Estimated Dose (man-rem)[a]
Immediate Dismantlement	0	1200
Safe Storage[b]		
Preparations for Safe Storage	0	420
Continuing Care	10	10
	30	14
	50	14
	100	14
Deferred Dismantlement	10	330
	30	24
	50	2.0
	100	1.0
Total for Safe Storage[b] with Deferred Dismantlement in year	10	760
	30	450
	50	440
	100	430

(a) Estimates of man-rem of radiation dose have been rounded to two significant figures

(b) Safe Storage consists of three phases: Preparations for Safe Storage, Continuing Care, and Deferred Dismantlement previously stated since the radiation levels at reactor shutdown are controlled by radionuclides with half-lives shorter than that of ^{60}Co. The decline in the radiation dose rate from the decay of residual radioactive contamination is controlled by ^{60}Co at 3 years and later after reactor shutdown as shown in Figure 7.4-3. Reducing the Immediate Dismantlement occupational radiation doses in proportion to the decay of ^{60}Co over the time interval of interest is necessarily based on the assumption that the decommissioning operations are performed the same way at each time period. For times of 30 years or longer after shutdonw, the preparations for Safe Storage contribute the majority of the total occupational radiation dose accumulated during the total decommissioning program.

The estimates of the occupational radiation exposure are sensitive to management philosophy and to the decommissioning methods utilized. Administrative controls are assumed to be in place that keep radiation records for each individual and assure that no one worker exceeds recommended limits. Estimates contained in Table 11.3-5 are based on decommissioning methods that utilize shielding devices and highly trained technicians. Different basic assumptions, decommissioning procedures or increased manpower may change the occuaptional radiation dose estimates significantly.

11.3.2 Safety Evaluation of Construction or Industrial Accidents

The potential exists for worker injuries and fatalities as a result of decommissioning operations. As for any industrial operation, proper management and industrial safety practices during decommissioning will minimize the potential for worker accidents. Estimates of worker injuries and fatalities were based on data provided by the U.S. AEC for the period 1943-1970.[3] Table 11.3-6 lists the estimates of worker injuries and fatalities for heavy construction, light construction and operational activities that are conducted during dismantlement and preparation for Safe Storage.[4] As shown in the table, about 4.0 lost-time injurises and 0.023 fatalities are expected during 4 years of dismantlement. For preparations for Safe Storage, about 0.9 injuries and 0.006 fatalities are expected furing 16 months of decommissioning activities.

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(3) Operational Accidents and Radiation Exposures Experienced Within the USAEC 1943-1970. WASH-1192, 1971.

(4) American National Standards Institute, Method of Recording and Measuring Work Injury Experience. ANSI 216.1, 1967.

TABLE 11.3-6. Estimated Occupational Lost-Time Injuries and Fatalities from Decommissioning Operations[a]

Activity	Frequency Accidents/10^6 man-hr: Lost-Time Injuries[b]	Frequency Accidents/10^6 man-hr: Fatalities	Immediate or Deferred Dismantlement: Man Hours[d]	Immediate or Deferred Dismantlement: Lost-Time Injuries	Immediate or Deferred Dismantlement: Fatalities	Preparations for Safe Storage: Man Hours[d]	Preparations for Safe Storage: Lost-Time Injuries	Preparations for Safe Storage: Fatalities
Heavy construction[c]	10.0	4.2×10^{-2}	1.6×10^{5}	1.6	6.7×10^{-3}	--	--	--
Light construction	5.4	3.0×10^{-2}	3.4×10^{5}	1.8	1.0×10^{-2}	1.2×10^{5}	6.5×10^{-1}	3.6×10^{-3}
Operational support	2.1	2.3×10^{-2}	2.8×10^{5}	0.59	6.4×10^{-3}	1.0×10^{5}	2.1×10^{-1}	2.3×10^{-3}
TOTALS				4.0	2.3×10^{-2}		8.6×10^{-1}	5.9×10^{-3}

(a) Estimates of injuries and fatalities have been rounded to two significant figures.

(b) Lost-time injuries are defined in Reference 5.

(c) Primarily facility demolition work.

(d) Labor estimates in man hours are given in Tables 10.1-2 and 10.2-2.

(5) American National Standards Institute, "Method of Recording and Measuring Work Injury Experience." ANSI 216.1, 1967.

Estimates of the number of injuries and fatalities that could occur to the surveillance and maintenance staff from industrial-related accidents for Continuing Care operations is given in Table 11.3-7. As shown in this table, about 1 lost-time injury and 8.7×10^{-3} fatalities could be expected over a 100 year period of Safe Storage.

11.4 TRANSPORTATION SAFETY EVALUATION FOR DECOMMISSIONING THE REFERENCE PWR

Radioactive materials are packaged and shipped offsite for burial during decommissioning of the reference PWR. Spent fuel from the final reactor core is assumed to be shipped to an unspecified fuel repository, located about 2,400 km (1,500 mi) from the reactor site by rail, as an initial operation of both the Immediate Dismantlement and Safe Storage decommissioning modes. Radioactive wastes (all non-TRU) generated during these modes are assumed to be shipped by truck to a shallow land burial facility located about 800 km (500 mi) away from the site. These wastes contain mainly small amounts of fission products from surface cleaning operations and large amounts of neutron-activated corrosion products from structural materials, and reactor components. The procedures and standards for the packaging and transport of the radioactive materials are discussed in Section 5 and Appendix F, and are summarized here. To minimize the risk that radioactive shipments pose to the public and to transportation workers, federal and state regulations prescribe the containers, contents, packaging, handling, and burial requirements.

11.4.1 Radiological Safety Evaluation of Routine Transportation Operations

Shipments of spent reactor fuel from the final reactor core and radioactive wastes from decommissioning operations will be made in either special rail casks or exclusive use vehicles. Department of Transportation regulations[5] set the following limits on radiation levels associated with radioactive material shipments:

- 100 mR/hr at 0.91 m (3ft) from the external surface of the package (provided the package is transported in a closed vehicle)

[5] U.S. Code of Federal Regulations. Title 49, Parts 170-189, "Transportation." Superintendent of Documents, GPO, Washington, DC, 20555.

TABLE 11.3-7. Estimated Occupational Lost-Time Injuries and Fatalities from Continuing Care Operations[a]

Activity	Estimated Man Hours/yr[b]	Frequency of Accidents per 10^6 Man Hours: Lost-Time Injuries[c]	Frequency of Accidents per 10^6 Man Hours: Fatalities	10 Years: Lost-Time Injuries	10 Years: Fatalities	30 Years: Lost-Time Injuries	30 Years: Fatalities	50 Years: Lost-Time Injuries	50 Years: Fatalities	100 Years: Lost-Time Injuries	100 Years: Fatalities
				Estimate of the Number of Occupational Safety Accidents per Surveillance Period							
Surveillance	2800	2 1	2 3 x 10^{-2}	5 9 x 10^{-2}	6 4 x 10^{-4}	1 8 x 10^{-1}	1 9 x 10^{-3}	2 9 x 10^{-1}	3 2 x 10^{-3}	5 9 x 10^{-1}	6 4 x 10^{-3}
Maintenance	770	5 4	3 0 x 10^{-2}	4 2 x 10^{-2}	2 3 x 10^{-4}	1 2 x 10^{-1}	6 9 x 10^{-4}	2 1 x 10^{-1}	1 2 x 10^{-3}	4 2 x 10^{-1}	2 3 x 10^{-3}
Accumulated Total				1 0 x 10^{-1}	8 7 x 10^{-4}	3 0 x 10^{-1}	2 6 x 10^{-3}	5 x 10^{-1}	4 5 x 10^{-3}	1 0	8 7 x 10^{-3}

[a] Estimates of injuries and fatalities have been rounded to two significant figures

[b] Labor estimates are given in Table 10 3-1

[c] Lost-time injuries are defined in Reference 4

(4) American National Standards Institute, Method of Recording and Measuring Work Injury Experience ANSI 216 1, 1967

- 200 mR/hr at the external surface of the vehicle
- 10 mR/hr at any point 1.8 m (6 ft) from the vehicle
- 2 mR/hr at any normally occupied position in the vehicle.

DOT regulations[5] further require the absence of significant removable surface radioactive contamination on the external accessible surfaces of packages when they are shipped. Levels of removable contamination on the surfaces are determined by a wipe test. The regulations state that the level is "not significant" if the radioactivity on the wipe does not exceed 10^{-11} Ci/cm^2 for beta-gamma emitters and 10^{-12} Ci/cm^2 for alpha emitters.[5]

The method used to estimate routine radiation doses from truck and rail transport of radioactive material is based on the method given in WASH-1238.[6] In addition, the following assumptions are made:

1) Two truck drivers during a 800 km (500 mile) trip would probably spend no more than 12 hr inside the cab and 1 hr outside the cab at an average distance of about 2 m (6 ft) from the truck.

2) Normal truck servicing enroute would require that two garagemen spend no more than 10 min about 2 m (6 ft) from a shipment.

3) Onlookers from the general public might be exposed to radiation when a truck stops for fuel or for the drivers to eat. The onlooker dose is calculated on the basis that 10 people spend an average of 3 min each at a distance of about 2 m (6 ft) from a shipment.

4) The cumulative dose to the general public from truck shipments is based on population dose of 1.2×10^{-5} man-rem per km.[6]

5) Train brakemen during the 2,400 km (1,500 mi) trip would probably spend 10 min in the vicinity of a shipping cask car during each stop. Assume an exposure rate of 25 mrem/hr at an average distance of about 2 m (3 ft) from the cask car. Assume two brakemen at each stop and a stop every 100 miles.

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(5) U.S. Code of Federal Regulations. Title 49, Parts 170-189, "Transportation." Superintendent of Documents, GPO, Washington, DC, 20555. January 1977.

(6) Directorate of Regulatory Standards, Environmental Safety of Transportation of Radioactive Materials to and From Nuclear Power Plants. WASH-1238, U.S. Atomic Energy Commission, Washington, DC, 1972.

6) The onlooker dose for train shipments is based on an onlooker population of 10 people who each spend 3 min at an average distance of about 2 m (6 ft) from a shipment.

7) The cumulative dose to the general public from rail shipments is based on a population dose of 1.2×10^{-5} man-rem/km.[6]

The estimated routine radiation doses from rail transport of the reactor fuel from the final operating core are listed in Table 11.4-1. The dose estimates are based on the maximum allowable dose rates for each shipment, assuming that 28 casks are needed to ship the fuel, and that each train transports only one cask. The estimated routine radiation dose for 28 trips of 2,400 km (1,500 mi) is 3.5 man-rem to the train brakemen and 0.9 man-rem total to the public along the shipping route.

