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Technology, Safety and Costs of Decommissioning Reference Light Water Reactors Following Postulated Accidents

Main Report

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Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory
Commission

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FOREWORD

BY

NUCLEAR REGULATORY COMMISSION STAFF

The NRC staff is reappraising its regulatory position relative to the decommissioning of nuclear facilities.[1] As a part of this activity, the NRC has initiated two series of studies through technical assistance contracts. These contracts are being undertaken to develop information to support the preparation of new standards covering decommissioning.

The basic series of studies covers the technology, safety, and costs of decommissioning reference nuclear facilities. Light water reactors and fuel-cycle and non-fuel-cycle facilities are included. Facilities of current design on typical sites are selected for the studies. Separate reports are prepared as the studies of the various facilities are completed.

The first report in this series covers a fuel reprocessing plant.[2] The second addresses a pressurized water reactor.[3] The third deals with a small mixed oxide fuel fabrication plant.[4] The fourth report, an addendum to the pressurized water reactor report,[5] examines the relationship between reactor size and decommissioning cost, the cost of entombment, and the sensitivity of cost to radiation levels, contractual arrangements, and disposal site charges. The fifth report in this series deals with a low-level waste burial ground.[6] The sixth covers a large boiling water reactor power station.[7] The seventh examines a uranium fuel fabrication plant.[8] The eighth report covers non-fuel-cycle nuclear facilities.[9] The ninth report, an addendum to the low-level waste burial ground report,[10] supplements the description of environmental radiological surveillance programs used in the parent document. The tenth report deals with a uranium hexafluoride conversion plant.[11] The eleventh report addresses the decommissioning of nuclear reactors at multiple-reactor power stations.[12] The twelfth report covers nuclear research and test reactors.[13] This report, thirteenth in the series, examines the decommissioning of light water reactors following postulated accidents.

Additional decommissioning topics will be reported on the tentative schedule as follows:

FY 1983 • Independent Spent Fuel Storage Installations

FY 1983 • Post-Accident Decommissioning at Fuel-Cycle Facilities

The second series of studies covers supporting information on the decommissioning of nuclear facilities. Five reports have been issued in the second series. The first consists of an annotated bibliography on the decommissioning of nuclear facilities.(14) The second is a review and analysis of current decommissioning regulations.(15) The third covers the facilitation of the decommissioning of light water reactors.(16) The fourth report covers the establishment of an information base concerning monitoring for compliance with decommissioning survey criteria.(17) The fifth report addresses the technology and cost of termination surveys associated with decommissioning of nuclear facilities.(18)

The information provided in this report on decommissioning of light water reactors following postulated accidents, including any comments, will be included in the record for consideration by the Commission in establishing criteria and new standards for decommissioning. Comments on this report should be mailed to:

Chief
Chemical Engineering Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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5. Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station. NUREG/CR-0130 Addendum, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, August 1979.

6. Technology, Safety and Costs of Decommissioning a Reference Low-Level Waste Burial Ground. NUREG/CR-0570, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, June 1980.

7. Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station. NUREG/CR-0672, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, June 1980.

8. Technology, Safety and Costs of Decommissioning a Reference Uranium Fuel Fabrication Plant. NUREG/CR-1266, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, October 1980.

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17. Monitoring for Compliance with Decommissioning Termination Survey Criteria. NUREG/CR-2082, Oak Ridge National Laboratory for U.S. Nuclear Regulatory Commission, June 1981

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ABSTRACT

Technical requirements, costs, and safety are conceptually evaluated for the post-accident cleanup and decommissioning of light water reactors.

The initial effort following a reactor accident is to bring the plant under control and to stabilize the reactor to prevent further accidents. Stabilization of the reactor is followed by accident cleanup and by decommissioning or refurbishment of the facility. This study provides an analysis of accident cleanup and decommissioning activities for three postulated accident scenarios. The scenario 1 accident is postulated to result in 10% fuel cladding failure, no fuel melting, moderate contamination of the containment structure, but no significant physical damage to buildings and equipment. The scenario 2 accident is postulated to result in 50% fuel cladding failure, a small amount of fuel melting, extensive radioactive contamination of the containment structure, moderate radioactive contamination of supporting buildings, and minor physical damage to buildings and equipment. The scenario 3 accident is postulated to result in 100% fuel cladding failure, significant fuel melting and core damage, severe radioactive contamination of the containment structure, moderate radioactive contamination of supporting buildings, and major physical damage to structures and equipment.

Accident cleanup is postulated to include: 1) the processing of contaminated water generated by the accident, 2) some initial decontamination of building surfaces to reduce the subsequent occupational dose to cleanup and decommissioning workers, 3) defueling of the reactor, 4) cleanup of the reactor coolant system, and 5) management of the resulting wastes. For the reference pressurized water reactor (PWR), accident cleanup is estimated to require 3.0 years and to cost $105 million following the scenario 1 accident, 5.3 years and $224 million following the scenario 2 accident, and 8.0 years and $404 million following the scenario 3 accident. For the reference boiling water reactor (BWR), accident cleanup is estimated to require 3.2 years and to cost $128 million following the scenario 1 accident, 5.3 years and $228 million following the scenario 2 accident, and 8.3 years and $421 million

following the scenario 3 accident. Costs are in early-1981 dollars. These costs and times include planning and preparation as well as the actual cleanup activities.

Decommissioning is assumed to follow accident cleanup. DECON at the reference PWR is estimated to cost \$49 million following cleanup after the scenario 1 accident, \$68 million following cleanup after the scenario 2 accident, and \$106 million following cleanup after the scenario 3 accident. Corresponding costs for SAFSTOR at the reference PWR with 100 years of safe storage are \$58 million, \$72 million, and \$102 million, respectively. Estimated costs for the entombment phase of ENTOMB at the reference PWR are \$38 million, \$52 million, and \$80 million, respectively. DECON at the reference BWR is estimated to cost \$67 million following cleanup after the scenario 1 accident, \$86 million following cleanup after the scenario 2 accident, and \$119 million following cleanup after the scenario 3 accident. Corresponding costs for SAFSTOR at the reference BWR with 100 years of safe storage are \$78 million, \$94 million, and \$120 million, respectively. Estimated costs for the entombment phase of ENTOMB at the reference BWR are \$52 million, \$67 million, and \$93 million, respectively. ENTOMB is a much less attractive decommissioning alternative following a reactor accident than it is following normal shutdown because of: 1) the higher levels of entombed radioactivity resulting from accident-generated contamination in the plant, and 2) the slower decay of the post-accident radionuclide inventory, which is controlled by ^{90}Sr and ^{137}Cs (with 30-year half-lives) rather than by ^{60}Co (with a 5.27-year half-life).

One of the most significant differences between post-accident cleanup and decommissioning and normal-shutdown decommissioning is the higher radiation exposure to workers during post-accident operations inside the containment structure. For accident cleanup and decommissioning following the scenario 2 accident at the reference PWR, total occupational radiation doses (external doses from gamma radiation) are estimated to be about 4580 man-rem for accident cleanup and about 3060 man-rem for DECON following accident cleanup. This compares with an occupational radiation dose of about 1200 man-rem for normal shutdown DECON at the reference PWR. For accident cleanup and decommissioning following the scenario 2 accident at the reference BWR, total occupational radiation doses are estimated to be about 4170 man-rem for accident cleanup and about 3180 man-rem for DECON following accident cleanup. This compares with an occupational radiation dose of about 1840 man-rem for normal shutdown DECON at the reference BWR. In order to ensure that worker doses are ALARA, careful planning and rehearsal of cleanup operations and the use of remote and semi-remote cleaning techniques are required to reduce occupancy times in high radiation areas and to minimize occupational exposures during accident cleanup. Manpower requirements must be adjusted during post-accident cleanup and decommissioning to ensure that doses to individual workers do not exceed specified limits.

The public safety impacts of post-accident cleanup and decommissioning are greater than the corresponding impacts of normal-shutdown decommissioning. However, radiation doses to the public from routine accident cleanup and decommissioning operations are below permissible radiation dose levels in unrestricted areas and are within the range of annual radiation doses from normal background. The primary contribution to public dose is the airborne release of particulate radioactivity during accident cleanup.

REPORT CONTENTS OUTLINE

CONTENTS

FIGURES

TABLES

CHAPTER 1

INTRODUCTION

This report contains the results of a study sponsored by the Nuclear Regulatory Commission (NRC) to conceptually evaluate post-accident decommissioning of light water reactors (LWRs). The primary purpose of this study is to provide information on the available technology, the safety considerations, and the probable costs of post-accident decommissioning and of accident cleanup activities that precede the decommissioning. Information from this study is intended as background data and to form a basis for the modification of existing regulations and the development of new regulations pertaining to decommissioning activities.

A post-operations activities flow sheet showing the sequence of operations leading to decommissioning and release of a reactor facility for unrestricted use is shown in Figure 1.0-1. As can be seen from the figure, the activities that follow shutdown of a facility that has been involved in an accident are somewhat different from the activities that follow normal shutdown. Post-accident activities include stabilization, accident cleanup, and active decommissioning. Stabilization is the period during which the accident is brought under control and the reactor is returned to a stabilized condition. Activities during this period include bringing the reactor to a safe shutdown condition, the restoration of essential systems and services required to maintain the reactor in a stabilized condition, and preliminary surveys to determine the extent of damage resulting from the accident and to assess the radiological condition of the plant. Once the situation is stabilized, accident cleanup can begin. Accident cleanup is considered to be those activities leading to defueling of the reactor, cleanup of contamination, and processing and disposal of wastes generated by the accident. As shown in Figure 1.0-1, accident cleanup could either be followed by recovery of the facility for restart or by decommissioning. If it is decided to retire the reactor from service, decommissioning activities are considered to begin following completion of the accident cleanup.

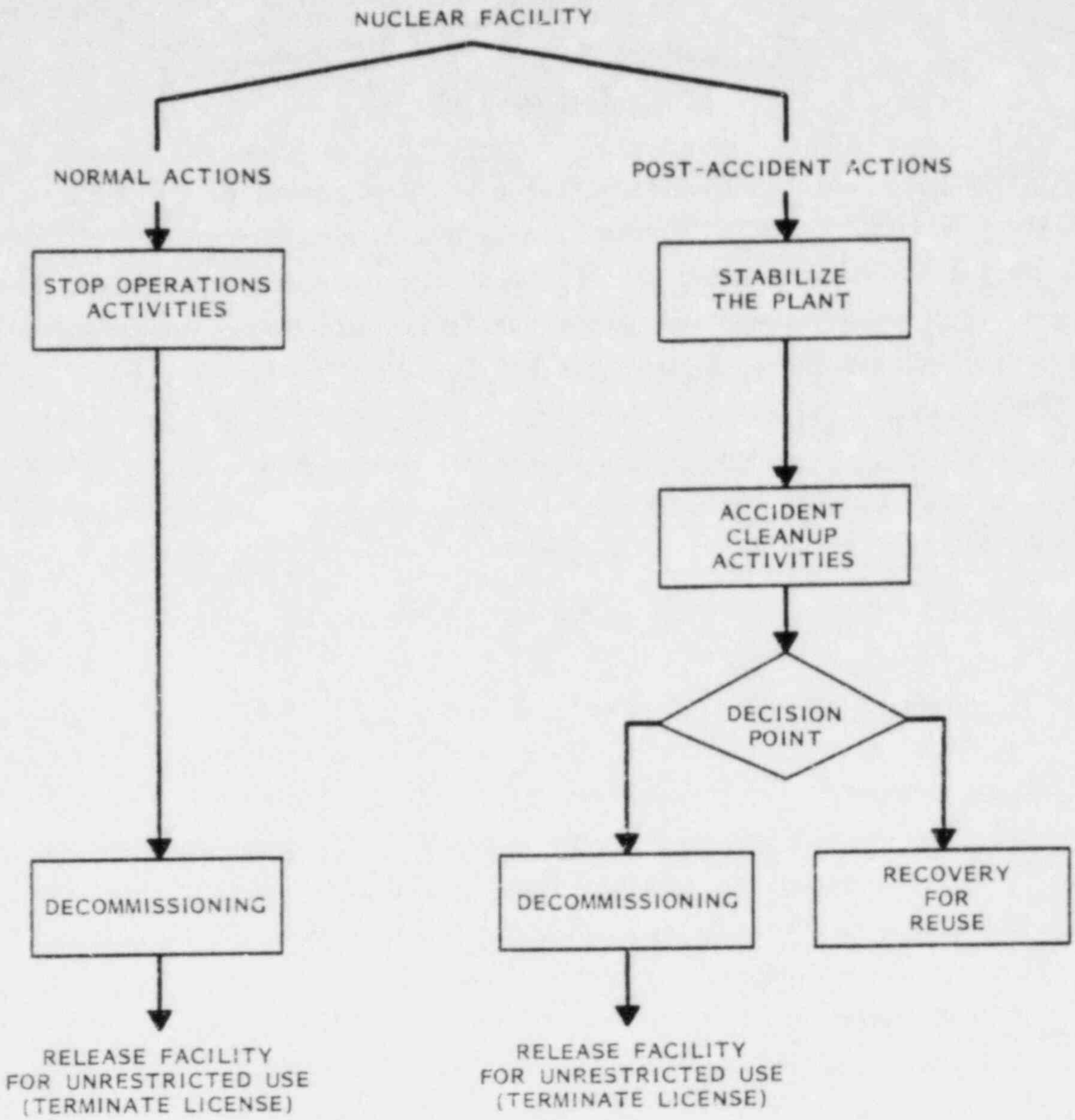

FIGURE 1.0-1. Post-Operations Activities Flow Sheet

This study considers the technology, safety, and costs of accident cleanup and decommissioning. Details of the stabilization period and of recovery of the reactor for restart are not analyzed. Because the overall accident cleanup activities would be similar whether the facility was ultimately restarted or decommissioned, the accident cleanup analysis is considered to be a good representation of this period independent of the ultimate use of the plant. Accident cleanup requirements are also shown to be essentially independent of the decommissioning alternative chosen.

Although many of the considerations involved with normal decommissioning also apply to cleanup and decommissioning following an accident, there are important differences in some of the specific requirements. To facilitate comparisons between the requirements and costs of post-accident and normal-shutdown decommissioning and to provide consistency with earlier decommissioning studies, previously published reports of reactor decommissioning following normal shutdown[1,2] are used for the reference facility descriptions and basic decommissioning information needed for this study. The reference PWR is the Trojan nuclear plant at Rainier, Oregon, operated by the Portland General Electric Company. The reference BWR is WPPSS Nuclear Project No. 2 (WNP-2) being built near Richland, Washington, by the Washington Public Power Supply System (WPPSS). The use of these reactors as reference facilities for this study should not be construed as implying anything about their reliability and/or safety relative to other LWRs in operation or under construction. Their use facilitates comparisons with the earlier, non-accident, decommissioning studies.

Requirements for post-accident decommissioning and for accident cleanup that precedes the decommissioning are analyzed for three accident scenarios. The scenarios are defined in terms of their consequences (e.g., radioactive contamination levels, radiation exposure rates, damage to the nuclear fuel core, and damage to buildings and equipment). The three scenarios are believed to be credible on the basis of reviews of accident literature, power reactor safety analysis reports, and NRC safety evaluation reports. The scenarios result in a range of accident cleanup and decommissioning requirements and costs. Minor accidents (i.e., spills, etc.) that would not result in a significant accident cleanup effort or that would not greatly impact the requirements and costs of decommissioning are not considered. Only accidents that result in widespread contamination and/or large-scale mechanical damage implying great difficulty in implementing accident cleanup and decommissioning procedures are considered.

The postulated accident scenarios, listed in increasing order of difficulty of the accident cleanup and decommissioning, are:

1. A small loss-of-coolant accident (LOCA) in which the emergency core cooling system (ECCS) functions to cool the core and limit the release of radioactivity. Some fuel cladding rupture is postulated, but no fuel melting. The consequence scenario includes moderate contamination of the containment structure but no significant physical damage to buildings and equipment.
2. A small LOCA in which emergency core cooling is delayed, resulting in 50% fuel cladding failure and a small amount of fuel melting. The consequence scenario includes extensive radioactive contamination of the containment structure, moderate radioactive contamination of supporting buildings, and minor physical damage to structures and equipment.
3. A major LOCA in which emergency core cooling is delayed, resulting in 100% fuel cladding failure and significant fuel melting and core damage. The consequence scenario includes severe radioactive contamination of the containment structure, moderate radioactive contamination of supporting buildings, and major physical damage to structures and equipment.

As discussed above, following the postulated accidents and stabilization periods, accident cleanup procedures begin. These accident cleanup procedures are designed to: 1) reduce the initial high levels of radioactive contamination present in the facility, thereby reducing the radiation dose received by cleanup and decommissioning workers, 2) process the contaminated water generated by the accident to reduce the amount of readily dispersible radioactivity present in the plant, and 3) safely defuel the reactor. The overall tasks that must be performed during accident cleanup are expected to be relatively independent of whether the plant is refurbished and restarted or decommissioned. However, the methods used to complete certain cleanup tasks may vary, depending on whether the decision is to restart or to decommission and, in the latter case, on the decommissioning alternative chosen.

It is beyond the scope of this study to evaluate the alternatives of permanently shutting down versus restarting a facility which has been involved in one of the accident scenarios described above. If the facility is to be permanently shut down, actual decommissioning activities are considered to begin following the completion of accident cleanup. Three alternative approaches to decommissioning are considered in this study. These are defined as:[3]

- DECON - The immediate removal from the facility of all material with residual radioactivity levels greater than those permitted for unrestricted use of the property. DECON meets the requirements for termination of the facility license and renders the facility and site available for unrestricted use within a finite time period.
- SAFSTOR - Activities designed to place (preparations for safe storage) and maintain (safe storage) a radioactive facility in such a condition that the risk to public safety is within acceptable bounds. At the conclusion of the safe storage period, the facility must be decontaminated to levels that permit its release for unrestricted use (deferred decontamination).
- ENTOMB - Cleanup and decontamination, to a lesser extent than for DECON, is coupled with the confinement of the remaining contaminated components in a strong and structurally long-lived material to assure retention until the radioactivity decays to levels that permit unrestricted release of the property.

The accident cleanup activities that precede decommissioning result in the initial decontamination of building surfaces and equipment and in the removal of major sources of radioactivity such as contaminated accident water, damaged fuel assemblies, and fuel debris. At the conclusion of the cleanup activities, many of the initial effects of an accident are eliminated or reduced in magnitude. Therefore, the requirements and costs of decommissioning following accident cleanup are not strongly affected by the specific condition of the plant immediately following an accident. One factor that can influence the decommissioning requirements and costs is the residual contamination remaining on building surfaces and equipment after accident cleanup activities are completed. Another factor that can influence the decommissioning is the need to decontaminate and dismantle equipment (such as the filter/demineralizer system for processing accident water) and structures (such as facilities for the interim storage of radioactive waste) that are installed and used during accident cleanup operations. A third factor that can influence decommissioning is the potential need to store waste onsite on an interim basis for an extended time period if the waste cannot be disposed of offsite because of technical, regulatory, or political constraints.

Decommissioning is analyzed in detail only for the scenario 2 accident. Estimates are made of differences in manpower requirements, occupational radiation exposures, and costs for decommissioning following the scenario 2 accident and for decommissioning following the scenario 1 and scenario 3 accidents. These differences are used as bases in estimating the safety impacts and costs of decommissioning for the scenario 1 and scenario 3 accidents.

Sets of work plans are developed for the conceptual accident cleanup of the reference LWRs and for decommissioning via the DECON, SAFSTOR, and ENTOMB alternatives. From these work plans, estimates are made of manpower requirements, major resource and equipment needs, volumes of contaminated material packaged for disposal, costs of accomplishing the work, and exposure of workers to radiation as a result of the accident cleanup and decommissioning efforts. These work plans and estimates of airborne releases of radioactive materials are used to evaluate the impacts of accident cleanup and decommissioning operations on the general public. The plans and techniques used in this study are believed to be representative of the state-of-the-art and to represent operations that would be required and could, therefore, be used to safely decontaminate and decommission an LWR that has been involved in an accident.

The safety impacts and estimated costs developed in this study are sensitive to the specifics of the reference LWRs and the postulated accidents. Such specifics include the levels of radioactive contamination and extent of physical damage associated with the accident, the operating history of the reactor, the location of the power plant relative to waste disposal sites, and the structural details of the plant. The safety impacts and costs are also sensitive to the assumptions made in this study about the techniques and procedures used for accident cleanup and decommissioning. The bases and assumptions on which the study is based should be carefully examined before using the study results.

The results of the PWR analysis are summarized in Chapters 10 through 15. Sufficient detail is provided in these chapters and their corresponding appendices to justify the results that are presented and to enable the reader to trace the logic used in arriving at these results. The BWR analysis, summarized in Chapter 16, is not performed to the same level of detail as the PWR analysis. To trace the logic and justify the assumptions used in making the BWR analysis, the reader must refer to the appropriate sections in Chapter 8 and in Chapters 10 through 15 where the reference accident scenarios and the technical requirements, costs, and safety impacts of PWR accident cleanup and decommissioning are discussed.

Detailed analyses of BWR post-accident cleanup and decommissioning activities are made where the results are expected to be significantly different from PWR accident cleanup results or from BWR normal-shutdown decommissioning results. For accident cleanup activities where the BWR results are judged not to be significantly different from the previously reported PWR results, the PWR results are used. For decommissioning activities where the post-accident results are judged not to be significantly different from the results for BWR normal-shutdown decommissioning reported in Reference 2, the normal-shutdown results are used. It is believed that this approach has resulted in reasonable estimates of time, manpower requirements, costs, and safety impacts of BWR accident cleanup and decommissioning while minimizing the analysis effort required to obtain the results.

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2. H. D. Oak, et al., Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station, NUREG/CR-0672, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, June 1980.

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CHAPTER 2

SUMMARY

This chapter summarizes the results of a study sponsored by the Nuclear Regulatory Commission (NRC) to conceptually evaluate post-accident decommissioning of light water reactors (LWRs) and accident cleanup that precedes the decommissioning. The purpose of the study is to provide information on the available technology, the safety considerations, and the probable costs of cleanup and decommissioning of LWRs that have experienced a significant accident. The principal results are given in the following sections.

2.1 STUDY APPROACH

Post-accident decommissioning and the accident cleanup activities are analyzed for three accident scenarios chosen to illustrate a range of technological requirements, occupational radiation doses, and cleanup and decommissioning costs that are substantially greater than those estimated for LWR decommissioning following normal shutdown. The parameters characterizing the reference accident scenarios that are important to accident cleanup and decommissioning activities are the resulting radioactive contamination levels, radiation exposure rates, damage to the nuclear fuel core, and damage to the containment structure and equipment. The postulated scenarios, listed in increasing order of the difficulty of post-accident cleanup and decommissioning, are:

1. A small loss-of-coolant accident (LOCA) (e.g., a small steam line break or the inadvertent opening of a safety or relief valve) in which the emergency core cooling system (ECCS) functions to cool the core and to limit the release of radioactivity. Some fuel cladding rupture is postulated, but no fuel melting. The consequence scenario includes moderate contamination of the containment structure but no significant physical damage to buildings and equipment.