TABLE 11.4-1. Estimated Accumulated Radiation Dose From Rail Transport of Spent Fuel

Group	Radiation Dose Per Shipment,[a] (Man-rem)	Total Radiation Dose (Man-rem)
Brakemen	0.12	3.5
Total Occupational		3.5
Onlookers	0.00025	0.07
General Public	0.029	0.8
Total Public		0.9

[a] Based on 28 shipments of one cask each for a distance of 2,400 km (1,500 mi).

The estimated routine radiation doses from truck transport of radioactive wastes from Immediate Dismantlement and from preparations for Safe Storage are listed in Table 11.4-2. These radiation dose estimates are based on the maximum allowable dose rates for each shipment in exclusive-use trucks,

[6] Directorate of Regulatory Standards, Environmental Safety of Transportation of Radioactive Materials to and From Nuclear Power Plants. WASH-1238, U.S. Atomic Energy Commission, Washington, DC, 1972.

TABLE 11.4-2. Estimated Accumulated Radiation Dose From Truck Transport of Radioactive Wastes

Mode	Group	Radiation Dose per Shipment,[b] man-rem	Total Radiation Dose per Mode, man-rem
Immediate[a] Dismantlement	Truck Drivers	0 07	95
	Garagemen	0 003	4
			99 Total Worker Dose
	Onlookers	0 005	7
	General Public	0 01	14
			21 Total Public Dose
Preparation for[a] Safe Storage	Truck Drivers	0 07	10
	Garagemen	0 003	0 4
			10 4 Total Worker Dose
	Onlookers	0 005	0 7
	General Public	0 01	1 4
			2 1 Total Public Dose

(a) Total Shipments 1,363 for dismantlement, 139 for Safe Storage
(b) Number based on 800 km (500 miles) per trip, one way

and are thus conservatively high. Table 11.4-2 is further based on the number truck shipments and shipping distances listed in Section 10 and in Appendices G and H. The estimated external radiation dose for routine transportation operations for Immediate Dismantlement is 99 man-rem to transport workers and 21 man-rem to the general public. For preparations for Safe Storage, the radiation dose is about 10 times less, or 10 man-rem to workers and 2 man-rem to the public.

11.4.2 Radiological Safety Evaluation of Postulated 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 such as 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 transport 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 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 are study based on accident data supplied by the U.S. Department of Transportation.[6] Accidents are classified by severity into five 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 11.4-3.

TABLE 11.4-3. Transportation Accident Severity Categories

Severity	Vehicle Speed, mph	Fire Duration, hr	Probability Per Vehicle Mile: Rail	Probability Per Vehicle Mile: Truck
Minor	0-30	<1/2	6×10^{-9}	6×10^{-9}
	0-30	0	4.7×10^{-7}	4×10^{-7}
	30-50	0	2.6×10^{-7}	9×10^{-7}
		Total	7.4×10^{-7}	1.3×10^{-6}
Moderate	0-30	1/2-1	9.3×10^{-10}	5×10^{-11}
	30-50	<1/2	3.3×10^{-9}	1×10^{-8}
	50-70	<1/2	9.9×10^{-10}	5×10^{-9}
	50-70	0	7.5×10^{-8}	3×10^{-7}
		Total	8.0×10^{-8}	3.1×10^{-7}
Severe	0-30	>1	7.0×10^{-11}	5×10^{-12}
	30-50	>1	3.9×10^{-11}	1×10^{-11}
	30-50	1/2-1	5.1×10^{-10}	6×10^{-12}
	50-70	1/2-1	1.5×10^{-10}	6×10^{-12}
	>70	<1/2	1×10^{-11}	1×10^{-10}
	>70	0	8×10^{-10}	1×10^{-10}
		Total	1.5×10^{-9}	8.2×10^{-9}
Extra Severe	50-70	>1	1.1×10^{-11}	6×10^{-13}
	>70	1/2-1	1.6×10^{-12}	2×10^{-13}
		Total	1.3×10^{-11}	8×10^{-13}
Extreme	>70	1	1.2×10^{-13}	2×10^{-14}
		Total	1.2×10^{-13}	2×10^{-14}

(6) Directorate of Regulatory Standards, Environmental Safety of Transportation of Radioactive Materials to and From Nuclear Power Plants. WASH-1238, U.S. Atomic Energy Commission, Washington, DC, 1972.

Estimated accident frequencies, release amounts and radiation doses to the maximum exposed individual for selected accidents involving rail transport of spent fuel and truck transport of radioactive waste materials are shown in Table 11.4-4. These frequencies are calculated by multiplying the total kilometers of radioactive material transport for each decommissioning mode times the total probability of accident per distance traveled for each accident severity class considered for rail or truck shown in Table 11.4-3.

The maximum exposed individual is assumed to be located 100 meters from the point of a transportation accident. The calculated dose values in Table 11.4-4 are for the first year dose and fifty year dose commitment to the bone and lung.

Each rail cask is assumed to contain 7 spent fuel elements, and each element is assumed to contain about 2×10^6 Ci of radioactivity.[6] Considering the nature of irradiated fuel, the low probability of a fuel cask being involved in a rail accident, the packaging design and requirements, and the controls placed on the transport operation, it is concluded that the risk of an accidental release of radioactivity from a rail cask is extremely small. It is beyond the scope of this study to attempt an in-depth analysis of these low probability low release level events. Thus, no dose calculations were required for the rail accidents listed in Table 11.4-4, and no other low probability events were considered.

The average radionuclide inventory per truck shipment is assumed to be 100 Ci of dispersible radioactive material, with an average shipping distance of about 800 km (500 miles). The waste inventory is assumed to be characterized by the activated corrosion products listed in Table 7.3-7.

11.4-3. Nonradiological Transportation Safety Evaluation

The number of truck shipments for transporting chemicals during decommissioning is minute compared to the total U.S. truck shipments. Therefore, the impact of truck accidents involving shipments of chemical effluents is believed to be negligible.

TABLE 11.4-4. Estimated Frequencies and Radioactivity Releases for Selected Rail and Truck Transport

Accident Description	Frequency of Accidents[b] per Dismantlement	Frequency of Accidents[b] per Safe Storage	Release, Curies	Radiation Dose for Maximum Individual, (rem)[a]: 1st Year Dose, Bone	1st Year Dose, Lung	50 Yr. Dose Commitment, Bone	50 Yr. Dose Commitment, Lung
Rail Transport of Spent Fuel[c]							
Moderate severity accident with cask	3.4×10^{-3}	3.4×10^{-3}	No release	-[d]	-	-	-
Severe accident with cask	6.3×10^{-5}	6.3×10^{-5}	No release	-	-	-	-
Truck Transport of Decommissioning Wastes[e,f]							
Minor accident with closed van	8.8×10^{-1}	9.0×10^{-2}	No release	-	-	-	-
Moderate accident with closed van	2.1×10^{-1}	2.1×10^{-2}	1×10^{-4}	0.01	0.2	0.01	0.2
Severe accident with closed van	5.6×10^{-3}	5.7×10^{-4}	1×10^{-2}	1.1	21	1.1	24

(a) Maximum exposed individual is assumed at 100 meters from the site of the accident.

(b) Based on accident probabilities given in Reference 6, and shown in Table 11.4-3 for various accident severity classes.

(c) Rail shipment of spent fuel occurs for both decommissioning modes considered. The number of trips is the same for each, so the accident frequencies are the same.

(d) A dash indicates no calculation was made.

(e) Based on an inventory of 100 Ci per truck shipment.

(f) Release fraction for respirable material for moderate and severe accidents are assumed to be 10^{-6} and 10^{-4} respectively.

As for any transport activity, a certain potential exists for injury or death from decommissioning transport operations.[6] Table 11.4-5 lists estimates for injuries and fatalities for transportation activities associated with Immediate Dismantlement and preparations for Safe Storage. The number of injuries and fatalities for each decommissioning mode are calculated by multiplying the round-trip distance traveled, times the probability of accidents per vehicle kilometer, times the injuries or fatalities expected per accident.

As shown in Table 11.4-5, there are about 0.03 injuries and 0.002 fatalities estimated for transporting 28 shipments of fuel from the reactor assuming a single cask car per shipment and a round-trip distance of about 4,800 km (3,000 mi). About 1.2 injuries and 0.067 fatalities are estimated to occur for Immediate Dismantlement during the 1,363 round trips by trucks to the burial ground. Each round trip is assumed to be about 1,600 km (1,000 mi). Since preparations for Safe Storage require about one tenth as many round-trips, only 139 total, about one tenth as many injuries and fatalities are estimated to occur.

[6] Directorate of Regulatory Standards, Environmental Safety of Transportation of Radioactive Materials to and From Nuclear Power Plants. WASH-1238, U.S. Atomic Energy Commission, Washington, DC, 1972.

TABLE 11.4-5. Estimated Injuries and Fatalities from Decommissioning Transportation Accidents[a]

Transportation Operation	Probability (Accidents per Vehicle Kilometer)	Injuries per Accident	Fatalities per Accident	Kilometers/Operation (Round-Trips)	Estimated Nonradiation Transportation Accidents[c] Injuries	Fatalities
Rail[b]						
Spent Fuel Shipments	8.7×10^{-8}	2.7	0.2	1.4×10^{5}	0.033	0.0024
Truck[d]						
Immediate Dismantlement	1.0×10^{-6}	0.51	0.03	2.2×10^{6}	1.1	0.066
Preparations for Safe Storage	1.0×10^{-6}	0.51	0.03	2.2×10^{5}	0.11	0.0067

(a) Accident frequencies are from Appendix C, Table 1 of Reference 6.

(b) Assuming one spent fuel cask per train, and 4800 km (3000 mi) round trip distance.

(c) Estimates of injuries and fatalities are rounded to two significant figures.

(d) Assuming truck transport of 1600 km (1000 mi) round trip to the burial ground, and 1363 trips for Immediate Dismantlement, and 139 trips for Preparations for Safe Storage.

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

2. Code of Federal Regulations. Title 10, Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Practicable' for Radioactive Materials in Light-Water-Cooled Nuclear Power Reactor Effluents". Superintendent of Documents, GPO, Washington, DC, 1977.

3. Operational Accidents and Radiation Exposures Experienced Within the USAEC 1943-1970. WASH-1192, 1971.

4. American National Standards Institute, Method of Recording and Measuring Work Injury Experience. ANSI 216.1, 1967.

5. U.S. Code of Federal Regulations. Title 49, Parts 170-189, "Transportation." Superintendent of Documents, GPO, Washington, DC, 20555, January 1977.

6. Directorate of Regulatory Standards, Environmental Safety of Transportation of Radioactive Materials to and From Nuclear Power Plants. WASH-1238, U.S. Atomic Energy Commission, Washington, DC, 1972.

12.0 COMPARISONS WITH OTHER STUDIES

Six other studies on decommissioning of large commercial nuclear power stations have appeared in the literature since this study was initiated. The earliest study was an in-house analysis by the Vereinigung Deutscher Elektrizitätswerke (VDEW), which appeared in summary form in German in June 1976,[1] with an English translation later that year. The next study to appear was also from Germany, carried out by Nuklear-Ingenieur-Service GmbH (NIS) for the Commission of the European Communities, which was published in English in November 1976.[2] The third study was by Nuclear Energy Services Division of Automation Industries, Inc. for the Atomic Industrial forum[3] and was also dated November 1976. The fourth study was a site-specific analysis for San Onofre 1, done by NUS Corporation for Southern California Edison Company.[4] The fifth and sixth studies are United Kingdom reports, one by Associated Nuclear Services for the Nuclear Installations Inspectorate, emphasizing decommissioning of a Magnox station[5], and the other is an analysis by the Nuclear Power Development Establishment for the United Kingdom Atomic Energy Authority of the feasibility of decommissioning the Windscale Advanced Gas-cooled Reactor.[6] A seventh study has recently become available in the form of Appendices to an application for a site certificate. This study is a site-specific analysis of the Pebble Springs Nuclear Plant by Nuclear Energy Services, Inc. for Portland General Electric Company.[7]

Each of these studies was undertaken with a variety of motives in mind, and the conclusions reported tend to reflect the particular interests of the study sponsor and the purpose for which the study was intended to be used. Each of these studies is discussed briefly in subsequent sections and the motivations are indicated, where known. Some discussion of results from these studies and some comparisons with results from this (NRC-BNW) study are given.

12.1 REFERENCE 1: VDEW STUDY

D. Brosche and J. Essman, "On the Decommissioning of Nuclear Power Stations," Atom and Strom, Volume 22, No. 3, pp. 81-87, (May/June 1976).

This paper is a summary of an internal report (which is in German and is not readily available) titled "Study on the Decommissioning of Nuclear Power Stations," published in May 1975. The principal motivation for the study was to be sure that current-generation nuclear stations met the requirements of Criterion 2.10 of the Safety Criterion for nuclear power stations, established by the Federal Government of Germany in 1974: "Nuclear power stations must be constructed in such a way that they can be decommissioned with continued observance of the radiation safety regulations. A plan must be made for the dismantling of the power station after final decommissioning, which will comply with the radiation safety regulations." The study examined primarily light-water reactors (LWR) but special problems related to High Temperature Reactors (HTR) were also examined. The purpose of the study was to show that Criterion 2.10 could be satisfied using present-day technology on present designs. Also, the economics of decommissioning were considered, since these costs must be included in the power generation costs and reserves generated to pay for decommissioning.

An analysis was made to estimate the inventory of radioactive materials that would be present after 40 years of operation at 80% load factor. Detailed assessments of work plans were made for two decommissioning alternatives: Dismantlement, and Safe Storage. Decontamination methods, dismantlement equipment and techniques were examined, as well as transport and storage of radioactive materials. Unfortunately, none of the details of these analyses are available in the summary published in the open literature, only final totals. Thus, a detailed comparison of the assumptions and conditions underlying these analyses with the assumptions and conditions of other analyses is not possible. However, a few important points can be identified. The inventory of radioactive materials is based on extrapolations from a smaller (350 MWe PWR) reactor, a procedure that can lead to significant overestimation of the inventory. Also, all radioactive components and materials were reduced to fit within standard drums for eventual disposal. Thus, the cutting and packaging operations would require a significant level of effort.

12.2 REFERENCE 2: ECC-NIS STUDY

R. Bardtenschlager, D. Bottger, A. Gasch and N. Majohr, "Decommissioning of Light Water Reactor Nuclear Power Plants," Nuclear Engineering and Design, Vol. 45, pp. 1-51, North-Holland Publishing Company, (1978).

The published abstract of this paper follows:

"This study deals with the technical and economic questions posed by the decommissioning of light-water reactor nuclear power plants of the 900-1300 MWe class, account being taken of the distinctions between boiling- and pressurized-water reactors. Possible decommissioning alternatives and the disposal or confinement of activity are discussed. It emerges from the discussion that decommissioning, and even total dismantlement of these nuclear power plants is in principle feasible.

The activity inventory, one year after shutdown, is calculated to be about 3 x 10^7 Ci for the BWR and 4 x 10^6 Ci for the PWR; 40 years after shutdown these figures are reduced to 2 x 10^6 and 4 x 10^5 Ci, respectively.

The decommissioning costs to be expected are also estimated. This estimate serves as the basis for an economic comparison by the present worth method. The economic comparison shows that total dismantlement after a cooling time of one year is more than four times as expensive as interim confinement followed by total dismantlement waiting period of 40 years. The present worths for immediate total dismantlement are estimated DM 200 million for the BWR and DM 170 million for the PWR; for the other alternative, they are put at DM 45 million for the BWR and DM 42 million for the PWR.

A still open question is posed by the final storage of the large quantities of bulky radioactive waste arising in partial or total dismantlement. Since no decision on the storage method has yet been taken, disposal in casks is stipulated as a boundary condition in the estimation of the costs, although this is an unrealistic assumption. It is to be presumed that the costs of disposal can be reduced given appropriate final storage."

This paper presents detailed bases for the analyses and evaluations made. Sequences of operations and costs for performance are developed for three approaches to decommissioning: Immediate Dismantlement, Partial Dismantlement with Safe Storage, and Safe Storage with Deferred Dismantlement. In general, upper boundary conditions are used in estimating the costs. A detailed calculation was performed to estimate the inventory of neutron-activated material, modeled after a 1200 MWe PWR (Biblis A). Radiation dose rates from the activated reactor vessel and internal components were also estimated, using standard calculational methods. All radioactive materials were assumed to be cut into pieces sufficiently small to be packaged in 200 or 400 liter (55 or 110 gallon) drums. Thus, the packaging operations would require a significant level of effort.

12.3 REFERENCE 3: AIF-NES STUDY

W. J. Manion and T. S. LaGuardia, An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternatives, AIF/NESP-009, Atomic Industrial Forum, Inc., November 1976.

This study reports detailed analyses of the costs, occupational radiation exposure and radioactive material volumes for disposal that result from the decommissioning of three generic reactor types: PWR, BWR, and HTGR. Three basic approaches to decommissioning of reactors were examined: Immediate Dismantlement; Hardened Safe Storage (entombment); and Custodial Safe Storage (mothballing). The latter two approaches were also examined when terminated by Deferred Dismantlement.

Since the reactors studied were generic rather than specific, design and site-specific details could not be treated fully. Rather, those items that were likely to be significantly influenced by design and site differences were identified for future consideration.

Detailed work descriptions for the tasks necessary to accomplish the decommissioning were developed. From these descriptions, manpower, occupational radiation exposure, and radioactive material disposal volumes were estimated.

Calculations were made to estimate the inventories of radionuclides that would be present at the reactor facilities during decommissioning. Based on these inventories, radiation dose rate estimates were made for activiated components from the reactor vessel, both at shutdown and after various time periods thereafter. The conclusion from this analysis was that some of the vessel components would remain sufficiently radioactive to preclude permanent entombment.

Estimates of airborne radionuclide releases to the environment resulting from decommissioning operations were also made, together with estimates of the radiation doses to the public resulting from transport of radioactive materials to disposal sites.

12.4 REFERENCE 4: SCE-NUS STUDY

R. J. Stouky and E. J. Ricer, San Onofre Nuclear Generating Station Decommissioning Alternatives, Report 1851, NUS Corporation, February 22, 1977.

This study reports the analyses for decommissioning San Onofre 1. Two basic approaches were examined with a variation on one approach. These were: Immediate Dismantlement; and Safe Storage with Deferred Dismantlement. In the latter case, the variation considered was on-site storage of spent reactor fuel until Deferred Dismantlement took place.

The emphasis in this study was a detailed examination of costs, with lists of Federal Power Commission (FPC) plant account cost classifications for equipment and estimates of removal costs for each item.

A thorough chemical decontamination of all fluid systems is assumed, to reduce radiation dose rates everywhere (except in relation to the activated reactor vessel) to less than 10 mR/hr. No detailed analyses and no estimates are given for occupational radiation exposure or for exposure to the public resulting from decommissioning operations. Also, little detail of the development of disposal costs for radioactive materials is given, just the assumed unit costs and the final totals.

A basic premise of this study was that all structures had to be removed from the site including the highly reinforced base mats for the structures, the tsunami wall, the site paving and the rail spur. The site was restored to the status of a public beach area, with totally unrestricted usage by the public. Also, the site was assumed to be vacated by the year 2023 in accordance with the terms of the property lease. These requirements placed time constraints and cost penalties on the decommissioning program that resulted in increased total costs.

12.5 REFERENCE 5: ANSI-NII MAGNOX STUDY

A. Martin, D. T. Read, R. W. Milligan, T. F. Kempe, and D. A. Briaris, A Preliminary Study of the Decommissioning of Nuclear Reactor Installations. ANS Report No. 155, Associated Nuclear Services, July 1977.

This study presents a brief analysis of the decommissioning of a Magnox-type gas-cooled reactor. No cost data or occupational radiation exposure data or material disposal data are presented. Roughly one half of the report is devoted to a listing of relevant decommissioning experience, largely in the United States, and reviews of other decommissioning studies, thus providing a good background review of decommissioning efforts in the past.

12.6 REFERENCE 6: UKAEA WINDSCALE AGR STUDY

K. Saddington, The Decommissioning of WAGR--A Feasibility Study, ND-R-20(W), Nuclear Power Development Establishment, United Kingdom Atomic Energy Authority, July 1977.

This report focuses on the task of decommissioning the Windscale Advanced Gas-cooled Reactor. As in the case of Reference 5, the design-specific aspects of the study make it difficult to find any points of comparison with PWRs. The study conclusions were that the WAGR could be dismantled and removed from the site successfully following shutdown but that to reduce occupational exposure significantly, it would be preferable to delay dismantlement of the reactor building for an extended period of time, up to 100 years.

12.7 REFERENCE 7: PGE-NES PEBBLE SPRINGS PWR STUDY

T. S. LaGuardia and R. A. Calabrese, Appendices I.1 and I.2, Pebble Springs Site Certificate Application. Nuclear Energy Services, Inc., for Portland General Electric Co., January 23, 1978.