2. A small LOCA in which emergency core cooling is delayed, resulting in 50% fuel cladding failure and a small amount of fuel melting. The consequence scenario includes extensive radioactive contamination of the containment structure, moderate radioactive contamination of supporting buildings, and minor physical damage to structures and equipment.

3. A major LOCA (e.g., the rupture of a main coolant line) in which emergency core cooling is delayed, resulting in 100% fuel cladding failure and significant fuel melting and core damage. The postulated consequences include severe radioactive contamination of the containment structure, moderate radioactive contamination of supporting buildings, and major physical damage to structures and equipment.

The activities that follow shutdown of a facility that has been involved in an accident are somewhat different from the activities that follow normal shutdown. Post-accident activities include stabilization, accident cleanup, and active decommissioning. Stabilization activities may include the efforts specifically designed to shut down the reactor and maintain it in a safe shutdown condition, such as use of emergency cooling systems and reactor control systems. They may also include efforts to stabilize the plant following the accident by isolating and containing accident contamination until cleanup facilities are available. In addition, stabilization activities may include the restoration of essential systems and services required to maintain the reactor in a safe shutdown condition. The specific nature of these stabilization activities can be very diverse depending on facility design and the nature of the accident. Because of this, and because the study primarily deals with decommissioning and accident cleanup, this study does not analyze the technology, safety, and costs of the stabilization period.

Once the situation is stabilized, accident cleanup can begin. Accident cleanup is considered to be those activities leading to defueling of the reactor, cleanup of contamination, and processing and disposal of wastes generated by the accident. Accident cleanup can either be followed by recovery of the facility for restart or by decommissioning. This study includes only an analysis of the activities related to decommissioning and does not analyze the activities related to refurbishment following accident cleanup.

As indicated above, this study provides an analysis of post-stabilization activities, including accident cleanup and decommissioning. The goals of accident cleanup are to: 1) reduce the initial high levels of radioactive contamination present in the facility, thereby reducing the radiation dose received by cleanup and decommissioning workers, 2) process the contaminated water generated by the accident to reduce the amount of readily dispersible radioactivity present in the plant, and 3) safely defuel the reactor. To achieve these goals, accident cleanup is postulated to include the following tasks:

- processing the contaminated water generated by the accident (and by decontamination operations) to remove radioactive materials
- initial decontamination of building surfaces and decontamination or disposal of some equipment
- removal of spent fuel (undamaged and damaged) from the reactor vessel and storage of the fuel in the spent fuel pool
- cleanup of the reactor coolant system
- solidification and packaging of wastes from the above accident cleanup operations.

These accident cleanup activities are necessary and would be approximately the same whether the reactor is ultimately refurbished or decommissioned, and if decommissioned would be independent of whichever decommissioning alternative is chosen. The rationale for this is discussed in detail in Section E.1 of Appendix E. Briefly stated, Section E.1 indicates that decontamination during the accident cleanup period (whether for eventual restart or decommissioning) cannot be too chemically corrosive or destructive, since this could compromise the integrity of systems that must remain intact during cleanup and decommissioning, especially if a delayed decommissioning alternative, such as SAFSTOR, is chosen. The work required to complete specific cleanup tasks is, of course, determined by the severity of the accident.

If it is decided that a facility is to be permanently shut down, actual decommissioning activities are considered to begin following the completion of the accident cleanup activities listed above. The alternatives for decommissioning are DECON (immediate decontamination to unrestricted release), SAFSTOR (safe storage with deferred decontamination to unrestricted release), and ENTOMB (entombment of radioactive materials with decay to unrestricted release). In this study, decommissioning is analyzed in detail only following the scenario 2 accident. Differences in manpower requirements, occupational radiation doses, and costs for decommissioning following the scenario 2 accident and for decommissioning following the scenario 1 and scenario 3 accidents are discussed in the text. In this manner, costs for decommissioning following scenario 1 and scenario 3 accidents are developed.

Work plans developed for the conceptual accident cleanup and for decommissioning provide the bases for estimates of manpower requirements, major resource and equipment needs, volumes of radioactive materials requiring disposal, exposure of workers to radiation as a result of cleanup and decommissioning efforts, and costs of accomplishing the work. Estimates of releases of radioactive material during accident cleanup and decommissioning operations are used to evaluate the impacts of these operations on the general public.

The results of the pressurized water reactor (PWR) analysis are given in Chapters 10 through 15. Sufficient detail is provided in these chapters and their corresponding appendices to justify the results that are presented and to enable the reader to trace the logic used in arriving at these results. The boiling water reactor (BWR) analysis, presented in Chapter 16, is not presented to the same level of detail as the PWR analysis and draws on the results of the PWR analysis whenever they are applicable. To trace the logic and justify the assumptions used in making the BWR analysis, the reader must refer to the appropriate sections of the PWR analysis where the reference accident scenarios and the technical requirements, costs, and safety impacts of PWR accident cleanup and decommissioning are discussed.

The results of the study are dependent on assumptions made with regard to facility design and location, accident scenarios, and resultant contamination. The results also depend on assumptions concerning capability to proceed with accident cleanup and eventual decommissioning and on work plans and techniques used to achieve the desired decommissioned condition. The choices of plans and techniques in the study are believed to be representative of the state-of-the-art and to represent operations that would be required and could therefore be used to safely decontaminate and decommission an LWR that had been involved in an accident. In using the results of this study, particularly as they apply to accident cleanup, consideration should be given to uncertainties in estimating contamination levels, core damage, and other factors that can affect the situation. Some of these factors are discussed in Sections 2.10.1 and 2.12. Application of the study results to situations different from those assumed in the study could produce erroneous conclusions.

2.2 KEY STUDY BASES

The major study bases are:

- The study must yield realistic and up-to-date results.
- Accident scenarios illustrate a range of accident cleanup and decommissioning requirements and are in the range of scenarios considered as design basis in NRC safety evaluations.
- To facilitate comparisons with the earlier non-accident LWR decommissioning studies, the same reference reactors are used for this post-accident study.
- The postulated activities are conducted within the framework of existing regulations and regulatory guidance.
- Current technology and techniques are used in descriptions of accident cleanup and decommissioning procedures.
- The postulated activities conform to ALARA occupational exposure philosophies.

- Unrestricted release of the decontaminated facility is predicated on decontamination to residual levels of radioactivity as specified by present regulatory guidance.
- Work schedules include allowances for inefficiencies associated with work in high-radiation areas.
- Manpower requirements are adjusted so that average individual radiation doses to workers do not exceed 5 rem/year with the intention of keeping occupational exposures ALARA.
- Costs of accident cleanup and decommissioning are in early-1981 dollars.
- Sufficient funding is available to carry out the accident cleanup and decommissioning without significant delays.
- No scheduling or funding allowances are made for research and development activities except those related to procurement of the special tools and equipment needed for accident cleanup.
- Based on proposed 10 CFR Part 61 criteria, low-level radioactive wastes are assumed to be disposed of by shallow-land burial. For other wastes that are unsuitable for shallow-land burial, criteria are not yet well defined. Hence, these wastes, specifically highly radioactive and/or transuranic wastes and damaged fuel assemblies, are assumed to be sent to a federal repository.

2.3 CLEANUP AND DECOMMISSIONING EXPERIENCE AT ACCIDENT-DAMAGED REACTOR FACILITIES

Very few reactor accidents have necessitated extensive post-accident cleanup operations or have resulted in a requirement to decommission the reactor. Most of the techniques and procedures used to decontaminate or decommission a reactor following an accident are similar to those used for reactor decommissioning following normal shutdown, although consideration must be given to the problems of working in higher than normal radiation areas.

Some reactor accidents have resulted in high levels of radioactive contamination on building surfaces and equipment and in high radiation exposure rates to accident cleanup personnel. In all cases where contamination has occurred, methods and procedures have been devised to safely remove the contamination with only modest total radiation doses to decontamination workers.

The 28 March 1979 accident at Three Mile Island, Unit 2 (TMI-2) resulted in an accident cleanup effort at that facility which will involve several years of work at a cost which is estimated to be about $1 billion.[1] Cleanup of TMI-2 will provide experience in procedures and techniques related to the processing of highly contaminated liquids, the removal of damaged fuel from a reactor, and the handling and disposal of high-activity radioactive waste.

2.4 REGULATORY GUIDANCE FOR ACCIDENT CLEANUP AND FOR DECOMMISSIONING

In general, regulations are in place that can be used to cover most aspects of the cleanup and decommissioning of a nuclear power reactor that has been involved in an accident. The existing regulations do not speak specifically to post-accident cleanup and decommissioning, but they can be interpreted as being applicable.

The decontamination and/or decommissioning of a reactor that has been involved in an accident is also subject to constraints imposed by statements, orders, and amendments to the facility license issued by the NRC subsequent to the accident. These constraints may relate to such activities as the controlled venting of the reactor building atmosphere, the use of special equipment for accident cleanup operations, the storage and/or disposal of radioactive wastes, and the release of processed accident water by evaporation or by discharge to a river. Statements, orders, and amendments to the facility license are of necessity specific to the particular reactor and accident and would be issued by the NRC on a case-by-case basis.

An important area of concern in the post-accident cleanup and decommissioning of a nuclear reactor is the management of the large volumes of radioactive wastes (gases, liquids, and solids) that result from the accident and from cleanup and decommissioning operations. Processed accident water will contain as radioactive contaminants the small amounts of fission product radionuclides not removed by filtration, evaporation, or ion exchange processes, as well as the tritium originally present in the water. Regulations and guidance exist that govern the disposal of this processed water by such alternatives as controlled discharge to a river, discharge to the atmosphere through natural or forced evaporation, or transportation as bulk liquid to an offsite location for disposal, although as indicated above these could be superseded on a case-by-case basis.

Solid radioactive wastes from the post-accident cleanup and decommissioning of a nuclear power reactor range from low-specific-activity trash and rubbish to high-specific-activity ion exchange resins, accident sludges, and spent filter cartridges. At the present time, only shallow-land burial grounds are available for the disposal of commercial radioactive wastes. The NRC proposes to add to its rules in 10 CFR a new Part 61 to provide licensing procedures, performance criteria, and technical criteria for licensing these burial facilities, including criteria for the classification of waste into different categories. The technical requirements on waste form and content imposed by Part 61 may result in some wastes from post-accident cleanup and decommissioning being deemed not suitable for near-surface burial.

A regulatory framework has not yet been developed to specifically address the disposal of low-level wastes that do not meet the criteria set out in 10 CFR Part 61 for near-surface disposal. Accordingly, certain of the post-accident cleanup and decommissioning wastes will have to be carefully evaluated on a case-by-case basis with regard to characteristics such as specific activity, radionuclide content, total radioactivity inventory, and waste form. Ultimate disposition of these wastes will depend on the unique characteristics they possess and on the availability of suitable facilities for their handling and disposal. In addition, in packaging and disposing of wastes resulting from chemical decontaminations, consideration must be given to applicable criteria on wastes containing chelating agents.

2.5 FINANCING FOR ACCIDENT CLEANUP AND FOR DECOMMISSIONING

Financing alternatives such as prepayment of decommissioning costs or a sinking fund or funded reserve that can be used to provide for the decommissioning of a nuclear power reactor following normal shutdown are generally not adequate for funding the costs of cleanup and the increased costs of decommissioning of a nuclear power rea ,or following an accident. Since reactor accidents occur very rarely, these expenses are more appropriately covered by insurance.

The accident at TMI-2 has provided a first major test of existing liability and property damage insurance coverage. While the liability insurance fostered under the Price-Anderson Act has been adequate in paying all claims to date, property damage insurance has been inadequate to pay the costs of accident cleanup. At the time of the TMI-2 accident in March 1979, the maximum amount of property damage insurance available was $300 million per insured unit.

To assure that licensees have the ability to finance the cleanup costs resulting from a nuclear-related accident, the NRC recently published an interim final rule(2,3) that deals with, among other items, the level of property insurance coverage necessary to cover decontamination and cleanup costs resulting from an accident at the licensed facility. The interim rule, published as an amendment to 10 CFR Part 50, Section 50.54, requires that each electric utility licensee obtain property damage insurance with a minimum coverage limit no less than the combined total of: 1) base coverage offered by either American Nuclear Insurers (ANI) and Mutual Atomic Energy Reinsurance Pool (MAERP) jointly or by Nuclear Mutual Limited (NML); plus 2) excess coverage offered by Nuclear Electric Insurance Limited (NEIL), the Edison Electric Institute (EEI), ANI and MAERP jointly, or NML. Currently, as of early 1982, the amount of base coverage is $450 million and the amount of excess coverage is $290 million.(4)

2.6 REFERENCE FACILITIES AND SITE

The reference PWR for this study is the Trojan nuclear plant at Rainier, Oregon, operated by the Portland General Electric Company. It is a 3500-MWt (1175-MWe) reactor of Westinghouse design and was used as the reference reactor for an earlier study[5] of PWR decommissioning following normal shutdown. The reference BWR for this study is WPPSS Nuclear Project No. 2 (WNP-2) being built near Richland, Washington, by the Washington Public Power Supply System (WPPSS). WNP-2 is a 3320-MWt (1155-MWe) reactor of the BWR/5 class and Mark-II containment design that was used as the reference reactor for an earlier study[6] of BWR decommissioning following normal shutdown. Use of these reactors as the reference facilities for this study is made to facilitate comparisons between the requirements and costs of post-accident decommissioning given in this study and the requirements and costs of normal-shutdown decommissioning given in the earlier studies. Their use is not intended to imply anything about the reliability and/or safety of these reactors relative to other LWRs in operation or under construction.

The reference site used in these analyses is typical of a midwestern or middle southeastern river site. This site has been developed for use in the series of decommissioning studies that is being performed for the NRC by Pacific Northwest Laboratory. Sufficient descriptive information is presented for both the facility and the site to permit the development of the detailed work plans, the cost estimates, and the safety assessments that are the results of this study.

2.7 REFERENCE ACCIDENT SCENARIOS AND RESULTANT CONTAMINATION LEVELS

Three reference accident scenarios provide the bases for the post-accident cleanup and decommissioning cost and safety estimates given in this report. From the viewpoint of this study, which deals with accident cleanup and decommissioning, the consequences of an accident (i.e., the radiological and physical condition of the plant following an accident) are much more important than the sequence of events that occur during the accident. Therefore, detailed descriptions of accident sequences are not given. The reference accident scenarios provide information about radioactive contamination, radiation exposure rates, and damage to the fuel core and to the containment structure. The consequence scenarios chosen for this study are believed to be credible with respect to initiating circumstances and are in the range of scenarios currently considered as design basis by the NRC in safety evaluations.

As this report is being completed, the NRC is performing research to assess behavior of nuclear power plant systems under a range of severe accident conditions, including a program to determine the radiological source term under these conditions.(7) As information from that program is developed, it should be used in potential addenda to this report.

Parameters that characterize the reference PWR following the postulated accidents used as bases in this study are listed in Table 2.7-1. Parameters that characterize the reference BWR following the postulated accidents are listed in Table 2.7-2. The parameter values listed in the tables refer to conditions at the reference reactors 1 year after the postulated accidents.

2.8 ACCEPTABLE RESIDUAL RADIOACTIVE CONTAMINATION LEVELS FOR A DECOMMISSIONED LWR

Unrestricted release of a nuclear reactor facility following decommissioning requires that the radioactivity remaining in the facility and on the site be reduced to levels that are considered acceptable for unrestricted access and subsequent NRC license termination. Criteria that currently exist or are being developed for the unrestricted release of a decommissioned facility are summarized in Chapter 9. Criteria under development will base allowable residual radioactivity levels for facility release on the dose anticipated to be received by individuals who use the facility or site after the license is terminated.

2.9 ACCIDENT CLEANUP AND DECOMMISSIONING ACTIVITIES

As described in Section 2.1, post-stabilization activities at an accident-damaged nuclear power reactor include: 1) preparations for accident cleanup, 2) accident cleanup of the containment building and of other buildings as necessary, and, if a reactor is to be permanently shut down,

TABLE 2.7-1. Reference PWR Accident Parameters

Parameter	Parameter Value[a] Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
Percent of fuel cladding failure	10	50	100
Percent of fuel melting	0	5	50
Volume of sump water (m^3)	200	1000	1600[b]
Depth of sump water (m)	0.2	1.0	1.6
Total fission product radioactivity in sump water (Ci)	2.5×10^4	3.5×10^5	2.5×10^6
Average fission product radioactivity in sump water (Ci/m^3)	125	350	1560
Total fission product radioactivity plated out on building surfaces (Ci)[c]	5	70	500
Average fission product radioactivity on building surfaces (Ci/m^2)			
• Floors	0.001	0.014	0.1
• Walls	0.00001	0.00014	0.001
Average gamma radiation exposure rate at operating floor level (R/hr)			
• Contribution from plateout	0.01	0.15	1.0
• Contribution from sump water	0.015	0.045	0.2
• Total exposure rate	0.025	0.20	1.2
Average gamma radiation exposure rate at lowest entry level (R/hr)			
• Contribution from plateout	0.01	0.15	1.0
• Contribution from sump water	8	30	170
• Total exposure rate	8	30	170
Damage to fuel core	Slight damage to some fuel elements as a result of fuel swelling and cladding rupture.	Oxidation of fuel cladding. Melting and fusing together of stainless steel fittings on center fuel elements. Cracking and crumbling of some fuel pellets. Melting of fuel in localized areas of central core.	Cracking, crumbling, and melting of fuel pellets. Melting and fusing together of stainless steel parts on adjacent fuel assemblies. Molten fuel present over much of core radius. Fuel and cladding fragments carried throughout primary coolant system.
Damage to containment building and equipment.	No significant physical damage.	Contamination of building ventilation system. Some electrical equipment and some valves inoperable due to water damage and corrosion. Minor structural damage. Polar crane inoperable.	Ventilation ductwork damaged. Doors, catwalks, pipes, and cable conduits dented or ripped away. Loss of electrical and other services. Erosion of concrete and metal surfaces. Polar crane inoperable.
Contamination of auxiliary and fuel buildings	--[d]	Plateout on building surfaces. CVCS contaminated with 20,000 Ci of fission product radioactivity. General area radiation exposure levels about 100 mR/hr.	Plateout on building surfaces. CVCS contaminated with 20,000 Ci of fission product radioactivity. General area radiation exposure levels about 100 mR/hr.

(a) Values refer to conditions inside the containment building approximately 1 year after the postulated accident.
(b) Based on refueling water storage tank volume.
(c) Plateout values are after washdown of the walls by condensing moisture.
(d) Contamination of the auxiliary and fuel buildings is not postulated for the scenario 1 accident.

TABLE 2.7-2. Reference BWR Accident Parameters

Parameter	Parmeter Value[a] Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
Percent of Fuel Cladding Failure	10	50	100
Percent of Fuel Melting	0	5	50
Volume of Suppression Pool Water (m^3)	3160[b]	3160[b]	3160[b]
Total Fission Product Radioactivity in Suppression Pool Water (Ci)	2.5×10^4	3.5×10^5	2.2×10^6
Average Fission Product Radioactivity in Suppression Pool Water (Ci/m^3)	8	110	700
Volume of Reactor Building Sump Water (m^3)	0	0	500
Total Fission Product Radioactivity in Reactor Building Sump Water (Ci)	0	0	3×10^5
Average Fission Product Radioactivity in Reactor Building Sump Water (Ci/m^3)	--	--	700
Total Fission Product Radioactivity Plated Out on Containment Vessel Surfaces (Ci)[c]	5.2	73	460
Average Fission Product Radioactivity on Containment Vessel Surfaces (Ci/m^2)			
• Floors	0.005	0.07	0.44
• Walls	0.00005	0.0007	0.0044
Average Gamma Radiation Exposure Rate at Operating Floor Level Inside Containment (R/hr)			
• Contribution from Plateout	0.052	0.720	4.6
• Contribution from Suppression Pool Water	0.006	0.070	0.5
• Total Exposure Rate	0.058	0.790	5.1
Total Fission Product Radioactivity Plated Out on Reactor Building Surfaces (Ci)	0	10	82
Average Fission Product Radioactivity on Reactor Building Surfaces (Ci/m^2)			
• Floors	--	0.001	0.008
• Walls	--	0.00001	0.00008
Average Gamma Radiation Exposure Rate at Refueling Floor level in Reactor Building (R/hr)[d]			
• Contribution from Plateout	--	0.002	0.020
• Contribution from Sump Water	--	--	0.0
• Total Exposure Rate	--	0.002	0.020

(contd on next page)

TABLE 2.7-2. (contd)

Parameter	Parameter Value(a) Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
Average Gamma Radiation Exposure Rate at Operating Floor Level in Reactor Building(e) (R/hr)			
• Contribution from Plateout	--	0.010	0.083
• Contribution from Sump Water	--	--	0.002
• Total Exposure Rate	--	0.010	0.085
Average Gamma Radiation Exposure Rate at Service Floor Level in Reactor Building(f) (R/hr)			
• Contribution from Plateout	--	0.010	0.083
• Contribution from Sump Water	--	--	30
• Total Exposure Rate	--	0.010	30
Damage to Fuel Core	Slight damage to some fuel elements as a result of fuel swelling and cladding rupture.	Oxidation of fuel cladding. Melting and fusing together of stainless steel fittings on center fuel elements. Cracking and crumbling of some fuel pellets. Melting of fuel in localized areas of central core.	Cracking, crumbling, and melting of fuel pellets. Melting and fusing together of stainless steel parts on adjacent fuel assemblies. Molten fuel present over much of core radius. Fuel and cladding fragments carried throughout water recirculation system.
Damage to Containment Vessel and Equipment	No significant physical damage.	Most electrical equipment and some valves inoperable due to water damage and corrosion. Minor structural damage.	Pipes and cable conduits dented or ripped away. Loss of electrical and other services. Recirculation system pump motors inoperable due to damage to electrical components and corrosion.
Damage to Reactor Building and Equipment	No significant physical damage.	No significant physical damage	Contamination of building ventilation system. Some electrical equipment and some valves inoperable due to water damage and corrosion. Minor structural damage. Bridge crane and refueling platform inoperable due to damage to electrical components and corrosion.
Contamination of Radwaste Building	--(g)	--(g)	Plateout on building surfaces. Reactor water cleanup demineralizer system grossly contaminated. General area radiation exposure levels about 50 mR/hr.

(a) Values refer to conditions approximately 1 year after the accident.
(b) Based on maximum water volume specified in Section C.2.1 of Reference 1.
(c) Plateout values are after washdown of walls by condensing moisture.
(d) The refueling floor level is the 185.0-m level. See Figure 16.2-3.
(e) The operating floor level is the 152.7-m level. See Figure 16.2-3.
(f) The service floor level is the 134.4-m level. See Figure 16.2-3.
(g) Contamination of radwaste building is postulated only for the scenario 3 accident.

3) decommissioning of the facility. This section briefly summarizes some requirements and procedures of these activities.

2.9.1 Preparations for Accident Cleanup

Preparations for accident cleanup include the following activities (not necessarily in the order listed):

- venting of radioactive gases (e.g., ^{85}Kr)
- containment entry and data acquisition
- preparation of documentation for regulatory agencies
- design, fabrication, and installation of special equipment
- development of detailed work plans and procedures
- selection and training of accident cleanup staff
- removal of the spent fuel stored in the spent fuel storage pool from prior reactor refuelings.