This report was prepared as an appendix to an application for a site certificate, submitted to the Oregon Energy Facility Siting Council, to demonstrate that decommissioning can be accomplished at the proper time with no unreasonable environmental impact and in a manner that restores the site to an unrestricted use status. The study is a design and site-specific application of the methodology developed in the AIF-NES Generic Study (Reference 3). Costs have been escalated to late 1977 levels. Adjustments in volumes of material to be handled and disposed of are made to reflect the specific design under consideration.

12.8 DISCUSSION OF STUDY RESULTS

The first point that becomes very apparent is that it is virtually impossible to make any detailed comparisons among the various studies. Each study gives a total cost for Immediate Dismantlement, but the development of the cost segments making up that total varies markedly among the studies, so that the segments cannot be examined on a common basis.

One point that can be compared among the studies is the size of the radioactive inventory as shown in Table 12-1. The estimated curie inventory is quite sensitive to many parameters in the calculation. The postulated initial cobalt content in the stainless steel reactor vessel components has a direct bearing on the resulting inventory. The magnitude and shape of the thermal neutron flux near the core barrel region directly affects the total number of curies from neutron-activated materials. The neutron cross sections for production and removal and the decay rates of the activated radionuclides also influence the final result. However, the impact of differences in total curie inventories is not very large, since much of the material is so radioactive that a factor of 2 or even 10 doesn't affect the basic work procedures employed to remove the material. The principal impacts would be in the cost of shielding for shipment and for curie surcharges at the burial site.

TABLE 12-1. Comparison of Estimated Inventories of Radioactivity in a Shutdown PWR

Study	Estimated Inventory (Curie)	Time After Shutdown (yrs)	Basis of Estimate
VOEW	1.3×10^7	1	Extrapolation from calculation for small reactor
ECC-NIS	4.6×10^6	1	Detailed calculation on Biblis-type reactor
AIF-NES	1.5×10^7	0	Detailed calculation on generic 1,160 MWe PWR
SCE-NUS	No estimate	--	----
NRC-BNW	4.8×10^6	0	Detailed calculations on reference TROJAN PWR

Estimates of occupational radiation exposure were made in the AIF-NES study (630 man-rem) and the NRC-BNW study (1408 man-rem) for Immediate Dismantlement. Both of these values are based on fairly detailed estimates of man hours and radiation dose rates. The differences reflect different levels of optimism about the effectiveness of chemical decontamination efforts, and different operational approaches to some dismantlement tasks, and different time frames for performing some of the work.

The total estimated cost for Immediate Dismantlement is given for each of the studies, based on estimates made in different years. These estimates are given in Table 12-2, together with suggested escalation factors to put the estimates on approximately the same time-cost estimate basis.

TABLE 12-2. Estimated Costs for Immediate Dismantlement of a Large PWR

Study	Reported Cost (Millions $)	Year of Estimate	Suggested Escalation Factor	Costs Estimated In Millions of 1978 Dollars
VDEW	44	1975	1.29	57
ECC-NIS	79	1976	1.17	88
AIF-NES	34	1975	1.29	43
SCE-NWS	51	1976	1.11	57
NRC-BNW	43	1978	1.0	43

[a] All costs are rounded to two significant figures.

It is seen in Table 12-2 that the estimated costs for Immediate Dismantlement differ by as much as a factor of 2. It is not possible to extract from the published ECC-NIS study the detailed reasons why their costs are significantly higher than the estimates made in the United States. Certainly, the more massive containment structures and the requirements for packaging all materials in relatively small drums contributed to increasing the estimated costs in the two German studies. The removal of everything from the site, including subsurface structures, increased the estimated costs of the NUS study significantly.

Several conclusions can be drawn from all of these studies. The first conclusion is that there are no major technical impediments to the successful decommissioning of a large PWR. The job can be done, using currently available technology, within the framework of present regulations, with virtually no impact on the safety of the general public.

The second conclusion is that the cost of disposal for radioactive materials is a significant fraction of the total decommissioning cost. Efforts to develop facility designs and decontamination techniques that minimize the quantities of contaminated material that must be disposed of as radioactive waste will pay large dividends in reducing overall decommissioning costs.

The third conclusion is that to develop a realistic estimate of the costs of decommissioning a reactor facility, it is necessary to perform a quite detailed analysis on the specific plant under consideration. Design differences among plants can have significant impacts on the types and amount of work involved in accomplishing decommissioning. Until such time as facility designs are truly standardized, these plant-specific analyses will be necessary.

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7. T. S. LaGuardia and R. A. Calabrese, Appendices I.1 and I.2, Pebble Springs Site Certificate Application. Nuclear Energy Services, Inc., for Portland General Electric Co., January 23, 1978.

13.0 DESIGN CONSIDERATIONS FOR THE FACILITATION OF DECOMMISSIONING

The present study on immediate dismantlement and safe storage of a pressurized water reactor describes activities that can be used to conceptually decommission a reference PWR. With this study as a basis, insights have been gained as to plant design characteristics that could simplify the task of decommissioning. This section summarizes some of these potential plant and equipment design features.

It is recognized that some of the ideas suggested to facilitate decommissioning may not be compatible with those plant characteristics that are desirable for normal plant operation. Some of the ideas may also be prohibitively expensive to implement. However, the purpose of this section is to identify design characteristics that could expedite and simplify the decommissioning task without much concern for the practicalities of the suggestions. Such evaluations are beyond the scope of this study. The ideal structural or equipment design change is one that would reduce or prevent personnel exposure, is relatively inexpensive, and one that assures an adequate margin of safety during inspection, maintenance, operation, safe storage and/or dismantlement.

Title 10 Code of Federal Regulations, Part 50, Appendix F.4 describes the Nuclear Regulatory Commission's position regarding facilitation of decommissioning of fuel reprocessing plants: "A design objective shall be to facilitate decontamination and removal of all significant radioactive wastes at the time the facility is permanently decommissioned." Application of this NRC objective to nuclear power reactor facilities is a logical extension of the intention of this regulation. Additionally, NRC Regulatory Guide 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will be as Low as Is Reasonably Achievable (ALARA), explicitly points out that "design concepts and station features should reflect consideration of the activities of station personnel (including decontamination and decommissioning) that might be anticipated."

Consequently, early consideration should be given to equipment design and location, accessibility requirements, and shielding requirements. It should be noted that the time honored methods to overcome increased radiation levels during reactor plant life, i.e., remote operations and more shielding, usually translate into increased costs. Presently, the shielding groups of architect-engineer (A/E) firms are actively involved in developing design methods for providing shield systems that serve two functions: 1) to meet the radiological criteria set by the NRC regulations; and 2) to facilitate inspection and maintenance, and to increase overall plant availability during the plant's operating lifetime. A third function must now be considered: to facilitate decommissioning.

An important input to the planning (design considerations) of a new nuclear power plant should be relevant experience from existing operational nuclear power plants. The general criteria for selecting design features for consideration should be the beneficial effect they might have in decreasing decommissioning cost, improving occupational or public safety, reducing total decommissioning time, creating less volume of radioactive waste, and the improved ease of performing the decommissioning.

13.1 CONCEPTUAL DESIGN CONSIDERATIONS

Design insights gained during the course of this study are aimed only at areas that present obvious complexity or difficulty to decommissioning a PWR; they are not all inclusive, and do not consider details, side effects, or variations of the alternatives. Such an analysis would require an in-depth study beyond the scope of this report. Basically, design considerations should include:

- Design of plant components to suit their function; this includes careful selection of material, good decontamination features, clear design, and ease of dismantlement.
- Systematic arrangement of components (short pipelines and transport paths).
- Analysis of the facility design and the conceptual layouts; the analysis should assess the probable types, quantities and locations of residual radioactive materials in the reactor systems at final shutdown; the analysis should include decontamination methods required for decommissioning and the hazards associated with implementation of those decontamination methods.

- Equipment design should provide for decontamination and equipment removal, and should as a minimum, include design features necessary to accomplish this; e.g., SFP storage racks which are to be removed from the pool during the immediate dismantlement mode option should be designed so that they can be removed with a minimum of direct exposure to decommissioning personnel.

- Equipment design should provide for either removal of the equipment itself or its in situ decontamination, this should include provisions for filling and draining decontamination solutions.

The high costs of past reactor decommissionings relative to nonnuclear power plants are usually attributed to the following reasons: 1) nuclear plants generally have more massive structures, usually of reinforced concrete; 2) packaging, transporting and storing or burying radioactive wastes can be a major cost item; 3) the need for remote operations, contamination control, and maintaining a radiological surveillance and protection system add significantly to the cost; and 4) manpower use under radiation zone conditions is less efficient.[1] Therefore, innovative ideas for improvements in the above four areas could offer considerable incentive for developing improved, specialized decommissioning technology. Design considerations worthy of attention as well as some possible solutions include those listed in Table 13-1. These findings regarding desirable features for decommissioning are presented with no attempt to rank their relative importance.

[1] K. M. Harmon, et al., "Decommissioning Nuclear Facilities." Proceedings of the International Symposium on the Management of Wastes from the LWR Fuel Cycle, Denver, CO, July 11-16, 1976.