The ^{85}Kr (10.7-year half-life) present in the containment structure atmosphere must be removed to reduce radiation exposure levels and permit worker access to the building for data gathering and accident cleanup operations. It is assumed in this study that environmental effects are minimal enough to permit the ^{85}Kr to be removed from the building atmosphere by controlled venting that involves release of the air from inside the building by way of filtering and monitoring equipment that leads to the building ventilation stack. Alternative methods for ^{85}Kr removal are considered as part of a sensitivity analysis presented in Chapter 11.

Initial entries into the contaminated and damaged containment structure are made for the purpose of obtaining data on the radiological and physical condition of the building. Data obtained during these entries include measurements of contamination levels and radiation exposure rates, estimates of physical damage, and information about the operational status of plant systems and services. These data are needed to provide a basis for planning accident cleanup operations and for preparing documentation for regulatory agencies.

A major task is the preparation by the licensee of the necessary documentation to amend the facility operating license to maintain the reactor in a safe shutdown condition and to obtain regulatory approvals to proceed with cleanup operations. The time requirement for furnishing information to regulatory agencies, issuing environmental statements and assessments, and securing regulatory approvals to proceed with specific cleanup tasks is a critical factor in determining when actual cleanup operations can begin and could be a cause of delay in accident cleanup.

Several major facility and equipment items needed for accident cleanup are designed and fabricated during preparations for cleanup. These items include:

- a filter/demineralizer system for processing contaminated water
- processed water storage tanks and associated piping and controls
- special tools for the removal and handling of damaged fuel assemblies
- a mockup of a section of the reactor vessel for use in testing fuel removal equipment and for training personnel to use the equipment
- storage facilities for interim storage of radioactive wastes.

Detailed work plans and work procedures for accident cleanup are developed during the preparations phase. Requirements for cleanup of the contaminated plant are based on an evaluation of the condition of the plant following the accident.

The selection and training of key staff for accident cleanup begins during the preparations phase.

It is postulated in this study that the filter/demineralizer system used to process contaminated water is installed in the spent fuel pool. The basis for this assumption is that the pool provides shielding and cooling of the equipment, and use of the existing pool and its associated facilities avoids the necessity of constructing a new building. To provide space in the pool for the filter/demineralizer system and for temporary storage of the damaged fuel from reactor defueling, it is necessary to remove the spent fuel already stored in the pool from prior refuelings of the reactor. This fuel is assumed to be transported to an independent spent fuel storage installation (ISFSI) for interim storage. Because of potential problems that could prevent transport of this fuel to an offsite storage location, this study includes an analysis of the added costs of constructing a new building to house the filter/demineralizer system.

2.9.2 Accident Cleanup Activities

Accident cleanup activities are postulated to include the following tasks:

- processing of contaminated liquids
- initial decontamination of the containment structure surfaces and equipment
- defueling of the reactor
- cleanup of the primary coolant system (PWR) or the reactor water recirculation system (BWR)
- treatment and disposal of wastes from cleanup operations.

Contaminated liquids that must be processed during accident cleanup include accident water (radioactively contaminated water that is released during the accident), contaminated water and chemical decontamination solutions generated during decontamination of containment structure surfaces, reactor coolant system water, and reactor coolant system chemical decontamination and flush solutions. Contaminated water is treated by use of the filter/demineralizer system installed in the spent fuel pool during preparations for cleanup. Chemical decontamination solutions are treated in an evaporator/solidification facility obtained from a commercial supplier.

The objective of initial decontamination of containment structure surfaces and equipment is to reduce surface contamination levels and resultant radiation exposure rates to permit reasonable occupancy times for workers engaged in reactor defueling and reactor coolant system cleanup operations. Initial decontamination includes the following steps:

- use of the containment spray system for a remote wash of building surfaces (if it is practical to do this)

- removal and packaging of small items of contaminated equipment that are easily disposed of
- use of high-pressure hose wash techniques for semi-remote decontamination of building surfaces and equipment
- decontamination and refurbishment of essential support systems
- hands-on decontamination of selected areas
- local shielding of "hot spots."

Defueling the reactor involves the following steps:

- preparations for defueling
- removal of the reactor pressure vessel head and inspection of the core
- removal of structural components above the fuel
- removal of intact fuel assemblies and removal and packaging of damaged fuel assemblies
- removal of fuel element debris.

The difficulty of the reactor defueling operation is determined by the extent of damage to the fuel core and to the reactor vessel during the accident. Damage to the fuel, to the reactor vessel head, and to internal support structures is postulated to be different for each accident scenario evaluated in this report.

Decontamination of the primary coolant system involves the removal of fuel debris (scenario 2 and scenario 3 accidents) and the removal of fission product plateout (all three scenarios). To dissolve the fuel debris that accumulates in pumps, piping, and other components of the RCS, an oxalic-peroxide-gluconic (OPG) solution is assumed to be used. To remove fission product plateout from internal surfaces of coolant system components, ethylenediaminetetraacetic acid (EDTA) is used in combination with citric and oxalic acid in a weak (5%) solution at controlled pH. A system flush with processed water is interposed between use of the two solutions and a second system flush completes the decontamination process.

Radioactive wastes from accident cleanup operations include:

- dry solid wastes (trash, contaminated equipment and material, and irradiated, activated hardware)
- contaminated sludges and process solid wastes from the treatment of accident water and decontamination liquids
- immobilized chemical decontamination solutions
- fuel assemblies (damaged and undamaged) and core debris.

Dry solid wastes and immobilized chemical decontamination solutions are packaged and shipped to a shallow-land burial ground for disposal. Because of their high radioactivity content, process solid wastes are assumed to be shipped to a federal repository for storage or disposal. Undamaged fuel assemblies are shipped to an ISFSI and damaged assemblies are packaged in steel canisters and shipped to a federal repository. This report also analyzes the situation in which these wastes must be stored onsite for an extended time period.

2.9.3 Decommissioning Activities

Following the completion of the accident cleanup activities, the actual decommissioning activities begin. In this study, this is considered to be following reactor defueling and reactor coolant system cleanup. As a result of the efforts during accident cleanup, the decommissioning activities are considered to be not greatly affected by the condition of the plant immediately following the accident. In addition, many of the uncertain conditions have been removed during the accident cleanup--specifically, the damaged core has been removed from the reactor, the large volumes of uncontained highly radioactive water have been processed, and large areas of contaminated building surfaces have been treated. Hence decommissioning can be carried out in a more stable environment than accident cleanup.

Many decommissioning tasks are common to both post-accident and normal-shutdown decommissioning. However, changes in the physical and radiological condition of the plant that still remain following accident cleanup can result in substantial changes in the time and manpower required for post-accident decommissioning. Radiation doses to workers during post-accident decommissioning are hi her than those following normal shutdown because of contamination on building and equipment surfaces that remains after accident cleanup activities are completed. Physical damage to the plant may compromise some systems and equipment needed for the performance of decommissioning tasks, thus necessitating repairs or substitutions and increasing the time and cost of post-accident decommissioning.

In this study, post-accident decommissioning is analyzed for the DECON, SAFSTOR, and ENTOMB alternatives.

DECON is the decommissioning alternative that leads to the earliest release of the facility and site for unrestricted use and to the earliest termination of the facility's nuclear license. Compared to the other two decommissioning alternatives, DECON results in a larger occupational radiation dose and a larger cost in the first few years after the completion of accident cleanup.

The decontamination and dismantlement activities during post-accident DECON are similar to activities during DECON following normal shutdown, described in References 2 and 3. These activities include:

- decontamination of the surfaces of process systems and equipment
- disassembly and disposal of neutron-activated components, including the reactor vessel and vessel internals
- disassembly and disposal of contaminated equipment, including ductwork, piping, and pool liners
- removal of contaminated concrete
- packaging and shipment of radioactive wastes to a waste disposal site
- a final radiation survey.

Some of these activities are initiated during accident cleanup. However, the bulk of this work is carried out during DECON, particularly the removal of large equipment components and of contaminated structural material. Accident-generated contamination results in a somewhat greater level of effort and greater volume of radioactive waste material for post-accident DECON than for DECON following normal shutdown.

Post-accident SAFSTOR includes preparations for safe storage of the accident-damaged facility, continuing care for a specified period during which the radioactivity within the plant is allowed to decay, and eventual deferred decontamination of the facility. An advantage of SAFSTOR is that it satisfies the requirements for protection of the public while minimizing initial commitments of time, money, occupational radiation dose, and offsite waste disposal space compared to DECON. Disadvantages of SAFSTOR include the need to maintain the nuclear license during a period of safe storage and the absence of personnel familiar with the plant and the accident to assist in deferred decontamination. If the wastes from accident cleanup and decommissioning, including the damaged fuel from defueling the reactor, cannot be shipped offsite for an extended period, it may be necessary to use SAFSTOR.

Post-accident SAFSTOR activities are similar to those for SAFSTOR following normal shutdown of a reactor. However, occupational doses to decommissioning workers will be significantly higher during post-accident SAFSTOR because of the higher radiation exposure levels within the plant. Decommissioning worker requirements during post-accident deferred decontamination are controlled by radiation dose rates based on the decay of ^{137}Cs (the controlling radionuclide in the post-accident radionuclide inventory with a 30-year half-life) rather than by ^{60}Co with a 5.27-year half-life.

Activities for post-accident ENTOMB are similar to ENTOMB activities following normal shutdown. PWR entombment is assumed to take place in the lower portion of the containment building inside the shielded structures that house the steam generators, the pressurizer, and the reactor vessel, and below the operating floor. BWR entombment is assumed to take place inside the containment vessel. Prior to entombment, the reactor vessel internals containing long-lived activation products (e.g., ^{59}Ni, ^{94}Nb) are removed and shipped to a waste repository. After emplacement of the waste to be entombed, all penetrations through the entombment structure exterior are sealed. The portions of the plant remaining outside of the entombment structure are decontaminated in the same manner as for DECON.

The use of ENTOMB following a serious accident is unlikely because of the presence of significant quantities of long-lived (i.e., 30-year half-life) radionuclides that could require several hundred years to decay to release levels. If it becomes desirable to terminate the nuclear license prior to the decay of the entombed radioactive material to unrestricted release levels, dismantlement of the entombment structure would be required. This represents a task that is much more difficult than dismantlement of the unentombed facility, since the entombment structure is built to endure for a long period of time.

2.10 COSTS OF ACCIDENT CLEANUP AND OF DECOMMISSIONING

Estimated costs of accident cleanup and of decommissioning at the reference PWR and the reference BWR are summarized in this section. All costs are in early-1981 dollars and include a 25% contingency.

2.10.1 Costs of Accident Cleanup

Accident cleanup costs at the reference PWR and the reference BWR are summarized in this section. These costs are based on the key study basis assumptions listed in Section 4.2 of Chapter 4, and on the accident cleanup activities described for the reference PWR in Chapter 10 and for the reference BWR in Chapter 16. The comparison of estimated costs of accident cleanup at the reference PWR with TMI-2 cleanup cost estimates, presented in detail in Section 11.5 of Chapter 11, is also summarized in this section. Finally, the discussion of the sensitivity of accident cleanup costs to various factors that can affect these costs, presented in detail in Section 11.6, is summarized here.

The cost impacts of a possible requirement for extended onsite storage of radioactive wastes from accident cleanup and decommissioning, discussed in Chapter 15, are summarized later in Section 2.12.

2.10.1.1 Summary of Estimated Costs of Accident Cleanup at the Reference PWR and the Reference BWR

Total estimated costs and estimated time requirements for accident cleanup at the reference PWR are shown in Table 2.10-1. Accident cleanup following the scenario 1 accident is estimated to cost $105 million and to require 3.0 years for completion. Accident cleanup following the scenario 2 accident is estimated to cost $224 million and to require 5.3 years for completion. Accident cleanup following the scenario 3 accident is estimated to cost $404 million and to require 8.0 years for completion. These costs and times include the requirements for planning and preparation as well as for the actual cleanup activities.

TABLE 2.10-1. Summary of Time and Cost Estimates for Accident Cleanup at the Reference PWR Following the Postulated Accidents

	Cleanup Following Scenario 1 Accident		Cleanup Following Scenario 2 Accident		Cleanup Following Scenario 3 Accident	
	Time (years)	Cost[a] ($ millions)	Time (years)	Cost[a] ($ millions)	Time (years)	Cost[a] ($ millions)
Preparations for Accident Cleanup	1.5	33.7	2.5	67.2	3.0	98.0
Accident Cleanup in Auxiliary and Fuel Buildings	--[b]	--[b]	--[c]	19.5[d]	-- [c]	19.5[d]
Accident Cleanup in Containment Building	1.5	71.5	2.8	137.1	5.0	287.0
Totals	3.0	105.2	5.3	223.8	8.0	404.5

(a) Costs are in early-1981 dollars and include 25% contingency.
(b) Accident cleanup in the auxiliary and fuel buildings is not postulated following the scenario 1 accident.
(c) Accident cleanup in the auxiliary and fuel buildings is postulated to be completed during preparations for accident cleanup in the containment building.
(d) Includes the costs of cleanup worker labor, waste management, and equipment, supplies, and services for accident cleanup in the auxiliary and fuel buildings. Management and support staff costs and incidental costs (e.g., energy, insurance, etc.) are included in the costs of preparations for accident cleanup.

Total estimated costs and estimated time requirements for accident cleanup at the reference BWR are shown in Table 2.10-2. Accident cleanup following the scenario 1 accident is estimated to cost $128 million and to require 3.2 years for completion. Accident cleanup following the scenario 2 accident is estimated to cost $228 million and to require 5.3 years for completion. Accident cleanup following the scenario 3 accident is estimated to cost $421 million and to require 8.3 years for completion.

TABLE 2.10-2. Summary of Time and Cost Estimates for Accident Cleanup at the Reference BWR Following the Postulated Accidents

	Cleanup Following Scenario 1 Accident		Cleanup Following Scenario 2 Accident		Cleanup Following Scenario 3 Accident	
	Time (years)	Cost ($ millions)[a]	Time (years)	Cost ($ millions)[a]	Time (years)	Cost ($ millions)[a]
Preparations for Accident Cleanup	1.5	30.1	2.0	49.7	3.0	90.3
Accident Cleanup in the Radwaste Building	--[b]	--[b]	--[b]	--[b]	--[c]	13.1[d]
Accident Cleanup in the Reactor Building & Containment	1.7	98.4	3.3	178.5	5.3	317.5
Totals	3.2	128.5	5.3	228.2	8.3	420.9

(a) Costs are in early-1981 dollars and include a 25% contingency.
(b) Accident cleanup in the radwaste building is not postulated following the scenario 1 and scenario 2 accidents.
(c) Accident cleanup in the radwaste building following the scenario 3 accident is postulated to be completed during preparations for cleanup in the reactor building.
(d) Includes the costs of cleanup worker labor, waste management, equipment, supplies, and services for accident cleanup in the radwaste building. Management and support staff costs and incidental costs (e.g., energy, insurance, etc.) are included in the costs of preparations for accident cleanup.

Accident cleanup costs at the reference PWR and the reference BWR are shown by cost category in Tables 2.10-3 and 2.10-4 to illustrate the relative importance of individual cost items. The major cost items are labor and waste management. Labor costs, including staff labor and engineering support, account for about 40 to 62% of the total costs of accident cleanup at the reference PWR and the reference BWR, depending on accident scenario. Waste management costs accounts for about 12 to 30% of the total costs of accident cleanup at the reference PWR and about 17 to 39% of the total costs of accident cleanup at the reference BWR, depending on accident scenario. The major waste management cost item is the cost of disposal of the damaged fuel from reactor defueling following an accident. Other important accident cleanup costs include energy costs and the costs of special facilities and equipment.

TABLE 2.10-3. Summary of Accident Cleanup Costs at the Reference PWR by Cost Category[a,b]

Cost Category	Accident Cleanup Following Scenario 1 Accident		Accident Cleanup Following Scenario 2 Accident		Accident Cleanup Following Scenario 3 Accident	
	Estimated Costs ($ millions)	Percent of Total	Estimated Costs ($ millions)	Percent of Total	Estimated Costs ($ millions)	Percent of Total
Preparations for Accident Cleanup[c]						
Utility Staff Labor	16.010	47.5	30.478	45.3	43.770	44.7
Waste Management	0.156	0.5	0.464	0.7	0.589	0.6
Energy	9.034	26.8	15.056	22.4	18.068	18.4
Special Equipment and Facilities	1.679	5.0	3.680	5.5	6.913	7.1
Miscellaneous Supplies	0.094	0.3	0.156	0.2	0.188	0.2
Specialty Contractors	3.879	11.5	12.731	18.9	22.846	23.3
Nuclear Insurance and License Fees	2.821	8.4	4.680	7.0	5.610	5.7
Subtotals for Preparations for Cleanup	33.673	100.0	67.245	100.0	97.984	100.0
Accident Cleanup in Auxiliary and Fuel Buildings[d]						
Cleanup Worker Labor			14.065	72.2	14.065	72.2
Waste Management			1.615	8.3	1.615	8.3
Special Tools and Equipment			0.356	1.8	0.356	1.8
Miscellaneous Supplies			1.794	9.2	1.794	9.2
Specialty Contractors			1.638	8.5	1.638	8.5
Subtotals for Cleanup in Auxiliary & Fuel Buildings			19.468	100.0	19.468	100.0
Accident Cleanup in Containment Building						
Operations & Support Staff Labor	13.385	18.7	29.170	21.3	61.171	21.3
Accident Cleanup Staff Labor	6.806	9.5	27.769	20.2	93.491	32.6
Waste Management[e]	4.880	6.8	9.968	7.3	18.107	6.3
Disposal of Fuel from Reactor Defueling[e]	26.055	36.4	26.180	19.1	26.430	9.2
Energy	9.675	13.5	18.145	13.2	32.253	11.2
Special Tools and Equipment	3.781	5.3	7.813	5.7	17.063	5.9
Miscellaneous Supplies	1.858	2.6	4.691	3.4	8.688	3.0
Specialty Contractors	2.298	3.2	7.835	5.7	20.453	7.1
Nuclear Insurance and License Fees	2.789	3.9	5.578	4.1	9.298	3.2
Subtotals for Cleanup in Containment Building	71.527	99.9[f]	137.149	100.0	286.954	99.8[f]
Total Accident Cleanup Costs	105.2		223.8		404.5	

(a) Costs are in early-1981 dollars and include a 25% contingency.
(b) Number of figures shown is for computational accuracy only and does not imply precision to the nearest one thousand dollars.
(c) Costs are based on assumed time periods of 1.5 years for preparations for cleanup following the scenario 1 accident, 2.5 years for preparations for cleanup following the scenario 2 accident, and 3 years for preparations for cleanup following the scenario 3 accident.
(d) Accident cleanup in the auxiliary and fuel buildings is assumed to be accomplished during preparations for cleanup in the containment building. Management and support staff costs and other incidental costs are included in the costs of preparations for cleanup.
(e) Costs for disposal of fuel are shown separately from other waste management costs.
(f) Total does not equal 100% because individual percentages are rounded to the nearest one-tenth.

TABLE 2.10-4. Summary of Accident Cleanup Costs at the Reference BWR by Cost Category[a,b]

Cost Category	Accident Cleanup Following Scenario 1 Accident: Estimated Costs ($ millions)	Percent of Total	Accident Cleanup Following Scenario 2 Accident: Estimated Costs ($ millions)	Percent of Total	Accident Cleanup Following Scenario 3 Accident: Estimated Costs ($ millions)	Percent of Total
Preparations for Accident Cleanup[c]						
Utility Staff Labor	16.199	53.7	24.381	49.0	43.771	48.6
Waste Management	0.188	0.6	0.444	0.9	0.569	0.6
Energy	4.845	16.1	6.460	13.0	9.690	10.7
Special Equipment and Facilities	2.074	6.9	4.324	8.7	7.593	8.4
Miscellaneous Supplies	0.094	0.3	0.125	0.3	0.188	0.2
Speciality Contractors	3.923	13.0	10.231	20.6	22.846	25.3
Nuclear Insurance & License Fees	2.821	9.4	3.751	7.5	5.610	6.2
Subtotals for Preparations for Cleanup	30.144	100.0	49.716	100.0	90.267	100.0
Accident Cleanup in Radwaste Building[d]						
Cleanup Worker Labor					8.040	61.3
Waste Management					1.005	7.7
Special Tools and Equipment					1.500	11.4
Miscellaneous Supplies					1.094	8.3
Specialty Contractors					1.488	11.3
Subtotals for Cleanup in the Radwaste Building					13.127	100.0
Accident Cleanup in the Reactor Building and Containment						
Operations and Support Staff Labor	15.168	15.4	[illegible].381	19.3	66.136	20.8
Accident Cleanup Staff Labor	14.218	14.5	38.116	21.4	100.915	31.8
Waste Management[e]	7.979	8.1	19.188	10.8	27.341	8.6
Disposal of Fuel from Reactor Defueling[e]	42.020	42.7	42.145	23.5	42.395	13.3
Energy	6.203	6.3	12.018	6.7	18.773	5.9
Special Tools and Equipment	3.781	3.8	7.813	4.4	17.063	5.4
Miscellaneous Supplies	2.571	2.6	8.910	5.0	12.891	4.1
Specialty Contractors	2.665	2.7	9.351	5.2	21.771	6.9
Nuclear Insurance & License Fees	3.750	3.8	6.540	3.7	10.259	3.2
Subtotals for Cleanup in the Reactor Building & Containment	98.355	100.0	178.462	100.0	317.544	100.0
Total Accident Cleanup Costs	128.5		228.2		420.9	

(a) Costs are in early-1981 dollars and include 25% contingency.
(b) Number of figures shown is for computational accuracy only and does not imply precision to the nearest one thousand dollars.
(c) Costs are based on assumed time periods of 1.5 years for preparations for cleanup following the scenario 1 accident, 2 years for preparations for cleanup following the scenario 2 accident, and 3 years for preparations for cleanup following the scenario 3 accident.
(d) Accident cleanup in the radwaste building following the scenario 3 accident is assumed to be accomplished during preparations for cleanup in the reactor building. Management and support staff costs and other incidental costs are included in the costs of preparations for cleanup.
(e) Costs for disposal of fuel are shown separately from other waste management costs.

2.10.1.2 Comparison of Accident Cleanup Costs at the Reference PWR with TMI-2 Cleanup Cost Estimates

The accident that occurred at Three Mile Island Unit 2 (TMI-2) is the only major accident that has occurred at a large power reactor. Hence comparisons between the estimated accident cleanup costs at a reference PWR, presented in this study, and cost estimates for cleanup of TMI-2 are useful. A detailed discussion of these cost comparisons is given in Section 11.5. A brief summary of that section is given here.

Consideration of the contamination levels and cleanup methodology to be used for cleanup of TMI-2 indicates that accident cleanup activities at TMI-2 contain some characteristics of the activities postulated for the scenario 2 and scenario 3 accidents of this study. A recent estimate(1) of the cost of TMI-2 cleanup puts that cost at $1.034 billion. Accident cleanup costs at the reference PWR following the scenario 2 and scenario 3 accidents are estimated to be about $224 million and $404 million, respectively.

Certain legitimate cost items are included in the TMI-2 estimate that are not included in this study of the engineering costs of accident cleanup. These cost items include $124 million for base operations and maintenance and $209 million for escalation of costs due to inflation during cleanup. (The costs presented in this study are in constant, early-1981 dollars.) An additional cost of about $84 million is listed in the TMI-2 cost estimate for the construction of cleanup systems and equipment not considered to be necessary at the reference plant because of differences in plant design. The TMI-2 cost estimate also includes an allowance of approximately $100 million for additional decontamination of the containment building following defueling, during which time it is assumed in this study that decommissioning has begun and, hence, much of this cost is charged to decommissioning. A final major difference is that approximately $226 million was spent on cleanup at TMI-2 in the first 2-1/2 years following the accident. This study estimates that for the reference PWR $87 million would be spent in the first 2-1/2 years following a scenario 2 accident, and that $118 million would be spent in the first 3 years after a scenario 3 accident. This difference may result in part from costs of facility stabilization at TMI-2, which are not included in this study. This difference also serves to illustrate that in a real situation there may be legitimately higher costs arising from unavoidable delay. In a report prepared by the General Accounting Office,(8) it was pointed out that significant costs are incurred simply to maintain the status quo during delays.

When these differences in cost are considered, the cost estimates made in this study of a reference plant are considered to be reasonable estimates of the engineering costs of accident cleanup.

2.10.1.3 Sensitivity of Accident Cleanup Costs to Factors That Can Affect These Costs

Accident cleanup is an activity that takes place at a time when conditions in the plant are uncertain and when social, political, financial, and regulatory constraints can affect the progress and costs of cleanup activities. In addition, the processing of accident-generated wastes and the defueling of the reactor may require the use of specialized procedures and techniques. The accident cleanup costs summarized in Section 2.10.1.1 are based on assumptions about plant characteristics and cleanup procedures and techniques that are discussed in Section 4.2 of Chapter 4 and in Chapters 10 and 11. To provide information about the cost impacts of possible alternatives, the sensitivity of accident cleanup costs to various factors that can influence these costs is discussed in Section 11.6 of Chapter 11. These factors and their effects on accident cleanup costs are summarized here.

1. The potential for delays in accident cleanup. The added costs resulting from a 1-year delay in accident cleanup activities at the reference PWR are estimated to range from \$20 to \$32 million, depending on accident scenario and on when the delay occurs. Each additional year of delay would result in similar costs.

2. The need to use special processing systems or equipment. As an example, a requirement to use a technique other than purging for removal of the ^{85}Kr from the containment building atmosphere is estimated to add \$25 to \$60 million to the cost of accident cleanup at the reference PWR, depending on the krypton removal alternative chosen.

3. The need to construct special buildings. Construction of a special building to house the filter/demineralizer system used to process accident water is estimated to add about $12 million to the cost of accident cleanup.

4. Special waste disposal requirements. In this study, water that has been processed for the removal of radioactive contaminants is assumed to be discharged to the river when the water purity meets regulatory standards. A requirement for immobilization and disposal of this processed water at a shallow-land burial ground could add more than $2 million to the cost of accident cleanup.

5. The use of contractors to accomplish accident cleanup activities. The added costs of using a contractor to perform the accident cleanup activities in the containment building of the reference PWR are estimated to be about $10 million for cleanup following the scenario 1 accident, about $25 million for cleanup following the scenario 2 accident, and about $60 million for cleanup following the scenario 3 accident.

6. In addition to the above factors, there are certain other factors as discussed in Section 2.10.1.2 that must be considered in assessing potential costs during the post-accident period. These include the effects of cost escalation during the cleanup period, costs of additional necessary plant-specific facilities, costs incurred during the stabilization period, and costs incurred for additional building decontamination leading to recovery of the facility. Also, the need to store wastes onsite temporarily for an extended period can represent a long-term activity with specific associated costs (see Section 2.12).

As can be seen from the above discussion, there are many factors affecting post-accident cleanup costs. It is considered that this study represents a reasonable estimate of the engineering costs that could be incurred in the cleanup of a reactor following an accident. However, because the conditions in the plant are uncertain, careful consideration should be given to the factors that can affect cost.

2.10.2 Costs of Decommissioning

Total estimated costs of decommissioning following accident cleanup are summarized in Table 2.10-5 for the reference PWR and in Table 2.10-6 for the reference BWR. Decommissioning cost details are presented in Chapter 13 for the reference PWR and in Section 16.7 of Chapter 16 for the reference BWR.

TABLE 2.10-5. Summary of Estimated Costs for Decommissioning at the Reference PWR Following Accident Cleanup

	Costs ($ millions)[a]		
Decommissioning Alternative	Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
DECON	49.4	67.8	106.2
SAFSTOR with 30 Years Safe Storage			
Preparations for Safe Storage	14.7	16.6	21.4
Annual Continuing Care Costs	0.110	0.110	0.110
Deferred Decontamination	42.0	57.6	90.3
Total SAFSTOR Costs	60.0	77.5	115.0
SAFSTOR with 100 Years Safe Storage			
Preparations for Safe Storage	14.7	16.6	21.4
Annual Continuing Care Costs	0.110	0.110	0.110
Deferred Decontamination	32.6	44.7	70.1
Total SAFSTOR Costs	58.3	72.3	102.5
ENTOMB[b]			
Entombment Phase	38.5	52.5	79.6
Annual Continuing Care Costs	0.055	0.055	0.055

(a) Costs are in early-1981 dollars and include a 25% contingency.
(b) If required, deferred decontamination at the end of the continuing care period for ENTOMB is estimated to be more costly than deferred decontamination at the end of the corresponding continuing care period for SAFSTOR because dismantlement of an entombed structure is more difficult than dismantlement of an unentombed facility.

TABLE 2.10-6. Summary of Estimated Costs for Decommissioning at the Reference BWR Following Accident Cleanup

Decommissioning Alternative	Costs ($ millions)[a] Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
DECON	67.0	85.7	119.4
SAFSTOR with 30 Years Safe Storage			
Preparations for Safe Storage	28.0	32.3	38.4
Annual Continuing Care Costs	0.10	0.10	0.10
Deferred Decontamination	54.3	69.1	96.7
Total SAFSTOR Costs	85.3	104.4	138.1
SAFSTOR with 100 Years Safe Storage			
Preparations for Safe Storage	28.0	32.3	38.4
Annual Continuing Care Costs	0.10	0.10	0.10
Deferred Decontamination	40.2	51.2	71.6
Total SAFSTOR Costs	78.2	93.5	120.0
ENTOMB[b]			
Entombment Phase	51.9	66.7	93.1
Annual Continuing Care Costs	0.05	0.05	0.05

(a) Costs are in early-1981 dollars and include a 25% contingency.
(b) If required, deferred decontamination at the end of the continuing care period for ENTOMB is estimated to be more costly than deferred decontamination at the end of the corresponding continuing care period for SAFSTOR because dismantlement of an entombed structure is more difficult than dismantlement of an unentombed facility.

For DECON at the reference PWR, the estimated costs are $49.4 million following cleanup after the scenario 1 accident, $67.8 million following cleanup after the scenario 2 accident, and $106.2 million following cleanup after the scenario 3 accident. For DECON at the reference BWR, the estimated costs are $67.0 million following cleanup after the scenario 1 accident, $85.7 million following cleanup after the scenario 2 accident, and $119.4 million following cleanup after the scenario 3 accident. About 30 to 60% of the total DECON costs are for staff labor. Waste management costs are about 20 to 30% of the total DECON costs.

For SAFSTOR at the reference PWR with 30 years of safe storage, the total estimated costs are $60.0 million following cleanup after the scenario 1 accident, $77.5 million following cleanup after the scenario 2 accident, and $115.0 million following cleanup after the scenario 3 accident. Corresponding costs for SAFSTOR with 100 years of safe storage are $58.3 million, $72.3 million, and $102.5 million, respectively. For SAFSTOR at the reference BWR with 30 years of safe storage, the costs are $85.3 million, $104.4 million and $138.1 million for SAFSTOR following cleanup after the scenario 1, scenario 2, and scenario 3 accidents, respectively. Corresponding costs at the BWR for SAFSTOR with 100 years of safe storage are $78.2 million, $93.5 million, and $120.0 million, respectively. Deferred decontamination accounts for the majority of these costs, while preparations for safe storage account for about 20 to 30% of the totals. Staff labor costs are the major costs for both preparations for safe storage and deferred decontamination.

For ENTOMB at the reference PWR, the estimated costs of the entombment phase are $38.5 million following cleanup after the scenario 1 accident, $52.5 million following cleanup after the scenario 2 accident, and $79.6 million following cleanup after the scenario 3 accident. For ENTOMB at the reference BWR, the estimated costs of the entombment phase are $51.9 million, $66.7 million, and $93.1 million, respectively. Annual continuing care costs for ENTOMB are estimated to be about one-half of the annual continuing care costs for SAFSTOR. If required, deferred decontamination at the end of a continuing care period for ENTOMB is estimated to be more costly than deferred decontamination at the end of the corresponding continuing care period for SAFSTOR because dismantlement of an entombed structure is more difficult than dismantlement of an unentombed facility.

2.11 OCCUPATIONAL AND PUBLIC SAFETY

Radiological and nonradiological safety impacts from routine activities and from potential industrial and transportation accidents are identified and evaluated for post-accident cleanup and decommissioning at the reference LWRs. Safety impacts are calculated for accident cleanup following each of the three reactor accident scenarios. Decommissioning safety impacts are calculated for the DECON, SAFSTOR, and ENTOMB alternatives following a scenario 2 reactor accident and are estimated for the other accident scenarios (see Section 14.3.2). The safety evaluation includes consideration of radiation dose to the public from routine activities and from postulated industrial accidents. The evaluation employs 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 impacts is believed to be conservative.

The results of the safety evaluations of post-accident cleanup activities are summarized in Table 2.11-1 for the reference PWR and in Table 2.11-2 for the reference BWR. Details of the safety evaluation of post-accident cleanup activities are given in Section 14.2 of Chapter 14 for the reference PWR and in Section 16.8.1 of Chapter 16 for the reference BWR. The principal source of radiation dose to the public is the atmospheric release of radionuclides from the facility during routine activities. Potential lost-time injuries to

TABLE 2.11-1. Summary of Safety Analysis for Accident Cleanup at the Reference PWR

Type of Safety Concern	Source of Safety Concern	Units	Accident Scenario 1	Accident Scenario 2	Accident Scenario 3
	Public Safety[a]				
Radiation Dose	Accident Cleanup Activities[b]	man-rem	6	20	40[c]
	Transportation[d]	man-rem	1.6	4.7	9.6
	Occupational Safety				
Serious Lost-Time Injuries	Accident Cleanup Activities	total no.	0.30	1.3	2.1
	Transportation	total no.	0.17	0.51	1.1
Fatalities	Accident Cleanup Activities	total no.	0.0023	0.0094	0.015
	Transportation	total no.	0.010	0.030	0.066
Radiation Dose	Accident Cleanup Activities	man-rem	670	4 579	12 103
	Transportation	man-rem	17	46	99

(a) Radiation doses from atmospheric releases during normal cleanup activities. Doses resulting from industrial accidents are not included.
(b) 50-yr committed dose equivalent to the bone, for the total population within 80 km of the site.
(c) Doses from activities in auxiliary and fuel buildings not calculated, expected to be negligible compared to those shown.
(d) 50-yr committed dose equivalent to the total body, for the population along the transport route.

TABLE 2.11-2. Summary of Safety Analysis for Accident Cleanup at the Reference BWR

Type of Safety Concern	Source of Safety Concern	Units	Accident Scenario 1	Accident Scenario 2	Accident Scenario 3
	Public Safety[a]				
Radiation Dose	Accident Cleanup Activities[b]	man-rem	6	20	40
	Transportation[c]	man-rem	3	5	11
	Occupational Safety				
Serious Lost-Time Injuries	Accident Cleanup Activities	total no.	0.54	1.0	2.3
	Transportation	total no.	0.31	0.54	1.3
Fatalities	Accident Cleanup Activities	total no.	0.0038	0.0072	0.016
	Transportation	total no.	0.019	0.032	0.076
Radiation Dose	Accident Cleanup Activities	man-rem	1 490	4 170	11 940
	Transportation	man-rem	28	50	120

(a) Radiation doses from atmospheric releases during normal cleanup activities. Doses resulting from industrial accidents are not included.
(b) 50-yr committed dose equivalent to the bone, for the total population within 80 km of the site.
(c) 50-yr committed dose equivalent to the total body, for the population along the transport route.

workers are primarily due to accident cleanup activities, although about one-third of the injuries are estimated to result from transportation tasks. Essentially no fatalities are predicted to occur during accident cleanup. Occupational radiation doses during accident cleanup result almost entirely from routine onsite activities.

The results of the safety evaluations of post-accident decommissioning activities are summarized in Table 2.11-3 for the reference PWR and in Table 2.11-4 for the reference BWR. Details of the safety evaluations of post-accident decommissioning activities are given in Section 14.3 of Chapter 14 for the reference PWR and in Section 16.8.2 of Chapter 16 for the reference BWR. The principal source of radiation dose to the public during decommissioning is the transport of radioactive materials from the reactor station to disposal facilities, and the estimated dose to the public resulting from decommissioning activities is small. About three or four lost-time injuries to workers from industrial-type accidents are predicted to occur during decommissioning. Fatalities from industrial-type accidents appear to be unlikely during decommissioning. Occupational radiation doses during decommissioning result primarily from routine onsite activities.

TABLE 2.11-3. Summary of Safety Analysis for Decommissioning at the Reference PWR Following the Cleanup of a Scenario 2 Accident

Type of Safety Concern	Source of Safety Concern	Unit	DECON	SAFSTOR with Deferred Decontamination after: 30 Years	100 Years	ENTOMB
	Public Safety[a]					
Radiation Dose	Decommissioning Activities[b]	man-rem	<0.001	<0.001	<0.001	<0.001
	Transportation[c]	man-rem	19	1.2[d]	1.2[d]	8.4
	Continuing Care	man-rem	--	neg.[e]	neg.	neg.
	Occupational Safety					
Serious Lost-Time Injuries	Decommissioning Activities	total no.	0.79	0.92	0.92	0.78
	Transportation	total no.	2.2	2.2	2.2	1.0
	Continuing Care	total no.	--	neg.	neg.	neg.
Facilities	Decommissioning Activities	total no.	0.0046	0.0052	0.0052	0.0045
	Transportation	total no.	0.13	0.13	0.13	0.060
	Continuing Care	total no.	--	neg.	neg.	neg.
Radiation Dose	Decommissioning Activities	man-rem	3063	1929	729	2518
	Transportation	man-rem	200	13[d]	13[d]	90
	Continuing Care	man-rem	--	120	225	neg.

(a) Radiation doses from postulated industrial accidents are not included.
(b) 50-yr committed dose equivalent to the bone, for the total population within 80 km of the site.
(c) 50-yr committed dose equivalent to the total body, for the population along the transport route.
(d) Includes only preparations for safe storage.
(e) neg. = negligible. Impacts of continuing care expected to be negligible compared to those of decommissioning activities.

TABLE 2.11-4. Summary of Safety Analysis for Decommissioning at the Reference BWR Following the Cleanup of a Scenario 2 Accident

Type of Safety Concern	Source of Safety Concern	Unit	DECON	SAFSTOR with Deferred Decontamination after: 30 Years	100 Years	ENTOMB
	Public Safety[a]					
Radiation Dose	Decommissioning Activities[b]	man-rem	<0.05	<0.05	<0.05	<0.04
	Transportation[c]	man-rem	16	2[d]	2[d]	7
	Continuing Care	man-rem	--	neg.[e]	neg.	neg.
	Occupational Safety					
Serious Lost-Time Injuries	Decommissioning Activities	total no.	1.9	2.2	2.2	1.7
	Transportation	total no.	1.9	1.9	1.9	0.87
	Continuing Care	total no.	--	neg.	neg.	neg.
Fatalities	Decommissioning Activities	total no.	0.0094	0.012	0.012	0.0085
	Transportation	total no.	0.11	0.11	0.11	0.051
	Continuing Care	total no.	--	neg.	neg.	neg.
Radiation Dose	Decommissioning Activities	man-rem	3181	2417	1137	2531
	Transportation	man-rem	170	24[d]	24[d]	78
	Continuing Care	man-rem	--	65	120	neg.

(a) Radiation doses from postulated industrial accidents are not included.
(b) 50-yr committed dose equivalent to the bone, for the total population within 80 km of the site.
(c) 50-yr committed dose equivalent to the total body, for the population along the transport route.
(d) Includes only preparations for safe storage.
(e) neg. = negligible. Impacts of continuing care expected to be negligible compared to those of decommissioning activities.

2.12 IMPACTS OF TEMPORARY INABILITY TO DISPOSE OF WASTES OFFSITE

A basic assumption of this study is that all radioactive waste materials from post-accident decommissioning and from accident cleanup that precedes the decommissioning are disposed of offsite. If offsite waste disposal capability were to be unavailable during this period because of technical, regulatory, or political constraints, the selection of the preferred alternative for completing the decommissioning could be affected. Of the three alternative approaches to decommissioning following accident cleanup (i.e., DECON, SAFSTOR, or ENTOMB), both SAFSTOR and a partial DECON[a] appear to be practical if temporary onsite storage of accident cleanup and decommissioning wastes were to be necessary. DECON of the total facility and ENTOMB both have characteristics that appear to make them generally unsuitable for post-accident decommissioning with onsite waste storage.

The costs and safety impacts of the need for extended onsite storage of radioactive wastes and/or spent fuel from accident cleanup and decommissioning at the reference PWR are discussed in Chapter 15. A basic assumption of the analysis in Chapter 15 is that if offsite disposal capacity is available (or unavailable) for a particular waste form, this condition exists for both accident cleanup and decommissioning wastes. An additional assumption is that the costs directly associated with interim storage of spent fuel (i.e., the costs of operating and maintaining the spent fuel pool and of the operating staff and security force during the safe storage period) are operational costs and hence are not included in the estimates of decommissioning costs. The conclusions of Chapter 15 are believed to be valid for both the reference PWR and the reference BWR.

Based on the assumptions used in this study, onsite storage of low-level waste is estimated to have virtually no effect on the total costs (in constant dollars) of post-accident cleanup and decommissioning. However, some costs

(a) In partial DECON, portions of the facility not required for waste storage would be decontaminated to levels that permit unrestricted use. A nuclear license would be retained for buildings not decontaminated to unrestricted use levels and wastes from accident cleanup and decommissioning activities would be packaged and stored inside these buildings.

that would normally occur prior to the safe storage period are delayed until deferred decontamination. Onsite storage of process solid wastes and of spent fuel increases the total cost of accident cleanup and decommissioning by a few percent. The increased cost is due primarily to increased expenditures during preparations for accident cleanup if a separate facility must be provided for the filter/demineralizer system rather than installing it in the spent fuel pool and, for partial DECON, to deferred decontamination of the auxiliary and fuel buildings.

As indicated above, the costs directly associated with interim storage of spent fuel are considered operational and hence are not included in the study as decommissioning costs. These costs include the security force made necessary by the presence of spent fuel, the costs of operating personnel, normal maintenance, energy, equipment, supplies, insurance, and license fees. These costs are estimated in Chapter 15 to be approximately $1 million per year of storage. If there were additional design or maintenance problems associated with the storage of spent fuel, these would add to the operational costs of storage. These activities and their costs are outside the scope of this study.

Occupational radiation doses are estimated to be essentially unaffected by onsite waste storage. Radiation doses to transport workers and the public from offsite waste shipments are estimated to be reduced by about a factor of 2 for cases involving SAFSTOR with temporary onsite waste storage but to be essentially unaffected for the case of partial DECON with temporary onsite storage of spent fuel. It should be noted that the estimated safety impacts to the public of accident cleanup and decommissioning are small, with or without temporary onsite storage of radioactive wastes.

2.13 CONCLUSIONS AND RECOMMENDATIONS

Cleanup and decommissioning of a nuclear power reactor that has been involved in a serious accident is technically feasible. Many of the techniques and procedures used to decontaminate or decommission a reactor following an accident are similar to those used for decontamination or decommissioning following normal shutdown and may be used with proper consideration of problems involved in working in higher than normal radiation areas. If a reactor has experienced severe damage to the fuel core, special tools would likely be required to remove the fuel from the reactor vessel. These tools can be designed and fabricated using currently available technology.

Existing regulations can be used to cover post-accident cleanup and decommissioning. However, some modifications and/or additions that deal specifically with waste management requirements should be developed. Radioactive wastes from accident cleanup and decommissioning range from low-specific-activity trash and rubbish to high-specific-activity wastes, including loaded ion exchange materials, accident sludges, evaporator bottoms, and the activated reactor vessel and vessel internals. Some of these wastes may not be suitable for disposal by shallow-land burial. Regulatory attention should be given to defining waste disposal criteria that will minimize the impacts of management of high-specific-activity wastes on the costs and occupational exposures for accident cleanup and decommissioning.

This study shows that the costs of accident cleanup and decommissioning of a nuclear power reactor following a serious accident can be much greater than the costs of decommissioning following normal shutdown. In addition, uncertainties in the plant conditions following an accident, need for construction of special facilities and equipment, regulatory requirements, problems in obtaining adequate financing, and the delays resulting from all of these factors can significantly increase accident cleanup costs to amounts greater than the engineering costs estimated in this study. Property damage insurance appears to be a suitable mechanism for ensuring the availability of funds for post-accident cleanup and the increased costs of decommissioning above those estimated for normal decommissioning, and efforts should continue to be made to provide an adequate level of property damage insurance coverage for utilities that operate nuclear power plants.

Reactor accidents can result in high levels of radioactive contamination on building surfaces and equipment and in high radiation exposure rates to personnel involved in accident cleanup operations. In reactor accidents that have occurred to date, procedures have been devised to remove the radioactive contamination with only modest radiation doses to individual workers. These procedures have included the use of shielding and of remotely operated cleaning equipment and the careful planning and rehearsal of cleanup operations to limit the times that individual workers spend in high-radiation areas. This study shows that large numbers of personnel may be required for post-accident cleanup and decommissioning to ensure that radiation doses to individual workers do not exceed the occupational radiation dose limit.

For decommissioning following normal shutdown, the radioisotope that controls occupational exposure is ^{60}Co, with a 5.27-year half-life. For decommissioning following normal shutdown, SAFSTOR with deferred decontamination after continuing care periods of 30 to 50 years is an appropriate alternative to use to reduce occupational radiation doses from decommissioning. However, for post-accident cleanup and decommissioning, the radioisotopes that control occupational exposure are ^{90}Sr and ^{137}Cs, with 30-year half-lives. For these radioisotopes, a continuing care period of 30 years would result in a reduction in radiation exposure of only a factor of 2 and a continuing care period of 100 years would result in a reduction in exposure of only a factor of 10. (This can be compared to the factor of almost 10^6 reduction in exposure from ^{60}Co after a period of 100 years.) Therefore, SAFSTOR appears less attractive as an alternative for limiting occupational exposure from decommissioning following an accident than it is for limiting occupational exposure from decommissioning following normal shutdown, although factors discussed in Section 2.12 could make SAFSTOR necessary.