TABLE 13-1. PWR Design Considerations and Innovative Ideas Related to Improvements in Decommissioning Technology

	Design Consideration		Possible Solution (Idea)[a]
1	Elimination of difficulties in decontaminating the internals of pipes and tanks	1	Build in decontamination spray systems or access ports for their eventual use
2	Elimination of difficulties in decontaminating and eventual demolition of concrete surfaces	2	A double layer' construction concept for concrete surfaces (see Figure 13-1 for conceptual design)
3	Reclamation, reuse, and/or recycling of valuable materials (both during operation and decommissioning)	3	A/E provide space for inclusion of a commercial-size electropolishing unit in most advantageous facility location
4	Reduce personnel radiation exposure while changing radio active filters and ion exchange units	4	Use of gatling gun device (see Figure 13-2 for conceptual design)
5	Development of remote operations	5	Improvements in high pressure remote quick disconnect fittings for easier, quicker operation (R&D required)
6	Improvement of waste packaging containers	6	Acceptable industry-wide standardization of size, design and material of construction of LSA containers would enable cost reductions
7	Consideration should be given to reducing the number of building penetrations to the outside environment thus minimizing the problems associated with the structural closing and sealing of these penetrations at the time of decommissioning	7	The solution may range from complete elimination of windows to radical structural designs such as underground buildings
8	Reactor Coolant System internal pipe treatment	8	Application during the manufacturing process via space age 'sputtering' process of pipe internals with a thin layer of erosion-corrosion resistant material (R&D required)
9	More cost-effective method to remove surface layers of bioshield concrete	9	Build in and cap or plug plastic lined bore holes in appropriate locations and sufficient numbers to reduce this job to a truly conventional demolition status Built-in provisions for other techniques, such as spalling of the concrete by heat or electric current, might also be employed
10	Accessibility of equipment for ease of dismantlement	10	Take down walls and removeable roofs in locations where eventual equipment removal needs might dictate, in many cases it is easier to build it into the original structure than to knock it out later
11	Low cobalt and low niobium alloy steels for reactor vessels and reactor internals	11	Expertise from industry contacts is recommended to determine possible methods, needs, and economics involved (cost-benefit), exhaustive analysis (R&D required) to determine any other trace metal impurities that could produce radioactive contaminants having an impact on decommissioning
12	Biological shield	12	Development of optimum barrier characteristics based on interlocking building block design concepts, perhaps encased in secondary structural metal frame which lies outside of potential radiation activation zone for ease of dismantlement
13	HVAC Systems	13	Incorporation of devices (unspecified) for purposes of either temporary or permanent isolation including capabilities for various disassembly modes and including design analysis of varying air flow conditions expected during disassembly and dismantlement
14	Equipment for reduction of volume of radioactive wastes	14	Digestion, incineration, and/or compaction systems for combustible wastes
15	*Replacement equipment in operating nuclear power plants*	15	*Incorporation of design considerations to facilitate decommissioning* should become an integral part of all specifications for replacement equipment in aging nuclear power plants The dual objective of replacement equipment should reflect modifications based on the experience gained from using the original equipment plus the design objectives regarding future decommissioning of that equipment
16	Radiation exposure reduction	16	Radical building design alternative in which a shielded working platform on railroad tracks has access from above to all compartments in buildings containing radioactive equipment Substantial radiation sources could thus be decontaminated, dismantled, removed to the shielded platform, and transported to a disassembly/electropolishing station at a greatly reduced cost in terms of dollars and personnel exposure Such a design would pay dividends during the operational lifetime of the PWR as well to servicing and maintenance personnel

[a] No cost-benefit analyses have been made. Such analyses could be expected to require participation by and good engineering judgment of the nuclear steam supply system (NSSS) vendor, the designer, the architect-engineer (A/E), the constructor, and the operator of the nuclear power facility on a case-by-case basis

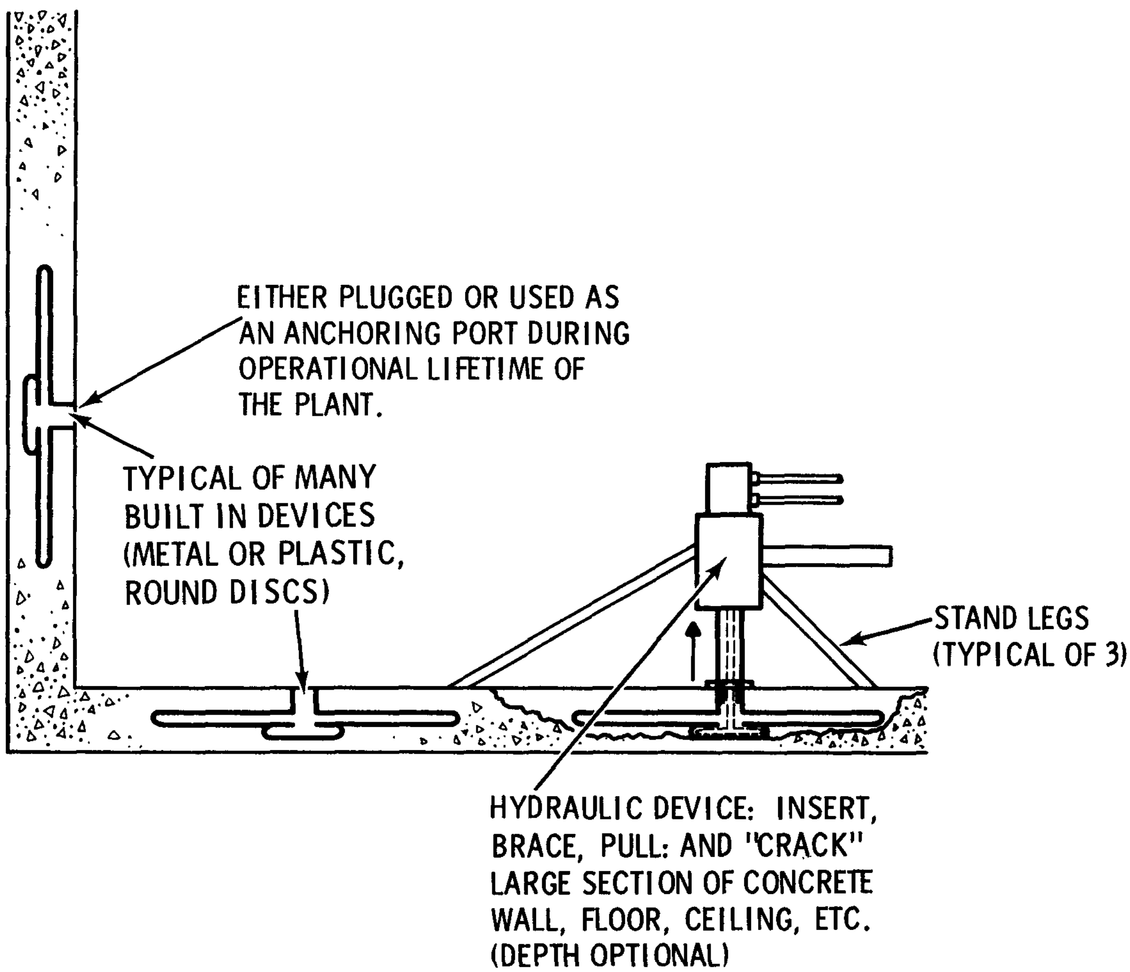
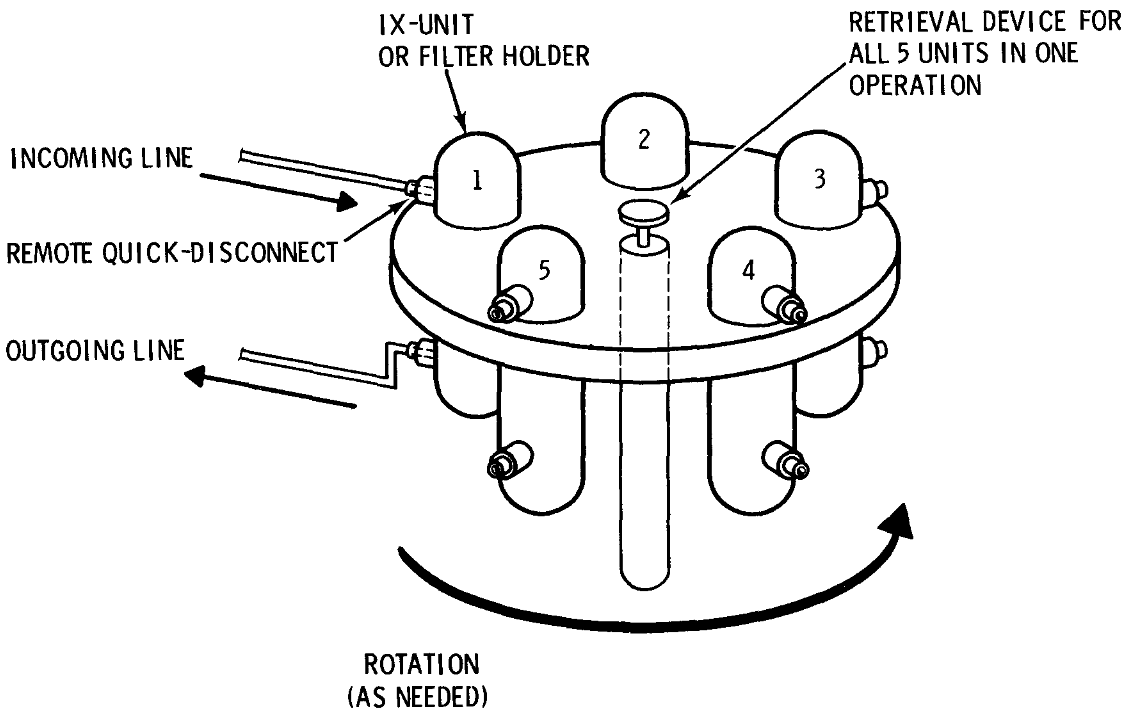


FIGURE 13-1. Conceptual Design for Concrete Removal



NOTE: GATLING GUN DEVICE MAY BE USED FOR 1) A SINGLE SYSTEM; I.E. AS NUMBER 1 UNIT IS DEPLETED OR BREAKS THROUGH, SIMPLY DISCONNECT REMOTELY AND GO ON TO UNIT NUMBER 2 OR 2) MULTIPLE SYSTEMS; I.E. FIVE OR MORE SEPARATE SYSTEMS WHICH UTILIZE ONLY ONE SHIELDING CUBICLE.

ADVANTAGES:

- SMALLER SHIELDED AREA REQUIRED
- ONLY ONE REMOTE REMOVAL OPERATION REQUIRED INSTEAD OF ONE FOR EACH INDIVIDUAL UNIT
- RAPID INSTALLATION AND REMOVAL CAPABILITY

FIGURE 13-2. Conceptual Design for Gatling Gun Device for Use in PWR Exposure Reduction Program

13.2 DESIGN CONSIDERATIONS BY OTHERS FOR THE FACILITATION OF DECOMMISSIONING

The need for design features to facilitate the decommissioning of nuclear reactors is widely recognized.[2,3] The proposed changes range from increased emphasis on in-plant radiation control features, to more remotely controlled operations, to increased use of prefabricated parts for concrete structures for later ease of dismantlement. It is stated in a German study on decommissioning[4] that "no fundamental difficulties with the decommissioning are perceptible which could give a reason for changing the basic design of present nuclear power plants." They do, however, consider a number of identified difficulties in detail for the purpose of reducing the decommissioning workload, and hence the costs involved. These include the increased use of the above mentioned prefabricated concrete parts, spatial separation of contaminated systems from "clean" systems, and allowing sufficient room for decontamination operations where applicable. The authors recognize that these aspects, to a large extent, are already increasingly observed at modern nuclear power stations. This is particularly true with regard to the facilitation of repair work, which ultimately serves also to aid the decommissioning.