The public safety impacts of post-accident cleanup and decommissioning are estimated to be greater than the corresponding impacts from normal-shutdown decommissioning. The primary contribution to public dose is the airborne release of particulate radioactivity during accident cleanup. In all cases, radiation doses to the public from routine accident cleanup and decommissioning operations at the reference LWRs are estimated to be below permissible radiation dose levels in unrestricted areas.

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CHAPTER 3

REVIEW OF CLEANUP EXPERIENCE AT REACTORS THAT HAVE BEEN INVOLVED IN ACCIDENTS

This chapter provides a review of decontamination and/or decommissioning experience at reactors that have been involved in accidents. In addition to U.S. reactor experience, accident cleanup activities at two Canadian and one Swiss reactor are reviewed. Nearly all of the reactor accidents reviewed in this chapter have occurred at small experimental or research reactors. These accidents are reviewed because they provide useful information about cleanup experience.

The nuclear power industry in the United States has an excellent safety record. The few accidents that have occurred have not resulted in death or injury to any member of the general public or in significant radioactive contamination of inhabited areas. In several instances, reactor accidents have resulted in high levels of radioactive contamination on building surfaces and equipment. In all cases where contamination has occurred, methods and procedures have been devised to remove the contamination with only modest total radiation doses to decontamination workers.

Many of the techniques and procedures employed to decontaminate or decommission a reactor following an accident are similar to those used for decontamination or decommissioning following normal shutdown. The removal of unwanted radioactive contamination from materials and equipment is a familiar and routine operation in the nuclear field. Decontaminations of various types have been conducted since the 1940s and the basic technologies of decontamination are well established.(1) The decontamination and decommissioning of light water reactors following shutdown after normal operations is described in previous reports in this series.(2-4)

Post-accident decontamination requirements will vary with the type of reactor and with the nature and severity of the accident. Typical post-accident decontamination activities include:

- entry into highly contaminated areas
- removal and processing of large volumes of contaminated water
- removal of contamination from building and equipment surfaces
- removal of damaged fuel and/or damaged core internals from reactor pressure vessels
- cleanup following fuel spills
- decontamination of reactor coolant systems to remove large quantities of fission product radioactivity.

In most instances, the operations required to complete post-accident decontamination differ from those of normal-shutdown decontamination in magnitude (i.e., level of difficulty of performing the operation) rather than in type of operation.

3.1 CANADIAN NRX REACTOR

The Canadian NRX reactor is a light-water cooled and heavy-water moderated research reactor designed for operation at 10 MW (thermal). Natural uranium fuel rods are centered in water-filled aluminum process tubes immersed in a pool of heavy water that serves as a neutron moderator. The aluminum calandria containing the heavy water is surrounded by a graphite reflector and by a 2.5-m-thick concrete shield. On 12 December 1952, a power surge caused the rupture of the aluminum cladding on several fuel rods, resulting in fission product contamination of the coolant and the reactor building. The accident and subsequent decontamination efforts are described in References 5 through 7.

3.1.1 Accident Description

The power surge resulted in failure of the aluminum sheaths for about 10% of the fuel rods. Both oxidation and melting of the uranium fuel accompanied this failure. Approximately 10,000 curies of radioactive fission products were released into 3800 m^3 of cooling water that flooded the basement area of the reactor building. After removal of the water, dose rates of 10 R/hr were measured along concrete walls and floors in the lower header area and the basement area. Localized hot spots gave readings as high as 3000 R/hr. Dose rates of approximately 50 mR/hr were measured on the ceilings, walls, and floors of the upper part of the reactor building.

3.1.2 Decontamination Procedures

The contaminated water that collected in the reactor building basement was pumped through a pipeline to open pits in a disposal area 2000 m away. Here the water was allowed to seep through the soil for removal of the radioactivity.

After extensive decontamination of the basement area and other areas to reduce the radiation dose to workers, electrical and piping connections were removed in preparation for core removal. All undamaged fuel rods were first withdrawn from the reactor. Many of the damaged rods were fused together, and attempts to remove these rods were unsuccessful because of breakage. The concrete shields on top of the reactor and the top thermal shield were removed to provide greater access to the fuel rods. Several rods were removed using a cutting procedure. The remaining rods were grasped near the bottom and pulled out as a unit. All fuel rods were placed in standard shipping casks for transport to a fuel reprocessing plant.

The reactor vessel was next removed. The vessel was isolated from the cooling systems, rigged, lifted, placed on a skid, towed to a burial ground, and covered with sand for shielding. The entire removal task was rehearsed using a nonradioactive mockup of the reactor vessel. The trained crew was able to perform the difficult remote removal operation on the contaminated vessel in about 30 minutes.

The general procedure for decontamination of containment building surfaces was the following:

- flush with water and pump the water to the disposal area
- flush with a high-velocity stream of hot water, sometimes with detergents

- remove equipment for disposal or for decontamination at designated areas
- perform a third flushing with hot water and detergents
- protect clean surfaces from recontamination by covering with paper.

Concrete was decontaminated by removing the surface by flame priming, chipping, sand blasting, and grinding. Where grinding and sand blasting were used, the equipment was connected to a vacuum system for control of the dust. In some instances where concrete had been contaminated with radioactive water it was very difficult to reduce the contact dose rate below 20 mR/hr. For these cases, the remaining radioactivity was sealed in place and concrete was poured over the surface as a shield.

Stainless steel surfaces were decontaminated by scrubbing with cotton wipers soaked in detergents and acids. This method worked better than scrubbing with brushes and flushing. Mild steel responded fairly well to scrubbing with brushes and detergents. If rust was present, inhibited hydrochloric acid was used to remove the rust and the associated radioactive contamination.

By 1 February 1954, approximately 14 months after the accident, the reactor building was completely decontaminated and the reactor was reassembled and returned to service. Radiation levels in working areas of the building were measured to be lower than they had been before the accident.

3.1.3 Radiation Doses

The total occupational radiation dose for decontamination of the NRX reactor building was 700 man-rem, involving 600 workers. Only one decontamination worker received more than 15 rem during the 14-month period. This worker received 17 rem.

3.2 CANADIAN NRU REACTOR

The Canadian NRU reactor is an engineering and research reactor with a maximum power output of about 200 MW (thermal). The reactor core is comprised of plate-type fuel elements that are fabricated from natural uranium metal clad with aluminum and are cooled and moderated by heavy water. On 23 May 1958 the interior of the reactor building was severely contaminated when a section of ruptured fuel element fell from the refueling cask. The accident and subsequent decontamination efforts are described in References 8 through 10.

3.2.1 Accident Description

During operation of the reactor at full power, three fuel elements ruptured releasing high levels of fission product radioactivity into the coolant stream. After the reactor was shut down, one damaged element was successfully removed from the core to the refueling cask and transported to the fuel storage block. During transfer of the second failed element, coolant was lost, and a 1-m section of fuel element fell into the nearby maintenance pit.

As a result of the accident, the interior of the reactor building was severely contaminated. The section of element that fell into the maintenance pit was estimated to have contained 2×10^5 Ci of mixed fission products, 700 Ci of ^{131}I, and 13 g of ^{239}Pu. Over and around the maintenance pit, radiation fields were in excess of 1000 R/hr. Exposure rates of up to 1000 R/hr were measured on top of the reactor deck plate. Other exposure rates were 1 R/hr near the refueling cask, 2.5 R/hr near the west walls of the reactor hall, 100 R/hr at spots on the floor, and 30 mR/hr in offices, corridors, and change rooms.

During the accident, the main exhaust fan on the roof of the reactor building was operating. Although the fan was immediately shut off to confine the contamination, some contamination was later detected at distances of 1.5 km downwind. Contamination in buildings downwind from the reactor was found to be mainly on ventilation supply system filters that were easily replaced. The hottest spots of ground contamination measured about 10^{-2} $\mu Ci/cm^2$.

3.2.2 Decontamination Procedures

A team of men carrying buckets of wet sand from outside the reactor building buried the burning fuel element in the maintenance pit. To remove the fuel element from the maintenance pit, a wooden tray with a metal lip on one side was constructed, filled with sand, and lowered into the pit. A team of men using long-handled tools (rakes, hoes and shovels with 7.3-m handles) raked the rod segment onto the tray and covered it with sand. A working time of 1 minute per man was allowed. The loaded tray was lifted by an overhead crane and placed on a shielded trailer for removal to the waste disposal area. Because radiation fields in the crane cab were 5 to 10 R/hr, crane operators were allowed 2 minutes working time.

To further reduce the radiation fields in the reactor building so that decontamination of building surfaces could proceed, it was necessary to remove the remaining debris and contaminated sand in the maintenance pit. This work was performed by men whose normal work did not involve exposure to radiation, under the direction of an experienced employee. These men, working in pairs, used long-handled tools to shovel the sand into metal garbage cans and carried the cans to an area from which easy disposal could be made. A working time of 1.5 minutes was prescribed for each man.

Decontamination of the reactor building was performed by workers who wore plastic suits, full-face respirators, rubber gloves, and rubber overshoes. Because body heat made the plastic suits uncomfortable to wear for long periods, workers were given a 20 to 30 minute rest after 1 to 2 hours of work. A public address system and closed-circuit television were used to give instructions and to monitor the work effort. Decontamination procedures included vacuuming, wet mopping, and wiping with damp rags soaked in detergent. Some equipment was decontaminated at a special location where a large metal-parts washer, an ultrasonic machine, and a commercial drum-type laundry machine were used.

After the reactor was defueled, visual inspection of the interior of the reactor vessel showed that it contained a broken piece of fuel rod and other pieces of debris. A container was lowered into the reactor vessel and a special tool was used to transfer the broken rod segment into the container. Other rod fragments were picked up with mechanical fingers. It was found that the smallest fragments would adhere to plasticine. A special cylindrical tool was made from which plasticine could be extruded and then sliced off after it had picked up a load of debris.

Some contamination was found on the roads used for transportation of the debris from the maintenance pit. The roads were cleaned by vacuuming, washing with fire hoses, and, when necessary, removing part of the road surface.

After decontamination of the reactor building and removal of bits of fuel from the reactor vessel, the reactor was refueled and resumed operation at the end of August 1958, some 3 months after the accident.

3.2.3 Radiation Doses

The average radiation dose received by workers during removal of the fuel element segment from the maintenance pit was 1.4 rem and the maximum individual dose was 6.4 rem. Great care in the timing of exposures and the use of experienced radiation workers were the main factors in keeping radiation doses low. The highest radiation doses were received by workers engaged in removing fuel debris and contaminated sand from the maintenance pit. Fourteen men exceeded the 5-rem limit, with 11 men receiving doses in the range from 5 to 6 rem, 2 men receiving about 10 rem, and one man who made an unauthorized trip with a can of sand receiving 19 rem.

Table 3.2-1, reproduced from Reference 10, gives a record of radiation doses to decontamination workers for the period of 24 May to 29 June 1958. The data in the table can be used to estimate bounding values for the average radiation dose to workers from cleanup operations at the NRU reactor. The average occupational radiation dose is in the range from 0.3 rem to 1.0 rem.

TABLE 3.2-1. Occupational Radiation Doses to Decontamination Workers from Decontamination of the NRU Reactor for the Period May 24-June 29, 1958[a]

	Number of Exposures at Various Dose Ranges					
	0.1-0.3 rem	0.3-1 rem	1-3 rem	3-5 rem	5-10 rem	10-20 rem
Company Personnel	492	127	58	28	16	3
Off-Project Personnel	323	112	46	9	0	0

(a) From Reference 10.

3.3 SL-1 REACTOR

The SL-1 reactor, a military reactor undergoing tests at the National Reactor Testing Station (NRTS) in Idaho, was a direct cycle, natural recirculation boiling water reactor designed for 3000 kW (thermal) and capable of producing 200 kW of electricity and 1.3 million Btu/hr for space heat. The reactor was developed in response to a Department of Defense request for a plant that could be used at remote military installations. On 3 January 1961 a nuclear excursion occurred that resulted in extensive damage to the reactor. The accident and subsequent decontamination efforts are described in References 11 through 14.

The reactor was located in a cylindrical building fabricated from steel plate with a thickness of 6.4 mm (1/4 in.). Access to the building was provided by ordinary doors. The building was not a pressure-type containment shell such as is used for commercial nuclear power reactors. Nevertheless, the building was able to contain most of the radioactivity released during the accident.

3.3.1 Accident Description

At the time of the accident, the reactor was shut down for routine maintenance, minor plant modifications, and the installation of flux wires in the core. On 3 January 1961 the three-man 4 to 12 p.m. shift was directed to reassemble the control rod drive mechanism and to prepare the reactor for the resumption of operations the following morning. A nuclear excursion was apparently caused by manual withdrawal, by one or more of the maintenance crew, of the central reactor control rod considerably beyond the critical position.

The nuclear excursion increased fuel plate temperatures to points near or above melting and caused a large steam void to form in the center of the core. The steam pressure apparently forced a slug of water to impact the reactor vessel head resulting in a water hammer phenomenon with pressures probably as high as 10,000 psi. The momentum of the water slug was transferred to the reactor vessel causing the vessel to be projected upward sufficiently to shear the steam nozzle and water lines and to drive some concrete insulating blocks from around the reactor onto the operating room floor. Subsequently, the reactor fell back approximately to its original position. The reactor core and pressure vessel were damaged beyond repair. The reactor building incurred only minor damage but was grossly contaminated. Some gaseous fission products, including radioactive iodine, escaped to the atmosphere outside the building and were carried downwind in a narrow plume.

Particulate fission material was largely confined to the interior of the reactor building, with slight radioactivity in the immediate vicinity of the building. Radiation levels within the reactor room a few hours after the accident measured approximately 500 to 1000 R/hr. Readings of 10 R/hr were observed in the reactor control room, and readings of 2.5 R/hr were measured outside the reactor building at a distance of about 8 m from the building. Two days after the accident, radiation readings of 250 mR/hr to 700 mR/hr were measured at various points along with SL-1 area fence.

3.3.2 Decontamination and Decommissioning Procedures

Initial decontamination efforts following the accident were directed at making radiation measurements, surveying the damage in the reactor room, and determining the condition of the reactor. Because of the high radiation levels inside the building, personnel access was restricted.

A mockup of the reactor building and reactor vessel was used to train crews in the manipulations required to handle photographic and television cameras and to operate and position the equipment used for radiation and other measurements inside the building. Some observations and measurements were made remotely by using the shielded crane. Photographic, motion picture, and television pictures, together with radiation measurements, were used to plan the decontamination effort. It was determined that the pressure vessel contained no water and that subsequent nuclear excursions could be prevented by keeping the vessel dry. Several thousand kilograms of steel and lead sheet and lead shot were installed remotely over the reactor head to reduce the radiation level and allow personnel access to the reactor building.

All operations were planned with the requirement of keeping water from entering the reactor. Manual and remote cleanup were used. Tanks were drained and all loose items and radioactive debris were removed from the reactor building. The removal of equipment and debris was accomplished remotely by using an electromagnet and manually by vacuuming and sweeping. The use of manual labor required a rapid, large-scale turnover rate to avoid overexposure to any individual.

In November 1961 the pressure vessel, with the core left inside, was removed from the reactor building and transported in a concrete shipping cask to a large disassembly hot cell located 40 miles from the SL-1 site. Hot cell examination of the pressure vessel disclosed that the vessel was not ruptured but was bulged below the head flange and above and below the core. The reactor head nozzles were also found to be bulged. The pressure vessel flange was so distorted that the head could not be raised off the head bolts after the nuts were removed. It was necessary to force the head upward using wedges.

After the reactor vessel was removed, decommissioning of the reactor building proceeded. Equipment and wall sections were cut out and sent to a disposal area. Concrete support columns were removed using a bulldozer, and the steel reinforcing rods were cut. In slightly contaminated adjoining buildings, wiping, vacuuming, and the use of surface-strippable films were sufficient for decontamination.

3.3.3 Radiation Doses

Of the several hundred people engaged in recovery and decommissioning operations, most received gamma radiation doses of less than 300 mrem. Twenty-two persons received total-body gamma radiation doses in the range of 3 to 27 rem.

3.4 PLUTONIUM RECYCLE TEST REACTOR (PRTR)

Fuel element rupture (i.e., cladding failure) has occurred in operating nuclear reactors. Generally, cladding failure results only in an increase in fission product radioactivity in the reactor coolant but no significant release of radioactivity to the containment building or to the environment. Decontamination is accomplished by removing the damaged element from the reactor core and by flushing the coolant system to remove some of the excess radioactivity. The reactor then continues in operation.

Decontamination of the Plutonium Recycle Test Reactor (PRTR) following the rupture loop failure incident of 29 September 1965, is described in this section. This incident, which is discussed in References 15 through 17, resulted in radioactive contamination of the reactor coolant system and in a release of fission product aerosol to the containment building atmosphere.

The PRTR is a heavy-water moderated and cooled test reactor. Fuel elements are centered in process tubes and are cooled with recirculating heavy water at 1050 psi and 250-280°C. Surrounding each process tube is a shroud tube which separates it from the low-temperature unpressurized heavy water moderator contained in the reactor calandria. Helium gas flows in the space between the process tube and the shroud tube in a low-pressure dry gas system that contains rupture discs to allow venting to the containment building atmosphere in case of overpressurization. The center process tube of the reactor is used as a fuel element rupture test facility and is cooled with light water supplied through an independent system.

3.4.1 Accident Description

On 29 September 1965 a purposely defected, partially molten fuel rod in the rupture loop test facility failed in an unexpected manner. The rupture, which resulted in a loss of about 39% of the UO_2 - PuO_2 fuel from the rod, was accompanied by the formation of a hole of approximately 12 mm diameter in the surrounding process tube. Failure of the loop pressure tube resulted in the flashing of the highly contaminated superheated water to the helium-filled annular space between the pressure tube and the aluminum calandria and in a release of fission product aerosol to the containment building atmosphere. This release resulted in about half of the noble gases, 1% of the radioiodine, and a somewhat smaller fraction of the solid fission products from the failed rod entering the containment building atmosphere.

The highly contaminated light water coolant that escaped through the rupture in the process tube flooded the low-pressure helium system and flowed into the moderator space in the calandria, apparently through poor seals on the many penetrations on the top face. It eventually filled the moderator dump tank, and the water level began to rise in the calandria. A rupture disk broke and allowed the calandria to drain onto the containment building floor and into containment building sumps.

The reactor primary and secondary coolant systems continued to function normally during and following the rupture loop failure. The temperature and pressure of both systems were allowed to decrease slowly following the reactor scram.

Approximately 7 curies of ^{131}I, in addition to other fission products, were released to the containment building atmosphere; however, only a small fraction of these were deposited on containment building surfaces. About 99% of the fission product aerosol and about 70% of the radioiodine were removed in the condensate from eight building air coolers that circulated air at a rate of 700 m^3/min (or 6% of the containment building's atmosphere per minute). Had the air coolers not been in operation, essentially all of the radioactive aerosol would presumably have deposited on surfaces in the containment building, and the general contamination level would have been 100-fold higher.

Ionization chambers in the containment building indicated radiation levels in excess of 20 rem/hr in the reactor hall immediately following the accident. After 17 hours, the radiation level had decreased to 4 rem/hr. No significant release of contaminants to the environment occurred, but radiation levels from the contained fission products were measured at 20 to 50 mrem/hr at 100 m from the containment building 2 hours following the accident.

3.4.2 Decontamination Procedures

Containment building air samples taken 17 hours after the radioactivity release indicated that the airborne radioactivity had decreased to the point where it was safe to purge the building. Flushing of containment building air through HEPA and charcoal bed filters for 2 days at 280 m^3/min resulted in a 10,000-fold decrease in airborne radioiodine and a 25,000-fold decrease in the radioxenon content of the air.

Decontamination of the reactor hall and the reactor face was accomplished by flushing with water and swabbing with water and detergent. Contaminated water in the containment building sumps was pumped out for disposal as radioactive waste. The failed fuel element was successfully removed from the rupture loop 1 month after the accident.

Decontamination of the rupture loop coolant system was performed using the following operations: First, mechanical removal of debris was accomplished by high-velocity flushing and filtration followed by draining and flushing of deadlegs. Chemical removal of oxide films was then performed using two applications of buffered oxalic-peroxide compounds. Finally, alkaline permanganate followed by inhibited oxalic acid was used to remove residual fission product and corrosion product activities. The final contact radiation dose rates were near 5 mrem/hr and the entire decontamination sequence required 10 days.

While the reactor was shut down following the rupture loop failure incident, a decision was made to decontaminate the primary coolant system. (The primary system is entirely independent of the rupture loop system and was largely unaffected by rupture loop failure.) Decontamination of the PRTR primary system had been performed following a fuel element rupture in 1962, using alkaline permanganate followed by 10% oxalic acid.[18] Although the films were removed, the oxalic acid formed an oxalate which precipitated and carried down some of the contamination. To achieve a satisfactory decontamination factor, it was necessary to remove the precipitate with dibasic ammonium citrate solution. Later development work showed that satisfactory defilming without precipitation could be obtained by combining the oxalate and citrate reagents.

To prepare the primary system for decontamination, all of the fuel elements were removed from the reactor core and placed in the storage basin. The D_2O was pumped from the system into storage tanks and barrels. Several light water rinses were used to insure complete collection of the D_2O. Some mechanical changes were made to provide system protection and to allow the decontamination to proceed with minimum delay. Decontamination was effected by use of alkaline permanganate[(a)] followed by an inhibited citrate-oxalate reagent (Citrox).[(b)]

The alkaline permanganate solution was pumped into the system, heated, and circulated for 2 hours. The solution was then drained and the system was rinsed with demineralized water. Citrox was pumped into the system, heated, and circulated for about 4 hours. After the Citrox was drained, the system was rinsed with demineralized water, then with degraded D_2O, then filled with D_2O. The entire flushing operation required approximately 3 days for completion.

Radioactivity levels were measured at various points on the primary system before and after the decontamination. Decontamination factors were uniformly high, ranging from 11 to 44, with an average of 25 (96% removal of contamination).

3.5 ENRICO FERMI-1 REACTOR

The Enrico Fermi-1 reactor was an experimental fast breeder reactor that was cooled by liquid sodium and operated at essentially atmospheric pressure. A secondary sodium system transported the heat from the primary sodium to steam generators where steam was produced for a conventional turbine-generator unit.

On 5 October 1966 a fuel melting incident occurred, caused by partial blockage of the coolant flow to several core subassemblies. The accident is described in Reference 19.

3.5.1 Accident Description

The coolant blockage was caused by the detachment of a segment of the zirconium liner from the conical flow guide. The detached segment blocked the coolant flow to four fuel subassemblies, resulting in partial melting of two of the subassemblies. An estimated 10,000 Ci of fission products were released

(a) Alkaline permanganate: 10 to 18% NaOH + 3% $KMnO_4$.
(b) Citrox: $H_2C_2O_4$ (25 g/liter), $(NH_4)_2\ HC_6H_5O_7$ (50 g/liter), $Fe_2(SO_4)_3 \cdot 9H_2O$ (2 g/liter), and diethylthiourea (1 g/liter).

to the primary coolant and reactor cover gas. Most of the released activity was deposited by plateout in the cold trap, a device for removing oxides and hydrides from the primary system coolant. The only release of fission products from the primary coolant system was in the form of inert fission-product gases to the reactor containment building atmosphere by way of leaking through the primary system argon-cover-gas seals.