The nuclear power station owner/operator is placing much emphasis on design features that will improve the availability of the station; i.e. a reduction in the overall downtime of the station. In fact, utilities that purchase and operate nuclear power plants demand an overall attack on unsatisfactory plant

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(2) A. Martin, et al., A Preliminary Study of the Decommissioning of Nuclear Reactor Installations, ANS Report No. 155, p. 63, July 1977. Associated Nuclear Services, 123 High Street, Epsom, Surrey, KT19 8EB.

(3) G. F. Stone, et al., "Control of Occupational Radiation Exposures in TVA Nuclear Power Plants Design and Operating Philosophy," ANS-SD-15. Proceedings on the Special Session on Plant and Equipment Design Features for Radiation Protection. CONF-750661. New Orleans, LA, June 8-13, 1975.

(4) R. Bardtenschlager, et. al., Decommissioning of Light Water Nuclear Power Plants, Nuclear Engineering and Design, Vol. 45, pp. 1-51, Copyright North-Holland Publishing Company, 1978.

availability by improvement of technology and design procedures.[5]

In a 1977 report by the National Research Council/National Academy of Sciences Committee on Radioactive Waste Management,[6] it is indicated that more thought must be given to the design of the fabric or basic structure of the buildings housing future commercial power stations "in order that their useful life can be extended even though their internal operational parts are replaced." By designing so that radioactive parts of a system can be replaced, several important goals can be achieved:

- an extension of the useful lifetime of the power generating station, and
- facilitation of the eventual decommissioning of the station (an item designed to be replaced is more readily removable than one that was not so designed).

Prevention and/or reduction of radiation exposure to operating personnel is an early consideration in the design of a nuclear power station. Cost benefit evaluations of design features to reduce radiation exposure imply a value for the reduced radiation exposure; i.e. the cost of additional manpower that would otherwise have been required. The direct economic savings due to reducing radiation exposure can range from very large to nothing. If the maintenance and other radiation related duties at the station can be readily performed within the bounds of existing operating personnel radiation limits (10 CFR § 20.101), the economic saving due to reducing radiation exposure is nil. However, if the radiation dose for the same duties exceeds that permitted by regulation so that additional manpower have to be employed, then the reduction of radiation exposure may produce economic benefits.

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(5) Proceedings on the Special Session on Plant and Equipment Design Features for Radiation Protection, ANS-SD-15, CONF-750661, p. 3, New Orleans, LA, June 8-13, 1975.

(6) The Shallow Land Burial of Low-Level Radioactively Contaminated Solid Waste, National Academy of Sciences, Washington DC, 1976. Library of Congress Catalog Card Number 76-56928.

Additional information on the economics of and the factors affecting radiation exposure reduction are given in Reference 7.

Early application of radiation dose reduction designs yield benefits extending over the whole operating life of the plant, whereas the cost to achieve the reduction will be incurred initially over a relatively short period. In general, a design feature that reduces exposure to operating personnel can be expected to reduce radiation exposure to decommissioning personnel as well.

The use of remote-controlled manipulators mounted on radiocontrolled vehicles are reported to have met with some success in decommissioning work.[8] These units were developed for special work in high radiation fields. They are limited in their applicability only where there is a lack of working space for them. Either new devices must be developed to work in areas of concern during decommissioning or sufficient space must be provided (e.g. the removal of a wall or ceiling during decommissioning). Advantages of remotely operated equipment include exchangeable tools on the same vehicle for excavation work, cleanup work, and transport of highly active, heavy, or bulky objects.[9]

In conclusion, a very important aspect of the planning of a nuclear power station is in fact relevant experience from plants which are already in operation. Particular emphasis in the planning stages, to facilitate operation, will in the final analysis also facilitate decommissioning.

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(7) G. G. Legg, "Designing to Minimize Radiation Exposure," AECL-5521. A paper presented at the Session on Control of Reactor Radiation Fields, Canadian Nuclear Association 16th Annual Meeting, Toronto, Ontario, June 13-16, 1976.

(8) Decommissioning of Nuclear Facilities, IAEA-179, Vienna 1975. pp. 152-154.

(9) W. Krüger, Kerntechnischer Hilfszug, Gesellschaft für Kernforschung mbH, Karlsruhe, 1973.

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4. R. Bardtenschlager, et al., Decommissioning of Light Water Nuclear Power Plants, Nuclear Engineering and Design, Vol. 45, pp. 1-51, Copyright North-Holland Publishing Company, 1978.

5. Proceedings on the Special Session on Plant and Equipment Design Features for Radiation Protection, ANS-SD-15, CONF-750661, p. 3, New Orleans, LA, June 8-13, 1975.

6. The Shallow Land Burial of Low-Level Radioactively Contaminated Solid Waste, National Academy of Sciences, Wasington DC, 1976. Library of Congress Catalog Card Number 76-56928.

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8. Decommissioning of Nuclear Facilities, IAEA-179, Vienna 1975. pp. 152-154.

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14.0 GLOSSARY

Abbreviations, terms, definitions, and symbols directly related to decommissioning work and related technology are defined and explained in this section. It is divided into two parts with the first part containing the abbreviations and Greek letters, and the second part containing terms and definitions (including those used in a special sense for this work). Common terms covered adequately in standard dictionaries are not included.

14.1 GLOSSARY ABBREVIATIONS

AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable[a]
ATMX	Army Transport Mobile for Explosives[a]
CFR	Code of Federal Regulations[a]
Ci	Curie[a]
DF	Decontamination Factor[a]
DOT	Department of Transportation
DPM	Disintegrations per Minute[a]
EDTA	Ethylenediamine tetraacetic acid
FSAR	Final Safety Analysis Report
HEPA	High Efficiency Particulate Air (Filters)
HP	Health Physicist[a]
HVAC	Heating, Ventilation and Air Conditioning
LAW	Low Activity Waste
LSA	Low Specific Activity
LWR	Light Water Reactor
MDP	Master Decommissioning Plan
mrad	Millirad[a]
mr	Milliroentgen[a]
mrem	Millirem, see rem also
MT	Metric Ton[a]
MWd/MTU	Thermal Megawatt-day per Metric Ton of Uranium, the Burnup[a]

(a) See Section 14.2 for additional information or explanation.

MOX	Mixed Oxide
NEL-PIA	Nuclear Energy Liability - Property Insurance Association
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PWR	Pressurized Water Reactor
Q.A.	Quality Assurance[a]
Q.C.	Quality Control[a]
R	Roentgen[a]
rad	Radiation Absorbed Dose[a]
rem	Roentgen Equivalent Man[a]
RWP	Radiation Work Permit
SFP	Spent Fuel Pool
SNM	Special Nuclear Material[a]
SS	Stainless Steel
$T_{1/2}$, T_R	Half Life, Radiological[a]
UF	Urea-formaldehyde

Greek Letters:

α	Alpha Radiation[a]
β	Beta Radiation[a]
γ	Gamma Radiation[a]
χ	Chi, Concentration, Ci/m^3
Q	Released Quantity of Radioactive Material, Ci/sec
Q'	Release Rate of Radioactive Material, Ci/sec
$\bar{\chi}/Q'$	Chi-bar/Q prime, normalized average air concentration (Ci/m^3 per Ci/sec released, also written sec/m^3). Also called the annual average atmospheric dilution factor.

14.2 GLOSSARY DEFINITIONS

Abnormal Environmental Occurrence:	An event that 1) results in noncompliance with, or is in violation of, an Environmental Technical Specification, or 2) results in uncontrolled or unplanned releases of chemical, radioactive, or other discharges from the plant in excess of Federal, State or local regulations.

[a] See Section 14.2 for additional information or explanation.

Absorbed Dose:	When ionizing radiation passes through matter, some of its energy is imparted to the matter. The amount absorbed per unit mass of irradiated materials is called the absorbed dose; it is measured in rems and rads.
Activity:	Sometimes used for the term "radioactivity".
Airborne Radioactive Material:	Radioactive particulates, mists, fumes, and/or gases, air.
ALARA:	A philosophy to maintain exposure to radiation <u>A</u>s <u>L</u>ow <u>A</u>s is <u>R</u>easonably <u>A</u>chievable.
Alpha Decay:	Radioactive decay in which an alpha particle is emitted. This transformation lowers the atomic number of the nucleus by two and its mass number by four.
Alpha Particle:	A positively charged particle emitted by certain radioactive materials. It is made up of two neutrons and two protons, hence it is identical with the nucleus of a helium atom. It is the least penetrating of the three common types of radiation (alpha, beta and gamma) emitted by radioactive material.
Alpha Emitter:	A radionuclide that undergoes transformation by emission of alpha particles.
ATMX:	Railcars that are effectively used in transporting large quantities of low-level waste; specially reinforced, they have interior dimensions of 2.74 m (9 ft) x 2.74 m (9 ft) x 15.24 m (50 ft) with a useful load of 45,813 kg (101,000 lb). The Army Transport Mobile Explosives units were originally used by the U.S. Army for the transport of conventional explosives. The cars used for transporting nuclear materials are modified versions of the standard car.
Atomic Number (Z):	The number of protons in the nucleus of an atom; also its positive charge. Each chemical element has its characteristic atomic number, and the atomic numbers of the known elements form a complete series from 1 (hydrogen) to 105 (hehnium).
Availability Factor:	The proportion of time during a given period that a nuclear power reactor is available for operation.

Background:	That level of radioactivity from external sources existing without the presence of a nuclear plant, adjusted for any change occurring during the lifetime of a nuclear facility such as might result from atmospheric weapons testing.
Beta Decay:	Radioactive decay in which a beta particle is emitted or in which an orbital electron capture occurs.
Beta Particle:	An electron, of either positive or negative charge, which has been emitted by an atomic nucleus in a nuclear transformation.
Burial Grounds:	Areas designated for storage of containers of packaged radioactive wastes in near-surface soils.
Burnup, Specific:	The total energy released per unit mass of a nuclear fuel. It is commonly expressed in megawatt-days per ton. (Also called fuel irradiation level.)
Byproduct Material:	Any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material.
Capacity Factor:	The electricity actually produced compared to the total amount a nuclear power plant could produce if it could operate at 100% capacity around the clock every day in the year.
Cask:	A heavily shielded shipping container for radioactive materials. Some casks weigh as much as 100 tons.
Chemical Limits:	Maximum concentrations or quantities imposed upon chemical releases in gaseous or liquid effluents discharged from a facility, and consistent with known air or water quality standards.
Chemical Reprocessing:	Operations involved in the recovery of fissile material from irradiated fuel assemblies by chemical treatment. Chemical processing usually is done by dissolving the fuel in liquids and performing separation of products (U and Pu) from wastes by chemical differences in the liquid phase. Chemical reprocessing includes such operations as dissolving fuel, solvent extraction, heating or transferring process solutions, and adjusting chemical composition of process solutions.

Code of Federal Regulations (CFR): The Code of Federal Regulations is a codification of the general rules by the Executive departments and agencies of the Federal Government. The Code is divided into 50 titles that represent broad areas subject to Federal regulation. Each title is divided into Chapters that usually bear the name of the issuing agency. Each Chapter is further subdivided into Parts covering specific regulatory areas.

Contact Maintenance: "Hands-on", or maintenance performed by direct contact of personnel with the equipment. Most non-radioactive maintenance is contact maintenance.

Contamination: Radioactive material or materials that have been deposited on the surfaces of structures or equipment or that have been mixed with another material.

Continuing Care Period: The surveillance and maintenance phase of Safe Storage, with the facility secured against intrusion.

Curie: The special unit of activity. One curie equals 3.7×10^{10} nuclear transformations per second. (Abbreviated Ci.) Several fractions of the curie are in common usage:

- Millicurie. One-thousandth of a curie. Abbreviated mCi.
- Microcurie. One-millionth of a curie. Abbreviated μCi.
- Nanocurie. One-billionth of a curie. Abbreviated nCi.
- Picocurie. One-millionth of a microcurie. Abbreviated pCi; replaces the term μμc.

Decay, Radioactive: A spontaneous nuclear transformation in which a particle, gamma radiation or x radiation are emitted following orbital electron capture or spontaneous fission of the nucleus.

Decommissioning: Preparation of nuclear facilities for retirement from active service accompanied by the execution of a program to reduce or stabilize radioactive contamination to reduce the potential health and safety impacts on the public.

Decontamination: Those activities employed to reduce the levels of contamination in or on structures and equipment.

Decontamination Agents:	Those chemical materials used to effect decontamination.
Decontamination Factor (DF):	The ratio of the initial concentration of an undesired material to the final concentration resulting from a treatment process. The term may also be used as a ratio of quantities.
De minimus Level:	That level of contamination acceptable for unrestricted public use or access.
Design Basis Accident:	A postulated accident believed to have the most severe expected impacts on a facility. It is used as the basis for safety and structural design.
Discount Rate:	The rate of return on capital that could have been realized in alternative investments, if the money were not committed to the plan being evaluated, i.e., the opportunity cost of alternative investments. This cost is equivalent to the weighted average cost of capital.(1)
Disintegration, Nuclear:	Spontaneous nuclear transformation (radioactivity) characterized by the emission of energy and/or mass from the nucleus. The process is characterized by a definite half-life.
Disintegration Rate:	The rate at which disintegrations occur, characterized in units of time; i.e. disintegrations per minute (dpm), etc.
Dismantlement:	Those actions required to remove all radioactive or contaminated material from the facility, thus permitting unrestricted release of the property.
Dispersion:	A process of mixing one material within a larger quantity of another. For example, the mixing of material released to the atmosphere with air causes a reduction in concentration with distance from the source
Disposal:	The disposition of materials with the intent that the materials will not enter man's environment in sufficient amounts to cause a health hazard.
Dose, Absorbed:	The mean energy imparted to matter by ionizing radiation per unit mass of irradiated material at the place of interest. The unit of absorbed dose is the rad. One rad equals 0.01 Joules/kilogram in any medium (100 ergs per gram.)

(1) R. W. Johnson, <u>Capital Budgeting</u>. Wadsworth Publishing Co., Inc., Belmont, CA, p. 48, 1970.

Dose, Equivalent:	Expresses the amount of effective radiation, in man, expressed in rems, when modifying factors have been considered. The product of absorbed dose multiplied by a quality factor multiplied by a distribution factor.
Dose, Occupational:	The exposure of an individual to radiation above background as imposed by his employment.
Dose, Radiation:	As commonly used, it is the quantity of radiation absorbed in a unit mass of a medium, frequently a human organ.
Dose Rate:	The radiation dose delivered per unit time and measured, for instance, in rems per hour.
Dosimeter:	A device, such as a film badge or ionization chamber, that measures radiation dose.
Enrichment:	The ratio (usually expressed as a percentage) of fissile isotope to the total amount of the element (e.g., the % of ^{235}U in uranium.)
Entombment:	The encasement of radioactive materials in concrete or other structural material sufficiently strong and structurally long-lived to assure retention of the radioactivity until it has decayed to levels which permit unconditional release of the site.
Exposure:	A measure of the ionization produced in air by x or gamma radiation. It is the sum of the electrical charges on all ions of one sign produced in air when all electrons liberated by photons in a volume element of air are completely stopped in air, divided by the mass of the air in the volume element. The special unit of exposure is the roentgen. (See Roentgen.)
Facility:	The physical complex of buildings and equipment within a site.
Fission:	The splitting of a heavy nucleus into two lighter parts (nuclides of lighter elements), accompanied by the release of a relatively large amount of energy and generally one or more neutrons. Fission can occur spontaneously, but usually it is caused by nuclear absorpton of gamma rays, neutrons, or other particles.
Fission Products:	The lighter nuclides (fission fragments) formed by the fission of heavy elements. It also refers to the nuclides formed by the fission fragments' radioactive decay.

Food Chain:	The pathways by which any material (such as radioactive material from fallout) passes through man's environment through edible plants and/or animals to man.
Fuel Assembly:	A grouping of fuel elements that supply the nuclear heat in a nuclear reactor. A fuel element is the smallest structurally discrete part of a reactor or fuel assembly that has nuclear fuel as its principal constituent.
Fuel Cycle:	The series of steps involved in supplying fuel for nuclear power reactors. Head end: Mining, milling, enrichment, and fabrication of fuel. Back end: Includes reactors, spent fuel storage, spent fuel reprocess ing, mixed-oxide fuel fabrica tion and waste management.
Fuel Element:	A rod, tube, plate or other form into which nuclear fuel is fabricated for use in a reactor.
Fuel Pool Cooling System:	The system that cools and purifies the water in the fuel storage pool.
Fuel Reprocessing:	Same as chemical reprocessing.
Fuel Storage Pool:	A large concrete box full of water that provides storage and servicing facilities for nuclear fuel elements.
Gamma Rays:	High-energy, short-wavelength, electromagnetic radiation. Gamma radiation frequently accompanies alpha and beta emissions and always accompanies fission. Gamma rays are very penetrating and are best stopped or shielded against by dense materials such as lead or depleted uranium. The rays are similar to x-rays, but are usually more energetic, and are nuclear in origin, i.e., they originate from within the nucleus of the atom.
Gaseous:	Material in the vapor or gaseous state, but can include entrained liquids and solids. A gas will completely fill its container regardless of container shape or size.
Greenhouse:	A temporary structure, frequently constructed of wood and plastic, used to provide a confinement barrier between a radioactive work area and the environs.

Guard:	An individual whose primary duty is the guarding and protection of material against theft and/or the protection of the facility against industrial sabotage.
Half-Life Biological:	The time required for a biological system, such as a man or animal, to eliminate, by natural processes, half the amount of a substance (such as a radioactive material) that has entered it.
Half-Life Effective:	The time required for a radionuclide contained in a biological system, such as a man or animal, to reduce its radioactivity by half as a combined result of radioactive decay and biological elimination.
Half-Life Radioactive:	The time in which half the atoms of a particular radioactive substance disintegrate to another nuclear form. Each radionuclide has a unique half-life. Measured half-lives vary from millionths of a second to billions of years.
Health Physicist:	A person trained to perform radiation surveys, oversee radiation monitoring, estimate the degree of radiation hazard, and advise on radiation hazards.
Health Physics:	The science concerned with recognition, evaluation, and control of health hazards from ionizing radiation.
High-Level Radioactive Waste:	It is radioactive waste separated from the nuclear fuel reprocessing from the first-cycle solvent extraction system, or equivalent, and other concentrated wastes, or equivalent. It also applies generally to highly radioactive wastes of other origins.
Hot Spots:	Areas of radioactive contamination of a concentration higher than average.
Immobilization:	Treatment and/or emplacement of materials (e.g., radioactive contamination) so as to impede its movement.
Interim Care Period:	A period of time starting after the decommissioning activities cease and wherein periodic surveillance and maintenance takes place. The duration of time can vary from a few years to more than 100 years; also called the continuing care period.
Interim Storage:	Storage operations for which a) monitoring and human control are provided and b) subsequent action in which final disposition is expected.

Concepts for interim storage include bulk or compartmented storage of solid, liquid and gaseous wastes.

Intrusion Alarm:	A secure, electrical, electro-mechanical, electro-optical, electronic, mechanical or similar device capable of detecting intrusion by individuals into a protected area by means of visible or audible alarmed signal.
Ion Exchange:	A chemical process involving the absorption or desorption of various chemical ions in a solution onto a solid material, usually a plastic or resin. The process is used to separate and purify chemicals, such as fission products or hardness in water (i.e., water softening).
Licensed Material:	Source material, special nuclear material, or by-product material received, possessed, used, or transferred under a license issued by the Nuclear Regulatory Commission.
Liquid Radioactive Waste:	Solutions, suspensions, and mobile sludges, contaminated with radioactive materials.
Long-Lived Nuclides:	For this study, radioactive isotopes with long half-lives typically taken to be greater than about 10 years. Most nuclides of interest to waste management have half-lives on the order of one year to millions of years.
Low-Level Waste:	Wastes containing types and concentrations of radioactivity such that no shielding or relatively little shielding to minimize personnel exposure is required.
Management (Waste):	The planning, execution, and surveillance of essential functions related to radioactive waste, including treatment, solidification, interim or long-term storage, transportation and disposal.
Man-rem:	Used as a measure of population dose and it is calculated by summing the dose equivalent in rem received by each person in the population. Also, it is used as the absorbed dose of one rem by one person with no rate of exposure inferred.
Mass Numbers:	The number of nucleons (protons and neutrons) in the nucleus of an atom. (Symbol: A).
Maximum Individual:	A hypothetical individual in the general population who is located at the highest ground level and is subject to the greatest concentration of the material that is discharged from the plant.