3.5.2 Decontamination Procedures

To minimize leakage of fission gases to the containment building atmosphere, the cover gas pressure was reduced and the system was purged to the waste-gas holdup tanks.

Following the removal of the damaged fuel elements and a complete drain of sodium from the primary system, efforts were made to discover the cause of the coolant-flow blockage. Special optical instruments and remotely operated tools were developed to locate and remove the detached segment and the five remaining zirconium segments from the reactor. During the remote recovery operations, the fuel subassemblies were modified with flow guards to prevent the recurrence of such an accident.

After decontamination of the primary coolant system, the reactor was refueled and returned to service. Full-power operation was achieved during October 1970. The plant was shut down in October 1972, and decommissioning of the reactor via the SAFSTOR alternative was initiated.[20]

3.6 LUCENS EXPERIMENTAL REACTOR

The experimental nuclear power station at Lucens was a CO_2-cooled, D_2O-moderated experimental reactor built in an underground cavern in a hill about 2 km from the city of Lucens, Switzerland. The reactor had a nominal power of 30 MW (thermal). The reactor fuel elements consisted of slightly enriched uranium rods clad with a Mg-Zr alloy and situated in channels in graphite rods that were centered in Zircaloy pressure tubes, cooling the fuel rods. The pressure tubes were located in an aluminum calandria that contained the D_2O moderator.

On 21 January 1969 an accident occurred in which one of the fuel elements overheated and its pressure tube ruptured. The accident is described in Reference 21. The decommissioning of the reactor following the accident is described in Reference 22.

3.6.1 Accident Description

The immediate cause of the accident was water that entered the CO_2 coolant system through a defective shaft gasket, causing corrosion of the cladding of several fuel elements. Corrosion products settled at the bottom of coolant channels, partially blocking the coolant flow. Interruption of the coolant flow resulted in the melting and subsequent ignition of one of the fuel elements and in the rupture of the pressure tube separating the CO_2 coolant from the heavy water moderator. The reactor coolant expanded into the moderator tank, increasing the pressure in the tank and causing the fracture of its pressure disks. Contaminated steam then entered the reactor cavern, carrying with it fission products, vaporized fuel fragments, and a large portion of the heavy water moderator.

Initially, a gamma dose rate of about 120 rads/hr was measured in the upper chamber of the reactor cavern. Some airborne radioactivity leaked from the reactor cavern to the adjoining plant cavern and to the control room through pipe and cable penetrations. This radioactivity consisted mainly of gaseous fission products with traces of radioactive iodine. After analyses of the atmosphere in the reactor cavern, the cavern was vented through the stack via iodine filters.

The accident destroyed one fuel element and seriously damaged the core of the reactor so that a decision was made to decommission the facility.

3.6.2 Decommissioning Procedures

Decommissioning of the Lucens reactor was started shortly after the accident and was substantially completed by the end of May 1973. The period from January 1969 to March 1970 was used for initial decontamination of the reactor cavern, for recovery of as much of the heavy water as possible, and for remote inspection (using endoscopes) of the region within the biological shield where most of the damage was concentrated. This was followed by disassembly and removal of the pressure tubes that carried the CO_2 coolant to the reactor, additional decontamination of the reactor cavern, and fixing some contamination by painting. The reactor was defueled (except for the damaged fuel element), and the pressure tubes were cut and removed. The reactor vessel was then sectioned and packaged for disposal. After the upper portion of the reactor vessel had been removed, the damaged fuel element and its pressure tube were recovered.

Final steps in the decommissioning included the disassembly of the highly contaminated CO_2 refrigeration system and the decontamination and disassembly of the station for the treatment of radioactive material. Final decontamination was completed during the last half of 1972 and the first half of 1973. Decommissioning operations were hampered by the confined space within the reactor cavern in which operations had to be carried out.

3.6.3 Radiation Doses

A total of 136 man-years of decommissioning worker effort was required for inspection of the damage to the facility, defueling of the reactor, and decommissioning following the defueling. The cumulative whole-body radiation dose received by these workers was approximately 132 man-rem.

3.7 THREE MILE ISLAND NUCLEAR STATION, UNIT 2

Unit-2 of the Three Mile Island Nuclear Station (TMI-2) is a 2770-MW (thermal) pressurized water reactor that began commercial operation in January 1979. On 28 March 1979 TMI-2 experienced a loss-of-coolant accident that was the result of a unique combination of equipment failure, design deficiencies, and operator errors. The reactor was operating at 97% of full power when the accident occurred.

The accident at TMI-2 has been the subject of widespread public and media interest,[23-25] and of several government-sponsored investigations.[26-27] Accident cleanup activities are proceeding as this report is being written.

3.7.1 Accident Description

On the morning of 28 March 1979 some feedwater pumps in the secondary system stopped operating, resulting in a loss of secondary coolant and the automatic shutdown of the reactor. An increase in pressure within the reactor vessel activated the pilot-operated relief valve (PORV) on the pressurizer and the pressure in the primary system was reduced. However, after the pressure had been reduced, the PORV failed to close. A faulty indication on a light in the control room led the operators to assume wrongly that the valve had closed. For more than 2 hours, loss of coolant continued through the valve, resulting in the release of about 3800 m^3 of highly contaminated water. The coolant overflow tank ruptured, and radioactively contaminated steam was released to the containment building. Water collected in the basement and sumps of the containment building and overflowed to the auxiliary and fuel handling building (AFHB) where it contaminated the floors, walls, and storage tanks in that building.

The continual loss of primary coolant through the failed relief valve and the reduction of pressure in the system, coupled with the operator's shutting off of the high-pressure injection system (which had been adding water to the primary system) resulted in a drop in water level, uncovering some of the reactor core. This produced core temperatures in excess of 2000°C. The reaction of Zircaloy cladding with water vapor and steam resulted in cladding tube rupture and the release of fission product radioactivity to the primary coolant and to the containment building atmosphere.

Shortly after the accident, xenon and krypton gases and iodine accounted for most of the radioactivity in the containment building atmosphere, but these decayed to nonradioactive forms within a few months. Ultimately, about 43,000 Ci of ^{85}Kr remained in the containment building atmosphere, and this was purged to the outside atmosphere during the period between 28 June to 11 July 1980.

At the conclusion of the accident, about 2600 m^3 of contaminated water was standing 2.4 m deep in the containment building basement. This water was estimated to contain about 5 x 10^5 Ci of radioactivity (primarily ^{134}Cs, ^{137}Cs, and ^{90}Sr).

Estimates of the total amount of plateout to be removed from walls and other exposed surfaces of the containment building have ranged from less than 100 Ci to more than 1000 Ci. The principal radionuclides in the plateout were estimated to be ^{134}Cs, ^{137}Cs, and ^{90}Sr.

Personnel entry into the containment building was not made for more than a year following the accident. Average gamma radiation levels were estimated to be 250 mR/hr at the 347-ft elevation (the operating floor level) and 500 mR/hr at the 305-ft elevation (the personnel entry elevation), based on measurements made during the summer of 1980. Higher radiation levels were found at localized "hot spots" where concentrations of plateout occur and above the open stairwell, which is not protected by an intervening floor or wall from radiation from the water in the basement of the building. Typical measurements of gamma radiation in such areas were 0.5 R/hr at the air coolers, 2-5 R/hr at the floor drains, 10 R/hr over the metal deck for the covered floor hatch, and 18 R/hr at the open stairwell.(28)

The auxiliary and fuel handling buildings (AFHB) were also contaminated by water and gas from the primary coolant system and from the containment building.(28) Approximately 1400 m^3 of contaminated water was contained in tanks and sumps in the auxiliary building. Surveys of general access areas (corridors and normally nonrestricted areas) shortly after the accident showed radiation levels of 150 to 500 mR/hr in the fuel handling building and 50 to 5000 mR/hr in the auxiliary building. At certain locations (referred to as "hot spots") and in individual areas (cubicles) containing contaminated filters, demineralizers, tanks, and pumps, the radiation levels were much higher, ranging up to 100-1000 R/hr. (Levels of 1000 R/hr were estimated for the reactor coolant bleed holding tank cubicles.) The auxiliary building also contained approximately 4 m^3 of sludge with about 9000 Ci of radioactivity.

3.7.2 Decontamination Procedures

Cleanup of the AFHB began in April 1979 and was largely completed by the end of 1981. Decontamination of radioactive water in the auxiliary building tanks and sumps was accomplished by the use of a demineralizer system, designated as EPICOR-II.[29] The treated water is being stored onsite pending a decision on its final disposal. Several m^3 of radioactive wet sludge from storage tanks and sumps in the AFHB remain to be processed.

Decontamination of surfaces in the AFHB started with the general areas where radioactive contamination was relatively small and proceeded to cubicles containing tanks and other equipment that were more heavily contaminated. Methods used for decontaminating building and fixed equipment surfaces included washing with a high-pressure water jet, wet and dry vacuuming, and manual wiping. Clean surfaces were sometimes protected from recontamination by the application of strippable coatings. Small demountable equipment items were cleaned by electrochemical or ultrasonic techniques. Surface decontamination was about 80% complete by the end of 1981.

Decontamination of the TMI-2 containment building is projected to require the following steps:[28]

- removal of contaminated water from the reactor building basement
- washdown of building surfaces using water jets
- draining and flushing of sumps
- installation of local shielding
- defueling of the reactor pressure vessel
- dissolution of remaining fuel fragments
- hands-on decontamination of building and equipment surfaces
- chemical decontamination of the reactor coolant system
- solidification and packaging of all dispersible radioactive wastes, including radioactive water.

During 1981, entries were made into containment to measure contamination levels and radiation exposure rates, evaluate proposed decontamination procedures, survey damage to equipment, and make minor repairs to essential items. The processing of contaminated water in the containment building basement (sump water) began in September 1981 using the submerged demineralizer system (SDS) that was installed in the spent fuel pool in the fuel handling building.[30,31] The SDS system uses zeolite resins, which can accommodate loadings in excess of 7×10^5 Ci/m^3, rather than the organic resins used in EPICOR-II, which normally accommodate loadings of about 1400 curies per m^3. Processing of contaminated sump water is expected to be completed in the spring of 1982 and will be followed by initial decontamination of the containment building surfaces to allow worker access for defueling operations. Cleanup of the containment building is anticipated to require 4 to 6 years for completion, depending on regulatory requirements and on the availability of funds. (Proposed mechanisms for funding the cleanup of TMI-2 are described in Section 6.3.)

3.7.3 Radiation Doses

As of 1 September 1980 about 280,000 man-hours had been expended on AFHB decontamination by workers who plan, prepare for, and carry out the actual decontamination work, such as operating the water jet and scrubbing and wet vacuuming. The cumulative whole-body dose received by these workers was 142 man-rem, and the highest dose received by an individual decontamination worker was 2.5 rem. The total cumulative dose during the period from 27 April 1979 through 1 September 1980 for all personnel with assignments that required them to enter contaminated areas within the AFHB was about 250 man-rem. Estimates of the total cumulative whole-body occupational dose when decontamination of the AFHB is completed range between 375 and 550 man-rem.[28]

During the period between 23 July 1980 to 11 December 1980, six entries were made into the containment building by teams of personnel to conduct radiation surveys and inspect for damage resulting from the accident. Entry times varied from 20 minutes to 2 hours. The maximum whole-body dose received by an individual during these entries ranged from 220 mrem for the entry of July 23 to 650 mrem for the entry of December 11.

The average exposure rate for decontamination of the containment building is projected to be higher than that for decontamination of the AFHB--possibly 10 to 20 times higher.[28] After the initial decontamination of containment building surfaces to reduce worker dose rates, fuel removal activities will begin. The estimated time-averaged radiation dose rate for a typical worker during defueling activities over a work shift is 10 mrem/hr. For some hands-on activities such as removing solid wastes from the building or unbolting the water-tight seal it may be necessary to work for short periods in radiation fields as high as 150 to 200 mrem/hr. The estimated cumulative occupational dose for cleanup of the containment building is expected to be in the range from 660 to 3000 man-rem.[28]

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CHAPTER 4

STUDY APPROACH AND KEY STUDY BASES

This report describes the post-accident decommissioning of LWRs, including the post-accident cleanup. The overall approach used in the study is discussed in this chapter. Accident cleanup and decommissioning activities are outlined in Section 4.1. The study bases (i.e., ground rules) established to guide the study are given in Section 4.2. The technical approach used in the study is described in Section 4.3.

4.1 ACCIDENT CLEANUP AND DECOMMISSIONING ACTIVITIES

Reactor accidents of the types postulated in this report could result in severe radioactive contamination and potential physical damage to the containment structure and other buildings, the accumulation of contaminated water on floors and in building sumps, damage to the fuel core, and the accumulation of fuel debris in the reactor coolant system (PWR) or reactor water recirculation system (BWR). Details of the accidents considered in this study are discussed more fully in Chapter 8.

The first activities following an accident are designed to bring the accident under control and to stabilize the plant. Once the situation is stabilized, accident cleanup can begin. Accident cleanup is followed by either decommissioning or refurbishment of the facility. The sequence of accident cleanup and subsequent decommissioning or refurbishment activities for a nuclear power reactor that has been involved in a serious accident is shown in Figure 4.1-1. As discussed in Section 4.1.2, the accident cleanup activities are necessary and would be similar whether the reactor is ultimately refurbished or decommissioned.

This study does not treat the details of the activities that would be needed to stabilize the facility or to refurbish and restart the reactor. Accident cleanup and decommissioning activities, which are analyzed in detail in the various chapters of this report and which are described briefly in the

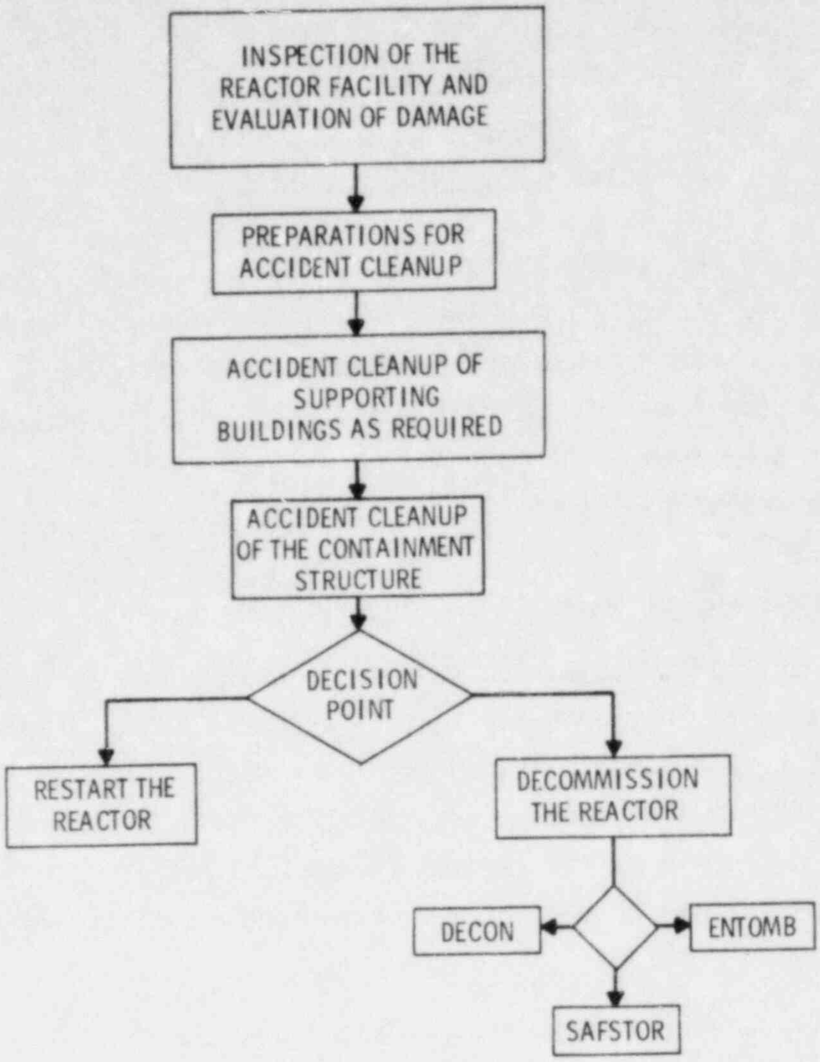

FIGURE 4.1-1. Postulated Sequence of Accident Cleanup and Decommissioning Activities for a Reactor Involved in a Serious Accident

following subsections, include: 1) preparations for accident cleanup, 2) accident cleanup of the containment and of other buildings as required, and 3) decommissioning of the facility by the DECON, SAFSTOR, or ENTOMB alternative.

4.1.1 Preparations for Accident Cleanup

A period of planning and preparation precedes the actual performance of cleanup operations within the accident-damaged reactor facility. Several tasks must be performed during this period to ensure the proper performance and the success of the operational phase of accident cleanup. These tasks include:

- venting of radioactive gases (i.e., krypton-85)
- containment entry and data acquisition
- preparation of documentation for regulatory agencies
- design, fabrication, and installation of special equipment
- development of detailed work plans and procedures
- selection and training of accident cleanup staff
- removal of accumulated spent fuel from the spent fuel storage pool.

In addition to these tasks, cleanup of buildings and systems outside of containment (specifically, the auxiliary and fuel buildings for the PWR and the radwaste building for the BWR) would be undertaken if these buildings were contaminated.

The tasks postulated for the planning and preparations phase are estimated to require from 1 to 3 years for completion, depending on the severity of the accident, the time needed to design, fabricate, install, and test special facilities and equipment, and the time required to secure the necessary regulatory approvals to proceed with the cleanup operations.

4.1.2 Accident Cleanup Activities

The operational phase of the cleanup and decommissioning of an accident-damaged nuclear power reactor begins with an accident cleanup campaign with three principal goals:

1) to reduce the initial high levels of radioactive contamination present on building surfaces and in accident water, thereby reducing the radiation dose received by workers engaged in cleanup and decommissioning operations

2) to collect and package for disposal the large quantities of water-soluble and otherwise readily dispersible radioactivity present in the plant

3) to safely defuel the reactor, placing the fuel in a configuration that is safe from nuclear criticality and/or fuel meltdown.

To achieve these goals, the accident cleanup campaign is postulated to include the following tasks (not necessarily in this order):

- processing of the contaminated water generated by the accident (and by decontamination operations) to remove radioactive materials
- initial decontamination of building surfaces and decontamination or disposal of some equipment
- removal of spent fuel (undamaged and damaged) from the reactor vessel and storage of the fuel in the spent fuel pool
- cleanup of the reactor coolant system or reactor water recirculation system
- solidification and packaging of wastes from accident cleanup operations.

Details of accident cleanup tasks are presented in Appendix E of Volume 2. The objective of reactor defueling is to remove all fuel and damaged reactor parts from the reactor pressure vessel so as to eliminate potential for further damage to the plant and to reduce radiation exposures. The accident cleanup activities of defueling the reactor and processing accident water are considered major steps in the cleanup process following a reactor accident and must be performed regardless of whether the reactor is to be restarted or decommissioned. Accident cleanup activities must establish conditions within the damaged plant that allow defueling to proceed. Some initial decontamination of contaminated structures must be performed to permit reasonable occupancy times without excessive exposure to workers engaged in defueling operations.

Because the processing of accident water and defueling of the reactor must be accomplished even if a decision is made to refurbish rather than decommission an accident-damaged reactor, in Figure 4.1-1 a decision point relating to reactor restart or decommissioning is shown following the completion of accident cleanup. This decision point could be earlier, but an early decision to restart would probably have minimal impact on the requirements for accident cleanup. Some of the decontamination procedures and the chemical solutions used for cleanup might be different because of the effort to minimize the effect of cleanup operations on a plant that was being returned to operation. However, whether accident cleanup is to be followed by restart of the reactor or by decommissioning, the use of highly corrosive chemicals would probably not be desirable because such chemicals are too destructive. Destructive chemical solutions could compromise the integrity of systems which need to remain intact during operations or decommissioning activities, especially if an extended decommissioning alternative, such as SAFSTOR, is chosen. Highly corrosive chemicals can also have an adverse effect on personnel using them.

Chemical decontamination of the reactor coolant system is considered part of accident cleanup because it preserves all the options for facility decommissioning in a way that is quite effective in terms of radiation exposure reduction. In addition, for the SAFSTOR decommissioning alternative, reactor coolant system decontamination should probably be done early, before the start of the period of safe storage. Deferral of reactor coolant system decontamination to the end of the safe storage period would greatly increase the difficulty of this task, since the pumps, valves, and associated equipment required to handle the decontamination solutions would likely be unusable after an extended storage period.

The need for accident cleanup of structures such as the auxiliary building (PWR), or the radwaste building (BWR), depends on the extent of contamination within these buildings and on whether or not significant quantities of contaminated water are present in tanks or pipes or in building sumps. If accident cleanup of the fuel and/or auxiliary building or the radwaste building is required, it would precede accident cleanup of the containment, since many of the systems and services required for cleanup of the containment are located in these other buildings.

4.1.3 Decommissioning the Reactor

The alternatives for completing the decommissioning of a nuclear power reactor that has been involved in a serious accident are DECON (immediate decontamination to unrestricted release), SAFSTOR (safe storage with deferred decontamination to unrestricted release), and ENTOMB (entombment of radioactive materials with decay to unrestricted release).[1,2] DECON permits termination of the facility operating license, while SAFSTOR and ENTOMB require continuance of an amended version of the license for extended periods of time. The amended nuclear license allows the licensee to possess but not to operate the facility. These alternatives, applied to post-accident decommissioning, are discussed in the following paragraphs.

4.1.3.1 DECON

DECON is a pseudoacronym defined by the NRC as the immediate removal of all radioactive material to permit license termination and release of the property for unrestricted use. To achieve an unrestricted use condition, the residual radioactivity levels in the facility must be reduced to values that do not exceed limiting values set by regulatory guidance.[a] Tasks that must be performed following accident cleanup, if DECON is chosen as the alternative for decommissioning, include:

- remove activated and contaminated materials from reactor building
- decontaminate reactor building to release levels
- dismantle and decontaminate other buildings with significant radioactive contamination (e.g., the fuel and auxiliary buildings for the PWR and the radwaste building for the BWR)
- dismantle and decontaminate turbine building and other structures with low levels of contamination
- package and ship radioactive materials
- survey site and decontaminate as necessary
- demolish buildings and complete site restoration (optional).

(a) Current regulatory guidance on acceptable surface contamination levels for unrestricted release of a nuclear reactor is contained in Regulatory Guide 1.86. The NRC is considering defining acceptable conditions for the unrestricted release of a decommissioned facility in terms of the potential radiation dose to an individual who uses the facility.[2]

DECON meets the requirements for termination of the facility license and renders the facility and site available for unrestricted use within a finite time period. Other advantages of DECON include the availability of personnel who are knowledgeable about the facility to form a decommissioning work force and the elimination of the need for long-term security, surveillance, and maintenance that would be required for the other decommissioning alternatives. Disadvantages of DECON are the larger initial commitments of personnel radiation exposure, waste-disposal-site space, and money than are required for the other alternatives.

For DECON, nonradioactive equipment and structures need not be torn down or removed as part of the decontamination procedure. In addition, once the radioactive structures are decontaminated to levels permitting unrestricted use, they may be put to some other use or demolished, at the owner's option. When a facility and site 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 NRC 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 the decommissioning and are done at the owner's option. Costs of demolition and site restoration are not included in this study.

4.1.3.2 SAFSTOR

SAFSTOR means to fix and maintain property so that risk to public safety is acceptable for a period of storage followed by decontamination and/or decay of residual radioactivity to an unrestricted level. SAFSTOR consists of: 1) a period of facility and site preparation (preparations for safe storage) that includes removal of the fuel from the reactor (for an accident-damaged reactor fuel removal would have been accomplished during accident cleanup that precedes the decommissioning) and concentration and immobilization of dispersible radioactive materials, 2) an interim period of continuing care (safe storage) that encompasses security, surveillance, and maintenance, and 3) the deferred removal of any remaining contamination to permit release of the facility for unrestricted use (deferred decontamination). An amended version of the nuclear license that does not permit operation of the facility would remain in force throughout the safe storage period, since materials having radioactivity levels above unrestricted release levels remain onsite. The duration of the safe storage period is undefined; however, periods of up to 100 years are consistent with recommended EPA policy on institutional control reliance for radioactivity containment.(2)

Tasks that must be performed following accident cleanup to prepare a reactor facility for safe storage include:

- decontaminate and fix residual contamination in the containment structure and in other supporting buildings
- deactivate equipment
- isolate contaminated areas
- package and ship contaminated materials
- install systems and equipment needed during safe storage.

Deferred decontamination includes whatever actions are required at the end of the safe storage period to terminate the nuclear license and release the property for unrestricted use. Some disassembly and disposal of radioactive components and equipment is still required, but the occupational radiation dose and disposal site requirements are potentially greatly diminished. The reduction in occupational radiation dose is determined by the half-lives of the radioisotopes that are controlling sources for occupational exposure. For fission product contamination that results from the reactor accident, the controlling isotopes for occupational exposure are ^{134}Cs with a half-life of 2.05 years, and ^{137}Cs and ^{90}Sr with half-lives of about 30 years. A waiting period of 100 years would result in a dose reduction of about a factor of 10 for an isotope with a 30-year half-life. For activated reactor components, the controlling isotope for occupational exposure is ^{60}Co with a half-life of 5.27 years. A waiting period of 100 years would result in a dose reduction of about a factor of 500,000 for this isotope.

The primary benefit of the SAFSTOR alternative would be the occupational dose reduction resulting from deferral of some decommissioning tasks until fission product and activation product radioactivity has a chance to decay.

An additional benefit would be the reduction in offsite waste disposal requirements for this alternative. Disadvantages of SAFSTOR include the need for continuing surveillance and physical security to ensure the protection of the public during the safe storage period. Maintenance of the facility structures and of equipment that provides essential services (fire protection, radiation monitoring, etc.) is also necessary. Deferral of the decommissioning to the end of the safe storage period has the disadvantages that personnel who are familiar with the facility may not be available to form the decommissioning staff and some onsite equipment needed for the decommissioning may not be in serviceable condition.

Use of the SAFSTOR alternative might be necessary if adequate disposal space for cleanup and decommissioning wastes is not available at shallow-land burial grounds. SAFSTOR could also be used if it is necessary to provide for interim onsite storage of spent fuel or of highly radioactive materials such as ion exchange resins and evaporator bottoms from accident cleanup operations. Disposal requirements for spent fuel or for highly radioactive wastes are not yet defined. A requirement for deep geologic disposal of these wastes may be forthcoming. Since a deep geologic facility does not now exist, temporary onsite storage of these wastes may be necessary until a permanent repository becomes available. It is unlikely that most reactor sites could qualify as permanent waste repositories because of such factors as nearby population densities and hydrology. Therefore, storage of wastes onsite would be an interim measure, followed ultimately by decontamination of the facility and site.

4.1.3.3 ENTOMB

ENTOMB means to encase and maintain property in a strong and structurally long-lived material (e.g., concrete) to assure retention and isolation from the environment until the contained radioactivity decays to an unrestricted level. ENTOMB is intended for use where the residual radioactivity will decay to levels permitting unrestricted release of the facility within reasonable time periods (i.e., within the time period of continued structural integrity of the entombing structure). Recommended EPA policy on institutional control reliance for radioactivity containment suggests that the entombing period not exceed approximately 100 years.[2] The use of ENTOMB following a serious reactor accident is unlikely because of the presence of significant quantities of long-lived (i.e., 30-year half-life) radionuclides that will not decay to release levels in 100 years.

ENTOMB is similar to SAFSTOR in that it consists of a period of facility and site preparations, followed by a period of safe storage that includes security, surveillance, and maintenance activities. Following entombment, these activities are minimal unless the irradiated fuel is stored onsite.

In the ENTOMB strategy considered in this study, the irradiated fuel and the reactor vessel internals that are contaminated with long-lived activation products are removed and shipped to a nuclear waste repository. As much as possible of the radioactive equipment from outside the entombment structure is consolidated and entombed within. Tasks that must be performed following accident cleanup, if ENTOMB is chosen as the decommissioning alternative, include:

- remove reactor vessel internals remaining after reactor defueling
- remove all contaminated equipment from above the operating floor level and store below the operating floor level - PWR (the corresponding activity for the BWR would involve the storage of contaminated material within the containment vessel)
- decontaminate onsite structures (other than containment structures)
- pour concrete slab within containment building at operating floor level - PWR (the corresponding activity for the BWR would involve sealing the containment vessel)
- install security and surveillance monitoring equipment
- package and ship contaminated materials
- remove residual noncontaminated structures and complete site restoration (optional).

Under existing regulations, the nuclear license must remain in force for an indefinite period, until either the entombed radioactivity has decayed to unrestricted release levels or the entombment structure is dismantled and the entombed radioactivity removed. If it becomes desirable to terminate the nuclear license prior to decay of the entombed radioactive material to unrestricted release levels, dismantlement of the entombment structure would be required. This represents a task that is much more difficult than dismantlement of the unentombed facility, since the entombment structure is built to endure for a long period of time. Therefore, while dismantlement of the entombment structure is not impossible, ENTOMB must be viewed as the almost irreversible creation of a radioactive waste repository on the site and commitment to long-term maintenance of the nuclear license.

4.2 KEY STUDY BASES

This study is intended to provide post-accident cleanup and decommissioning information useful to regulators, designers, and operators of light water reactors. A number of key study bases (i.e., study ground rules) are established to guide the emphasis of the study and to ensure that the primary objective of the study is achieved.

The requirements and costs of accident cleanup and decommissioning following a reactor accident depend on the nature of the accident and on the radioactive contamination and physical damage that result, as well as on such factors as facility location, specific facility design, and operating practices during the facility lifetime. In addition, requirements and costs depend upon the specific methods and techniques used in the accident cleanup and decommissioning, and upon the capability to dispose of wastes. There can also be an impact from specific rules and orders imposed on the facility after the accident. The bases and assumptions used in this study have a major impact on the estimates of time and manpower requirements, safety, and cost. These bases and assumptions must therefore be carefully examined before the results can be applied to a different facility.

The key bases are:

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

2. Accident scenarios are chosen that are illustrative of a range of accident cleanup and decommissioning requirements and for which the costs of cleanup and decommissioning are substantially greater than those for reactor decommissioning following normal shutdown. An attempt is made to choose credible accident scenarios related to NRC licensing criteria for design basis accidents. Scenarios are not restricted to accidents that have occurred. An accident in which the partially molten core melts through the reactor pressure vessel is outside the scope of design basis considerations and is not analyzed.

3. The reference PWR for this study is the same reactor (Trojan) that was used as the reference reactor for a previous study (NUREG/CR-0130)[3] of PWR decommissioning following normal shutdown. The reference BWR is the same reactor (WNP-2) that was used as the reference reactor for a previous study (NUREG/CR-0672)[4] of BWR decommissioning following normal shutdown. The use of these reactors as reference facilities should not be construed as implying anything about their reliability and/or safety relative to other LWRs in operation or under construction. Their use facilitates comparisons with the earlier, non-accident, decommissioning studies.

4. The reference reactors are assumed to have operated for 30 effective full-power years and to be operating at full power at the time of a postulated accident. The fission product radioactivity in the reactor core at the time of the accident is based on a burn-up condition corresponding to 550 full-power days averaged over the entire core (i.e., equilibrium core conditions). These time periods are used because they are considered to be conservative in estimating contamination levels.

5. This study is conducted within the framework of existing regulations and regulatory guidance. No assumptions are made regarding what future regulatory requirements or guidance might be.

6. This study focuses on the technical effort and associated costs of accident cleanup and decommissioning. The study does not attempt to assess the effects of political and social considerations on the cleanup and decommissioning effort. It is recognized, however, that these considerations can have significant impacts on cleanup and decommissioning requirements, delaying the time of completion and increasing the cost of these activities. An effort is therefore made to examine the sensitivity of study results to delays or to alternative engineering fixes which may occur (see Chapter 11, Section 11.6).

7. Current technology and techniques are used in descriptions of accident cleanup and decommissioning procedures. Where developmental techniques are specified in the study, they are in an advanced state of development and believed to be ready for application.

8. Decontamination and decommissioning and radiation protection philosophies and techniques applied in the study conform to the principle of keeping public and occupational radiation doses as low as reasonably achievable (ALARA).

9. Sufficient funding is available to carry out the accident cleanup and decommissioning without significant delays.

10. Release of facilities and/or equipment for unrestricted use is assumed to be predicated on the decontamination of surfaces and equipment to residual levels of radioactivity as specified by present regulatory guidance.[5,6]

11. It is recognized that work in high radiation areas proceeds much more slowly than work in areas with minimal contamination. In preparing work schedules, allowances are made for the unavoidable inefficiencies associated with this work. However, no allowances are made for unforeseen events that might impede the conduct of the work. A 25% contingency is added to cost totals to account for work delays and unanticipated material and equipment costs.

12. Some of the tasks related to cleanup and decommissioning of an accident-damaged reactor must be performed in high radiation areas where workers can receive their allowed quarterly or annual radiation dose in relatively short periods of time. In this study, manpower requirements for accident cleanup and decommissioning are adjusted upward beyond the numbers required for efficient performance of the work to provide a sufficient manpower pool so that no individual worker exceeds 5 rem/year.(7)

13. The aftermath of a reactor accident provides unique opportunities for research in accident consequences (e.g., damage to electrical systems and components, fuel core evaluation, requirements for management of radioactive wastes, etc.). As demonstrated at TMI-2,(8) accident research and development activities can be time-consuming and costly additions to accident cleanup activities. Because most of these research and development activities do not directly contribute to accident cleanup, no scheduling or funding allowances are made for them in this study. Exceptions are those activities related to the design, fabrication, and testing of special tools and equipment needed for decontamination and defueling operations for which costs are included in this study.

14. Costs are in early-1981 dollars. To make cost comparisons with previous studies(3,4) of LWR decommissioning following normal shutdown, the cost results from these earlier studies, which are in 1978 dollars, are converted to the 1981 cost base.

15. Based on 10 CFR Part 61 criteria, low-level radioactive wastes from accident cleanup and decommissioning operations are assumed to be transported by truck to an authorized shallow-land burial ground for disposal. The criteria of 10 CFR Part 61 may result in certain highly radioactive and/or transuranic wastes from accident cleanup and decommissioning being deemed unsuitable for near-surface disposal. Because criteria for disposal of these wastes are not defined and because a deep geologic facility to dispose of these

wastes and damaged fuel assemblies does not now exist, the estimated requirements and costs of interim storage of highly radioactive wastes and spent fuel assemblies at a federal repository are evaluated in this study. In addition, because of the potential that a reactor involved in an accident may be unable to dispose of its wastes, either because of lack of disposal capacity or regulatory or political constraints, this study also analyzes extended onsite storage of both low-level wastes and highly radioactive and/or transuranic wastes and spent fuel assemblies.

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

4.3 TECHNICAL APPROACH

The technical approach used to conduct this study is illustrated in Figure 4.3-1.

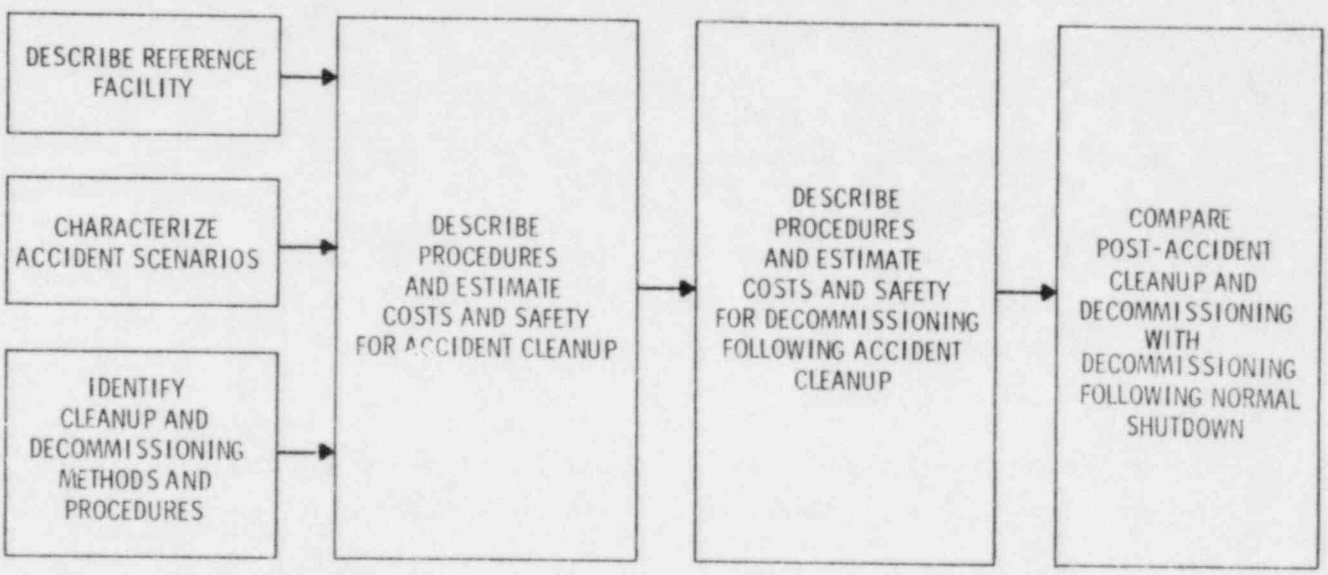

FIGURE 4.3-1. Technical Approach of This Study

The first step is to describe the reference facility. The reference PWR is the Trojan nuclear plant at Rainier, Oregon, operated by Portland General Electric Company. The reference BWR is WPPSS Nuclear Project No. 2 (WNP-2) being built near Richland, Washington, by the Washington Public Power Supply System (WPPSS). These reactors were used as reference facilities for previous studies[3,4] of reactor decommissioning following normal shutdown. Use of these same reactors in this study facilitates comparisons between the requirements and costs of post-accident cleanup and decommissioning and of normal-shutdown decommissioning and is not intended to imply anything about their safety and reliability relative to other LWRs in operation or under construction.

Accident scenarios are chosen that are believed to be credible based on reviews of reactor accident experience, of safety analysis reports, and of current NRC safety evaluation reports and licensing criteria. The accidents analyzed require a significant cleanup and decommissioning effort. Each accident scenario is characterized in terms of radioactive contamination of building surfaces and equipment, damage to the fuel core and to the containment structure and equipment, and radiation dose rates to decommissioning workers. The three scenarios analyzed are:

1. A small loss-of-coolant accident (LOCA) in which emergency core cooling functions to limit the release of radioactivity. Some fuel cladding rupture is postulated, but no fuel melting. The consequence scenario includes moderate contamination of the containment structure but no significant physical damage to buildings and equipment.

2. A small LOCA in which emergency core cooling is delayed, resulting in 50% fuel cladding failure and a small amount of fuel melting. The accident results in extensive radioactive contamination of the containment structure, moderate radioactive contamination in supporting buildings, and minor physical damage to structures and equipment.

3. A major LOCA in which emergency core cooling is delayed, resulting in 100% fuel cladding failure and significant fuel melting and core damage. The accident results in extensive radioactive contamination in the containment structure, moderate radioactive contamination in supporting buildings, and major physical damage to structures and equipment.

This study provides an analysis of the technical requirements for decommissioning an accident-damaged nuclear power plant, including planning and preparation, accident cleanup activities, and procedures for decommissioning the facility after accident cleanup is completed. The study does not include considerations of immediate stabilizing measures taken following an accident (such as bringing the accident under control, rescuing casualties, extinguishing fires, or removing decay heat). However, the costs and occupational exposures for activities needed to prepare for decontamination of the containment structure (such as venting inert gases, entries into the containment structure and supporting buildings to make measurements, or repairing essential systems and services) are included in this analysis. As discussed previously, the accident cleanup activities would be similar whether the reactor is refurbished for restart or decommissioned. Hence, the requirements, costs, and safety analyses given in this report are considered to be a good representation independent of the ultimate use of the plant. Alternatives for completing the decommissioning after accident cleanup include DECON, SAFSTOR, and ENTOMB. The study does not include consideration of the activities related to the refurbishment and restart of a reactor following the accident cleanup period.

Regulatory guidance is reviewed, summarized, and used as an aid and basis in the study. Guidance for determining allowable contamination levels for unrestricted release of a facility based on realistic dose assessments to an exposed individual is described.

Techniques for the decontamination of facilities are reviewed. Work and time schedules are developed to conceptually perform accident cleanup activities and to decommission the reference reactors by each decommissioning alternative. Postulated work schedules include allowances for time needed to prepare for entry into contaminated areas and for inefficiencies associated with work in high radiation fields.

Safety assessments are performed to estimate radiological hazards to workers and to the public from accident cleanup and decommissioning operations. These analyses include radiological exposures to workers and the public from normal cleanup and decommissioning operations and from potential accidents. Nonradiological industrial accidents to workers are also estimated. The safety analyses use established data and methodology to estimate the release mechanisms, dispersion, and pathways and exposure modes of the released materials.

Direct costs of accident cleanup and decommissioning are estimated including labor, materials, equipment, and the packaging, transportation, and disposal of radioactive wastes. Costs are estimated for planning and preparation, for accident cleanup, and for completion of the decommissioning. Costs are also estimated for the continuing care periods of the SAFSTOR and ENTOMB alternatives. Cost ranges are defined to estimate the sensitivity of the total cost to variations in key cost elements such as levels of radioactive contamination and waste treatment practices. Alternatives for financing the costs of post-accident cleanup and of decommissioning are examined.

The requirements and costs of post-accident cleanup and of decommissioning are compared with those for normal-shutdown decommissioning as defined in previous studies.[3,4] For PWR accident cleanup, the requirements and costs are also compared with data from TMI-2. (The accident at TMI-2 on 28 March 1979 was the only commercial power reactor accident which has caused significant contamination.) In addition, because accident cleanup is an activity which takes place at a time when the condition of the plant is uncertain and when social, political, financial, and regulatory constraints can affect the progress and costs of cleanup activities, the study also analyzes the sensitivity of accident cleanup costs to various factors which could impact these costs.

The results of the PWR analysis are given first in this report and are presented in more detail than those of the BWR analysis. The reference PWR is described in Chapter 7. PWR accident scenarios are described in Chapter 8. The technical requirements, costs, and safety impacts of PWR accident cleanup and decommissioning are summarized in Chapters 10 through 14. Impacts of alternative scenarios for waste disposal on the costs and safety of PWR accident cleanup and decommissioning are discussed in Chapter 15. The BWR analysis is summarized in Chapter 16. To trace the logic and justify the assumptions used in making the BWR analysis, the reader must refer to the appropriate sections in Chapter 8 and in Chapters 10 through 15 where reference accident scenarios and the technical requirements, costs, and safety impacts of PWR accident cleanup and decommissioning are discussed.

REFERENCES

1. G. D. Calkins, Plan for Reevaluation of NRC Policy on Decommissioning of Nuclear Facilities, NUREG-0436, Revision 1, Supplement 1, U.S. Nuclear Regulatory Commission, Washington, D.C., August 1980.

2. Draft Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, NUREG-0586, U.S. Nuclear Regulatory Commission, Washington, D.C., January 1981.

3. R. I. Smith, G. J. Konzek, and W. E. Kennedy, Jr., Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station, NUREG/CR-0130, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, June 1978.

4. H. D. Oak, et al., Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station, NUREG/CR-0672, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, June 1980.

5. Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, U.S. Atomic Energy Commission, Washington, D.C., June 1974.

6. Plan for Reevaluation of NRC Policy on Decommissioning of Nuclear Facilities, NUREG-0436, Revision 1, December 1978, (See Attachment A).

7. U.S. Code of Federal Regulations, Title 10, Part 20.101, "Radiation Dose Standards for Individuals in Restricted Areas," (10 CFR 20.101), U.S. Government Printing Office, Washington, D.C., 1980.

8. TMI-2 Seminar, December 4-5, 1981, San Francisco, California.

CHAPTER 5

REGULATORY CONSIDERATIONS FOR POST-ACCIDENT CLEANUP AND FOR DECOMMISSIONING

The responsibility for post-accident cleanup and for decommissioning of a commercial nuclear facility belongs to the licensee. To properly discharge this responsibility, the licensee must be aware of applicable regulatory requirements and must accomplish the cleanup and decommissioning in compliance with these requirements. In this chapter, existing regulations, guides, and standards that apply to these activities at an accident-damaged reactor are identified and discussed. A comprehensive review of existing statutes, regulations, and guidelines for decommissioning is contained in Reference 1. Detailed discussions of regulations and guides that apply to decommissioning of LWRs are given in References 2 and 3. Regulations and guides that relate to post-accident cleanup are discussed in Reference 4.

Statutory authority for the regulation of activities related to the commercial nuclear fuel cycle is contained in the Atomic Energy Act of 1954 (42 U.S.C. 2011 et seq.) and the Energy Reorganization Act of 1974 (42 U.S.C. 5841 et seq.) and in subsequent amendments. Regulatory authority is delegated to the Nuclear Regulatory Commission (NRC) and the NRC has promulgated regulations in Title 10 of the Code of Federal Regulations (10 CFR) to carry out the provisions of these acts. The NRC has published Regulatory Guides to assist applicants and licensees in carrying out their regulatory obligations. The decontamination and/or decommissioning of an accident-damaged reactor by the licensee is also subject to statements, orders, and amendments to the facility license issued by the NRC pursuant to its statutory responsibility for the regulation of nuclear fuel cycle activities.

5.1 EXISTING REGULATIONS AND GUIDANCE FOR POST-ACCIDENT CLEANUP AND FOR DECOMMISSIONING

Existing regulations and guidance that pertain to the post-accident cleanup and decommissioning of nuclear power reactors are contained in the following references:

- 10 CFR 50.54(a)[(a)] - requires that a licensee take reasonable steps to obtain onsite property damage insurance to cover reasonable decontamination and cleanup costs resulting from an accident at the licensed facility.
- 10 CFR 50.82 - requires that a licensee desiring to terminate a license provide procedures for the disposal of radioactive material, the decontamination of the facility and site, and the assurance of public safety.
- 10 CFR 51.5(b)(7) - specifies that an environmental impact statement may be required prior to decommissioning a nuclear power reactor.
- Regulatory Guide 1.86 - provides guidance in satisfying the requirements of 10 CFR 50.82. This guide also states the requirements for a possession-only license and the criteria by which a decontaminated reactor is judged to be suitable for release for unrestricted access or use.