Megawatt-day per metric ton:	A unit for expressing the burnup of fuel in a reactor; specifically, the number of megawatt-days of heat output per metric ton of fuel in the reactor.
Millirad:	A unit of absorbed dose (one thousandth of a rad). (See absorbed dose.)
Milliroentgen:	A submultiple of the roentgen, equal to one-thousandth of a roentgen. (See Roentgen.)
Monitoring:	Taking measurements or observations for recognizing adequacy, significant changes in, conditions or performance of a facility or area.
MOX:	An acronym for mixed oxide. A mixture of uranium and plutonium dioxide.
MT:	Metric Ton. (See Tonne.)
MW/MTU:	Thermal megawatts per metric ton of uranium.
MWd/MTU:	Thermal megawatt-days per metric ton of uranium; also called burnup. (See also specific power.)
Normal Operating Conditions:	Operation (including startup, shutdown, and maintenance) of systems within the normal range of applicable parameters.
Nuclear Reaction:	A reaction involving a change in an atomic nucleus, such as fission, fusion, or particle capture, or radioactive decay.
Nuclear Steam Supply System:	A contractual term which designates those components of the nuclear system and its related safeguards and instrumentation furnished by the nuclear steam supply system supplier.
Nuclear System:	Generally includes those systems most closely associated with the reactor vessel which are designed to contain or be in communication with the water coming from or going to the reactor core. The nuclear system includes the following:

- reactor,
- reactor assembly and internals,
- reactor core,
- neutron monitoring system,
- reactor recirculation system,
- control rod drive system,
- residual heat removal system
- chemical volume control system
- emergency core cooling systems.

Offsite:	Beyond the boundary line marking the limits of plant property.
Onsite:	Within the boundary line marking the limits of plant property.
Operable:	Capable of performing the required function.
Overpack:	Secondary (or additional) external containment for packaged nuclear waste.
Package:	The packaging plus the contents of radioactive materials.
Packaging:	The assembly of radioactive material in a container and other components necessary to assure compliance with prescribed regulations.
Possession-only License:	A license issued to a nuclear facility owner by the NRC entitling the licensee to own a nuclear facility but not operate it.
Power Reactor:	A generator of heat through controlled nuclear fission. Such heat energy, in turn, is used to generate power.
Present Value of Money:	When different business activities require disbursement of funds over different time frames, it is difficult to compare the actual cost of each activity to the sponsoring organization. One generally accepted method of placing these various disbursements on a common basis is to compute the value of those disbursements in terms of current dollars, i.e., the present value of money to be paid out or received at some time other than the present. For an investor, "the present value of a future payment or series of payments is the present investment necessary to secure the promise of that future payment or series of payments, with interest at a given rate."(2)
Primary Wastes:	Wastes that are generated as part of a primary operation. Secondary wastes are generated from a supporting operation, such as waste treatment.
Process Cells:	Heavily shielded rooms housing radioactive systems.
Process Equipment:	The functional equipment items or systems associated directly with the operation of a chemical or mechanical operation.

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

Protective Clothing:	Special clothing worn by a person in a radioactively contaminated area to prevent contamination of his body or personal clothing.
Protective Survey:	An evaluation of the radiation and its hazards incidental to the production, use or existence of radioactive materials. It normally includes a physical survey of the arrangement and use of equipment and measurements of the radiation dose rates under expected conditions of use. Also called protection survey.
Q-Designated Items:	The safety-related characteristics of those structures, systems, and components, both active and passive, that prevent or mitigate the consequence of postulated accidents that could cause undue risk to the health and safety of the public. Items defined as "Q" require the implementation of Quality Assurance Programs as set forth in Appendix B of 10 CFR Part 50. (See Safety Related also.) These items will withstand Design Basis Earthquakes or Design Basis Tornadoes.
Quality Assurance:	The systematic actions necessary to provide adequate confidence that a material, component, system, process, or facility performs satisfactorily or as planned in service.
Quality Control:	The quality assurance actions that control the attributes of the material, process, component, system, or facility in accordance with predetermined quality requirements.
Rad:	The unit of absorbed dose. The energy imparted to matter by ionizing radiation per unit mass of irradiated material at the place of interest. One rad equals 0.01 Joules/kilogram of absorbing material.
Radiation:	(1) The emission and propagation of radiant energy: for instance, the emission and propagation of electromagnetic waves, or of sound and elastic waves. (2) The energy propagated through space or through a material medium; for example, energy in the form of alpha, beta, and gamma emissions from radioactive nuclei.
Radiation Area:	Any area, accessible to personnel, in which there exists radiation at such levels that a major portion of the body could receive in any one hour a dose in excess of 2 millirem, or in any 7 consecutive days a dose in excess of 100 millirems.

Radiation Background:	See Background.
Radiation Leakage (Direct):	All radiation coming from a source housing except the useful beam.
Radiation Scattered:	Radiation that has been deviated in direction during its passage through a substance. It may also have been modified by a decrease in energy.
Radiation Stray:	The sum of leakage and scattered radiation; also called "shine".
Radioactive Material:	Any material or combination of materials which spontaneously emits ionizing radiation and which has a specific activity in excess of 0.002 microcuries per gram of material. (49 CFR 173.389(e)).
Radioactivity:	The property of certain nuclides of spontaneously emitting particles or gamma radiation or of emitting x radiation. Often shortened to "activity".
Radioactivity Artificial:	Manmade radioactivity produced by particle bombardment or electromagnetic irradiation, as opposed to natural radioactivity.
Radioactivity Induced:	Radioactivity produced in a substance after bombardment with neutrons or other particles. The resulting radioactivity is "natural radioactivity" if formed by nuclear reactions occurring in nature, and "artificial radioactivity" if the reactions are caused by man.
Radioactivity Natural:	The property of radioactivity exhibited by more than fifty naturally occurring radionuclides.
Radioactive Series:	A succession of nuclides, each of which transforms by radioactive disintegration into the next until a stable nonradioactive nuclide results. The first member is called the "parent", the intermediate members are called "daughters", and the final stable member is called the "end product".
Radiological Protection:	Protection against the effects of internal and external human exposure to radiation and to radioactive materials.
Regulatory Guides:	Regulatory Guides are issued to describe and make available to the public, methods acceptable to the NRC staff for implementing specific parts of the

Regulatory Guides: (con'd)	NRC's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide other guidance to applicants for nuclear operations. Guides are not substitutes for regulations and compliance with them is not explicitly required. Methods and solutions different from those set out in the guides are acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the NRC.
Rem:	(Acronym for Roentgen Equivalent Man.) A unit of dose equivalent. The dose equivalent in rems is numerically equal to the absorbed dose in rads multiplied by the quality factor, the distribution factor, and any other necessary modifying factors.
Remote Maintenance:	Maintenance by remote means, i.e., the human is separated from the item being maintained by a shielding wall.
Report Levels:	Those levels or parameters called out in the Environmental Technical Specifications, the Decommissioning Order, and/or the Possession-only License which do not limit decommissioning activities but which may indicate a measurable impact on the environment.
Repository (Federal):	A site owned and operated by the Federal Government for long-term storage or disposal of radioactive materials.
Restricted Area:	Any area to which access is controlled for protection of individuals from exposure to radiation and radioactive materials.
Roentgen:	A unit of exposure to ionizing radiation. It is that amount of gamma or X rays required to produce ions carrying one electrostatic unit of electrical charge (either positive or negative) in one cubic centimeter of dry air under standard conditions. One roentgen equals 2.58×10^{-4} coulomb per kilogram of air. (See also Exposure.)
Safe Storage:	Those actions required to place and maintain a nuclear facility in such a condition that future risk from the facility to public safety is within acceptable bounds and that the facility can be safely stored for as long a time desired.

Safety-Related: Structures, systems, and components, whose functions tend to prevent or mitigate the exceeding of safety limits, as defined in Regulatory Guide 3.6, and set forth in Technical Specifications which are part of the Operating License for a nuclear power plant. Quality Assurance Programs as defined in Appendix B of 10 CFR Part 50 are not required for safety-related items except those defined also as "Q".

Scarfing: A removal technique used to mechanically decontaminate concrete by chipping, cutting, jackhammering, or blasting the surface layer(s) away.

Secondary Wastes: Forms and quantities of all wastes that result from treatment of primary wastes or effluents.

Security Officer: A guard or watchman whose primary duty is the protection of material and property.

Shield: A body of material used to reduce the passage of particles or radiation. A shield may be designated according to what it is intended to absorb (as a gamma ray shield or neutron shield), or according to the kind of protection it is intended to give (as a background, biological, or thermal shield).

It may be required for the safety of personnel or to reduce radiation enough to allow use of counting instruments for research or for locating contamination or airborne radioactivity.

Short-Lived Radionuclides: For this study, those radioactive isotopes with half-lives less than about 10 years.

Shutdown: The time during which a site is not in production operation.

Site: The geographic area upon which the facility is located and which is subject to controlled public access by the facility licensee (includes the restricted area as designated in the NRC license).

Solid Radioactive Waste: Material that is essentially solid and dry but may contain sorbed radioactive fluids in sufficiently small amounts as to be immobile.

Solidification: Conversion of radioactive wastes (gases or liquids) to dry, stable solids.

Special Nuclear Material:	Plutonium, uranium enriched in the isotope, 233 or 235, and any other material as defined in 10 CFR 51 by the NRC.
Specific Power (of Fuel Assemblies):	Commonly expressed in units of thermal megawatts per metric ton of uranium (MW/MTU). It represents the rate at which thermal energy is extracted from the fuel; burnup, commonly expressed in thermal megawatt-days per metric ton of uranium (MWd/MTU), represents the total integrated energy extracted. For MOX fuel, the unit of fuel is a metric ton of heavy metal (MTHM); i.e., a metric ton of (U + Pu).
Surface Contamination:	The deposition and attachment of radioactive materials to a surface.
Surveillance:	Those activities necessary to assure that the site remains in a safe condition (including periodic inspection and monitoring of the site, maintenance of barriers to access to radioactive materials left on the site, and prevention of activities on the site which might impair these barriers).
Survey:	An evaluation of the radiation hazards incident to the production, use, release, disposal or presence of radioactive materials or other sources of radiation under a specific set of conditions.
Technical Specifications:	Requirements and limits which encompass nuclear safety but are simplified to facilitate use by plant operation and maintenance personnel. They are prepared in accordance with the requirements of 10 CFR 50.36, and are incorporated by reference into the Operating and/or Possession-only license issued by the NRC.
Tonne:	A metric ton, or 1000 kg, or 2204.6 lb.
Track Drill:	A self-propelled, air operated drill rig with an extendible boom capable of drilling 60 foot deep vertical holes in concrete and lifting the boxed material back to the surface.
Waste, Radioactive:	Equipment and materials (from nuclear operations) that are radioactive and for which there is no further use.
X-Ray:	A penetrating form of electromagnetic radiation emitted either when the inner orbital electrons of an excited atom return to their normal state (characteristic X rays) or when a metal target is bombarded with high speed electrons. X rays are always nonnuclear in origin, i.e., they originate external to the nucleus of the atom.

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