In addition to the regulations and guidance listed above, a licensee engaged in the decommissioning of a nuclear power plant must abide by applicable aspects of federal regulations and guidance that pertain to many different topics, including:

- licensing procedures
- public and occupational radiation standards
- radiation monitoring
- training of workers
- industrial safety
- financial requirements
- packaging and transportation of radioactive wastes
- special nuclear materials handling
- system design and quality control
- physical protection of plants and material.

References to federal regulations and regulatory guides and to industry standards pertaining to these topics are found in Chapter 5 of Reference 2 and in Chapter 5 of Reference 3.

(a) This notation designates Section 50.54(a) of Title 10, Code of Federal Regulations, Part 50 (typical).

5.2 LICENSE AMENDMENT PROCEDURES

During the planning and preparation phase that precedes the actual decommissioning of an accident-damaged nuclear reactor, the facility operating license is amended to permit cleanup and decommissioning to proceed.

The facility operating license is regulated by 10 CFR Part 50, Licensing of Production and Utilization Facilities. Requirements for the termination of the operating license are presented in 10 CFR 50.82, "Application for Termination of Licenses." Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, provides guidance in satisfying the requirements of 10 CFR 50.82. Regulatory Guide 1.86 specifies the procedures and the documentation requirements for amending the license to possess but not to operate the facility and for obtaining a dismantling order. In addition, it delineates the applicability of the amended license and the dismantling order to the various decommissioning alternatives, the surveillance and security requirements if the final decommissioning status (i.e., long-term care) requires a possession-only license, and the procedures for terminating the license.

In requesting to amend a facility license to possess but not to operate the facility, the licensee must provide the following information:

- a description of the current status of the facility
- an inventory of the radioactive materials and their location in the facility
- a description of the decommissioning activities to be performed
- a description of measures to be taken to prevent criticality or reactivity changes and to minimize releases of radioactivity from the facility
- any proposed changes to the technical specifications that reflect the possession-only facility status and the decommissioning activities to be performed
- a safety analysis of both the activities to be accomplished and the proposed changes to the technical specifications.

If major plant changes are required (as is the case for the DECON and ENTOMB decommissioning alternatives), an NRC dismantling order is required. The request for a dismantling order must be accompanied by a dismantlement plan that includes, but is not limited to, the following information:

- a description of the ultimate status of the plant
- a description of the dismantling activities, including radioactive waste disposal and site decontamination, and the associated environmental and safety precautions
- a safety analysis of the dismantling activities, including any effluent that may be released
- a safety analysis of the plant in its ultimate status.

As part of the license amendment request or the dismantlement plan, quality assurance of the decommissioning activities should be addressed as outlined in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The requirements in Appendix B pertain to such topics as design, purchasing, and fabrication, but do not specifically address decommissioning. However, the principles and objectives of such guidance should be applied to all decommissioning activities. Additional guidance is found in the NRC's Standard Review Plan, Section 17.1, "Quality Assurance During the Operating Phase,"(5) and in Regulatory Guide 1.143, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants.

In accordance with the provisions of 10 CFR 51.5(b)(7), the cleanup and the decommissioning of a nuclear power reactor following an accident may require the preparation by the NRC of an environmental impact statement. The licensee may be asked to prepare documentation in support of such a statement. (An example is the programmatic environmental impact statement prepared for the post-accident cleanup of TMI-2.)(4) This documentation describes the proposed decontamination and/or decommissioning operations and assesses their impact on man and the environment. The impacts of various decommissioning alternatives are considered. The need to prepare an environmental impact statement for the overall decommissioning operation or for some phase of the operation is determined by the NRC on a case-by-case basis.

The decontamination and/or decommissioning of a reactor that has been involved in an accident is also subject to constraints imposed by statements, orders, and amendments to the facility license issued by the NRC subsequent to the accident. Such NRC actions in connection with the decontamination of TMI-2 for a 2-year period following the accident of March 28, 1979, are detailed in Reference 4. They include requirements related to the controlled venting of the reactor building atmosphere, the use of special equipment (EPICOR-II) for processing accident water, prohibition of the discharge of accident water to the river, the onsite storage of radioactive wastes, and the removal of decay heat from the damaged reactor core. Statements, orders, and amendments to the facility license are of necessity specific to the particular reactor and accident and are issued by the NRC on a case-by-case basis.

5.3 REGULATIONS ON MANAGEMENT OF RADIOACTIVE WASTES FROM POST-ACCIDENT CLEANUP AND FROM DECOMMISSIONING

One of the most important areas of concern in the post-accident cleanup and subsequent decommissioning of a nuclear reactor is the management of the large volumes of radioactive wastes (both liquids and solids) that result from the accident and from cleanup and decommissioning operations. This section provides information about the current status of regulations and guidelines pertaining to certain areas of waste management. Regulatory considerations are discussed for the following topics:

- management of contaminated water
- special requirements for radioactive waste transportation
- disposal of radioactive wastes.

5.3.1 Management of Contaminated Water

A major consideration in the initial cleanup of a nuclear power reactor following an accident is the management of large quantities of contaminated water (both accident-generated water and water-based decontamination solutions) from the accident. Measures for the processing and storage or disposal of this water must be consistent with NRC regulations relating to the release of liquid effluents from licensed nuclear facilities and with EPA drinking water standards.

Processed accident water will contain as radioactive contaminants those fission product radionuclides not removed by filtration, evaporation, or ion exchange processes, as well as all of the tritium originally present in the water. In addition to solidification and packaging of this water for shallow-land burial, four other alternatives are possible for its disposal:

- controlled discharge of processed water to the river
- discharge to the atmosphere through natural or forced evaporation
- transportation as bulk liquid to an offsite location for disposal
- long-term, onsite storage.

Discharge to the River

Criteria governing the potential discharge of processed water to the river include:

1) the requirements of 10 CFR Part 20 which provide release limits for specific radionuclides - The discharge of processed water into the river would require dilution and release under controlled conditions that would implement 10 CFR Part 20 criteria. Table II, Column 2 of Appendix B of Part 20 gives limiting values for concentrations of radioactivity at the nearest downstream drinking water intake resulting from releases of radioactivity at the plant.

2) 10 CFR Part 50 Appendix I criteria for offsite radiological exposure

3) the Safe Drinking Water Act criteria related to the EPA's Primary Drinking Water Standards contained in 40 CFR Part 141[6] - These criteria establish maximum contaminant levels for ^{226}Ra, ^{228}Ra, gross alpha particle radioactivity, and beta particle and photon radioactivity from man-made radionuclides in community water systems. These levels must not be exceeded at a drinking water intake downstream from the discharge point.

4) state and local ordinances governing point source discharges to the river

5) specific post-accident orders such as that issued by the NRC at TMI-2 which prohibited the discharge of accident water to the river.[4]

Discharge to the Atmosphere

A second alternative for the disposal of processed accident water is to discharge it to the atmosphere, either through natural evaporation from ponds or through forced evaporation from a cooling tower. Releases would need to be controlled to maintain airborne concentrations below 10 CFR Part 20 Appendix B exposure limits and in accordance with 10 CFR Part 50 Appendix I dose design objectives. In addition, the concentrated feed blowdown from forced evaporation, which might contain 5 to 10% of the original tritium and any fission product radionuclides not removed during processing of the water, would require dilution to the extent required to satisfy 10 CFR Part 20 Appendix B and Safe Drinking Water Act criteria prior to discharge to the river.

Transportation to an Offsite Location for Disposal

A third alternative for the management of processed accident water is to package it as bulk liquid in tank trucks for transportation away from the site. The water could then be disposed of in a deep injection well, packaged for ocean disposal, or released at another site.

Current federal regulations do not prohibit bulk liquid shipments of tritiated water. Under the LSA[a] definitions in 10 CFR Part 71, tritium oxide in aqueous solution with concentrations up to 5 mCi/mL qualifies as LSA material. Under 49 CFR 173.392, bulk liquids may be transported in tank trucks provided the concentrations do not exceed 10% of the LSA concentrations. Thus, water with tritium concentrations up to 500 μCi/mL could be transported in bulk.

The disposal of chemically hazardous or radioactive waste in deep wells is regulated by the EPA through its Underground Injection Control (UIC) program under the Safe Drinking Water Act. Deep well injection may be authorized by the state authority responsible for water quality. Technical criteria and standards for deep well injection are contained in 40 CFR Part 146, and procedures for issuing state permits are described in 40 CFR Part 124. Within

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(a) LSA: low specific activity.

these regulations, Class I wells may be used to inject industrial, municipal, and nuclear wastes beneath the deepest stratum containing an underground drinking water source. Historically, these wells have not been used for the disposal of nuclear waste and it is uncertain whether a cognizant state authority would grant a permit to dispose of processed accident water through a Class I well within its borders.

Regulations governing ocean dumping are found in 40 CFR Part 220. Under these regulations, an application for an ocean dumping permit must include a statement of the need for the proposed dumping, an evaluation of alternative means of disposal, treatment, or recycle of the material, and an assessment of the anticipated environmental impacts of the proposed ocean dumping operation. In addition, under EPA's proposed regulations, the tritiated water would have to be packaged in containers that would retain their integrity until radioactive decay reduced the activity to "environmentally innocuous" levels. The level of tritium activity deemed environmentally innocuous in an ocean environment and the packaging requirements for ocean dumping are not specified in the regulations.

Long-Term, Onsite Storage

A fourth alternative is onsite storage of processed accident water. The water could be stored in liquid form in large storage tanks or it could be solidified with cement and stored in solid form.

If the water were stored as bulk liquid, long-term storage would be required to reduce tritium concentrations to levels consistent with primary drinking water standards. Assuming an initial tritium concentration of 1 μCi/mL, attainment of EPA primary drinking water standard concentrations (2×10^{-5} μCi/mL) would require a decay period of about 200 years. A criterion for storage of processed accident water in tanks is that the content of radioactivity in each tank be limited such that a tank failure would not result in greater than 10 CFR Part 20 (Table 2, column 2) concentrations at the nearest drinking water intake.

If the water were mixed with cement, it could be stored onsite in the form of large concrete blocks. These concrete blocks would immobilize the water, and when coated with asphalt or other weather-resistant material, they could be stored outside for relatively long periods. Following such storage, they could be shipped offsite to a radioactive waste storage facility.

An estimate of the requirements for storage of processed accident water from TMI-2 in the form of concrete blocks indicates that 960 blocks having dimensions of 1.8 m x 1.8 m x 3.05 m would be required to immobilize 5800 m^3 of water.(4) Each block would weigh approximately 18,000 kg. The blocks would occupy about 16,000 m^2 (4 acres) of land if stored one layer only.

Storage of processed accident water for relatively long periods (10 to 20 years) is a practical alternative that would permit deferral of a decision on ultimate disposition of the water. However, onsite storage for time periods (100 to 200 years) that would result in reductions of tritium concentrations to either innocuous levels or to primary drinking water standards would have the effect of converting the reactor site to a low-level waste disposal site, which would require an evaluation of the acceptability of the site for low-level waste disposal.

5.3.2 Special Requirements for Transport of Radioactive Waste

NRC regulations pertaining to the packaging and transport of radioactive materials are contained in 10 CFR Part 71. Applicable Department of Transportation (DOT) regulations are contained in 49 CFR Parts 170-189. These regulations are reviewed in References 1-3 and are not discussed in detail in this report.

DOT and NRC regulation of the transportation of radioactive materials has, until recently, focused on packaging controls (package design, quantity of radioactivity per package, package surface dose rates, etc.). New DOT regulations that become effective February 1, 1982, focus on routing and operational controls for highway transportation of radioactive materials including radioactive waste. These new regulations are contained in revisions to 49 CFR Parts 173 and 177 and include the following:

1) A requirement that a motor vehicle carrying placarded radioactive material be operated on a route that presents a risk to the fewest persons, unless there is not any practicable alternative highway route or it is operated on a "preferred" highway.

2) A requirement that any motor vehicle transporting a package containing a "large quantity" of radioactive materials be operated on "preferred" highways in accordance with a written route plan prepared by the carrier before departure. Preferred highways would be designated by state agencies based on a policy of overall minimization of the impacts from normal transportation and from transportation accidents. This rule would require use of an interstate urban circumferential, or bypass route, if available, to avoid the transport of radioactive material through cities.

3) Notification of states in advance of Type B shipments. NRC approval of routing for spent fuel shipments is also now required (10 CFR 73.37).

5.3.3 Disposal of Radioactive Wastes

Radioactive wastes from the cleanup and decommissioning of a nuclear power reactor involved in an accident range from low-specific-activity trash and rubbish to high-specific-activity wastes, including loaded ion exchange materials, accident sludges, and spent filter cartridges. Cleanup and decommissioning wastes also include the activated reactor vessel and vessel internals, activated and contaminated concrete, contaminated equipment, and undamaged and damaged spent fuel.

At the present time, only shallow-land burial grounds are available for the disposal of commercial radioactive wastes. The NRC proposes to add to its rules in 10 CFR a new Part 61 to provide licensing procedures, performance criteria, and technical criteria for licensing these burial facilities, including criteria for the classification of waste into different categories.(7) These different categories permit the use of different waste form and burial requirements to be applied for near-surface burial, depending on the activity of the waste. The technical requirements on waste form imposed by Part 61 may result in some wastes being deemed not suitable for near-surface burial.

No regulatory framework has yet been developed to specifically address the disposal of low-level wastes that do not meet the criteria set forth in 10 CFR Part 61 for near-surface disposal. Accordingly, certain of the post-accident cleanup and decommissioning wastes will have to be carefully evaluated on a case-by-case basis with regard to characteristics such as specific activity, radionuclide content, total radioactivity inventory, and waste form. Ultimate disposition of these wastes will depend on the unique characteristics they possess and on the availability of suitable facilities for their handling and disposal.

In the packaging and disposal of wastes resulting from chemical decontaminations, consideration must be given to criteria contained in 10 CFR Part 61 on wastes containing chelating agents. Table 1 of 10 CFR 61.55 indicates that wastes containing chelating agents in concentrations greater than 0.1% are not permitted for near-surface disposal except as specifically approved by the Nuclear Regulatory Commission.

Interim storage may be required for post-accident cleanup and decommissioning wastes that cannot be disposed of by shallow-land burial and for which alternative disposal facilities have not been established. At the TMI-2 site, high-specific-activity wastes such as spent ion exchange resins from the processing of accident liquids are currently being stored on an interim basis in specially constructed reinforced-concrete silos. An Order for Modification of License, issued by the NRC in October 1979, prohibits the offsite shipment of spent resins from TMI-2 except with prior NRC approval.[4]

The only presently available treatment facilities for volume reduction, immobilization, and packaging for eventual repository disposal of high-specific-activity wastes are at federally owned (DOE) sites such as the Idaho National Engineering Laboratory, the Savannah River Plant, and the Hanford Site. A Memorandum of Understanding (MOU) outlining responsibilities for the removal and disposition of certain solid nuclear wastes from cleanup of TMI-2 was agreed to by the NRC and the DOE on March 15, 1982.[8] The memorandum relates to currently identified TMI-2 solid wastes that may not be suitable for disposal at a shallow-land burial site.

Under the terms of the MOU, following the removal of the damaged TMI-2 fuel core from the reactor vessel, the entire core will be shipped to a DOE facility to survey and select those portions most appropriate for DOE's R and D program. The remainder of the core will remain in storage at the DOE facility and will ultimately be disposed of under an agreement to be negotiated between DOE and the owner. The DOE has also agreed to take possession of and retain the highly radioactive purification system resins (the EPICOR-II and Submerged Demineralizer System resins) on a reimbursable basis.

5.4 REGULATIONS AND GUIDANCE FOR LICENSE TERMINATION AND FACILITY RELEASE

A primary goal of decommissioning is to terminate the nuclear license and release the facility for unrestricted use. Termination of the nuclear license is regulated by 10 CFR 50.82. Methods and procedures for implementing the requirements of 10 CFR 50.82 are described in Regulatory Guide 1.86.

Release of a facility for unrestricted public access requires that residual radioactive contamination be reduced to levels considered acceptable for public protection. Table 1 of Regulatory Guide 1.86 provides guidance on currently acceptable surface contamination levels for unrestricted areas. Guidance in performing the final radiation survey to verify that a site has been decontaminated to acceptable release levels is contained in a recent Oak Ridge National Laboratory (ORNL) report[9] on monitoring for compliance with termination survey criteria. Another recent ORNL report provides guidance on methods and costs of conducting terminal radiological surveys.[10] Regulatory Guide 1.86 by itself is not explicit enough in relating contamination levels to annual dose rates or justifying that the values presented are ALARA, which has resulted in the need for case-by-case licensing decisions to set acceptable residual levels.

The draft generic environmental impact statement (GEIS) on decommissioning of nuclear facilities[11] contains a recommendation that the allowable residual radioactivity level for facility release be based on the dose anticipated to be received by individuals who use the facility. As set forth in the Energy Reorganization Act of 1974, the Environmental Protection Agency (EPA) has responsibility for establishing radiation dose standards for the protection of the public health and safety. Thus, the EPA has responsibility for establishing criteria for residual radioactivity limits considered safe for decommissioning a nuclear facility to unrestricted access. EPA has not yet instituted these criteria and is not scheduled to begin its review until 1984 with standards being set at some later time. In the absence of standards in this area, the NRC, based on the statutory authority contained in the Atomic Energy Act to protect the health and safety of the public, has issued the GEIS and announced its intention to promulgate regulations establishing criteria for acceptable residual radioactivity levels.[12]

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CHAPTER 6

FINANCING FOR POST-ACCIDENT CLEANUP AND FOR DECOMMISSIONING

A nuclear plant licensee has responsibility for decommissioning the facility prior to license termination and release of the facility for unrestricted use. This responsibility includes an obligation to provide adequate funds to accomplish the decommissioning in a safe and efficient manner. Post-accident cleanup and decommissioning of a nuclear power reactor presents special financial problems because the accident is not anticipated and the combined costs of cleanup and decommissioning can be several times the cost of decommissioning following normal shutdown.

Mechanisms for assuring the availability of funds for post-accident cleanup and for decommissioning are discussed in this chapter. Current regulatory developments in the area of decommissioning financing are summarized. Methods for funding decommissioning following normal shutdown are reviewed with an explanation of why these methods are not adequate for funding post-accident cleanup. Because the accident at TMI-2 illustrates the financial difficulties that can result for a licensee attempting to fund the cleanup of an accident-damaged nuclear power reactor,[1,2] the problem of funding the cleanup of TMI-2 is briefly discussed. Insurance alternatives for providing funds for post-accident cleanup and for decommissioning are described.

6.1 REGULATORY REQUIREMENTS FOR FINANCING DECOMMISSIONING

Currently there are not specific regulatory requirements relating to the assurance of funds for decommissioning of a nuclear power reactor. The U.S. Nuclear Regulatory Commission (NRC) is, however, considering revisions to its decommissioning funding requirements within the broader context of an overall reevaluation of its policies on decommissioning nuclear facilities.[3]

Financing strategies to provide assurance of funds for nuclear power plant decommissioning are described in a contractor report prepared for the NRC.[4] The NRC has issued a staff report on financial assurance of decommissioning[5] that contains separate chapters on funding the decommissioning of reactors and other nuclear facilities. Neither of these reports specifically addresses the problem of funding post-accident cleanup and decommissioning activities. The design and feasibility of an insurance pool set up by the utilities to pay the costs of post-accident cleanup and decommissioning is addressed in a recent NRC contractor report.[6]

To assure that licensees have the ability to finance the cleanup costs resulting from a nuclear-related accident so as to protect the public health and safety, the NRC recently promulgated an amendment to is regulations in 10 CFR, Part 50.[7] The regulation requires that each electric utility licensee obtain onsite property damage insurance to cover decontamination and cleanup costs resulting from an accident at the licensed facility. The regulation further states that the insurance must have a minimum coverage limit no less than the combined total of 1) base coverage offered by either American Nuclear Insurers (ANI) and Mutual Atomic Energy Reinsurance Pool (MAERP) jointly or by Nuclear Mutual Limited (NML); plus 2) excess coverage offered by Nuclear Electric Insurance Limited (NEIL), the Edison Electric Institute (EEI), ANI and MAERP jointly, or NML. Currently, as this study is completed, the base coverage is $450 million and the excess coverage is $290 million.

6.2 ALTERNATIVES FOR FUNDING DECOMMISSIONING FOLLOWING NORMAL SHUTDOWN

Several methods are available for providing funds for decommissioning following normal shutdown, including:[8]

- prepayment of estimated decommissioning funds into an account prior to facility startup
- decommissioning insurance, surety bonds, letters of credit, and/or lines of credit
- a sinking fund or funded reserve in which a prescribed amount of funds is set aside annually in a decommissioning account
- an internal reserve or unsegregated sinking fund.

These alternatives are discussed in References 5 and 8 in two previous studies of nuclear power plant decommissioning following normal shutdown.[9,10] The alternatives presuppose that both the operating lifetime of the plant and the amount of funds required for decommissioning are known with some degree of certainty. In the event of premature shutdown not involving an accident, only the first two funding mechanisms could provide the necessary funds for decommissioning.

If the sinking fund approach is chosen, several options are available to reduce the risk of unavailability of funds in the event of nonaccident-initiated premature shutdown. These options include one or more of the following:

- a large initial payment to a sinking fund prior to startup
- higher per-unit payments (in constant-value dollars) to a sinking fund during the early years of operation
- a surety bond posted by the utility
- a decommissioning fund insurance pool.

These risk-reducing options are discussed in Reference 10. In all cases, the options assume that the costs of decommissioning are relatively fixed and reasonably well known.

The decommissioning funding alternatives listed above are generally not adequate for funding the combined costs of cleanup and decommissioning of a nuclear power reactor following an accident, since an accident can result in very large cleanup expenses that would be difficult or impossible to cover by these funding methods. In addition, since reactor accidents occur very rarely, these expenses are more appropriately covered by insurance.

Since the accident at TMI-2, the utility operating that plant has experienced great financial difficulty in trying to fund the cleanup of the plant. Alternatives that have been suggested for providing funds for the cleanup of TMI-2 are described in Section 6.3.

The use of property damage insurance (up to approximately $1 billion) to provide for post-accident cleanup costs is currently being established within the electric utility industry. Approaches to providing post-accident cleanup insurance are discussed in Section 6.4.

6.3 FUNDING THE CLEANUP OF THREE MILE ISLAND

A loss of-coolant accident at Unit 2 of the Three Mile Island Nuclear Station (TMI-2) on 28 March 1979 resulted in a release of about 2600 m^3 of radioactively contaminated water to the reactor building basement, contamination of building surfaces and equipment in the reactor building and the auxiliary and fuel handling building, and damage to the fuel core. (The accident is described in Chapter 3, Section 3.7.) The cost of cleanup of TMI-2 has been estimated at between $600 million and $1.3 billion (net of $300 million in insurance proceeds), depending on which components of cost are included.[1,11] The accident has had serious financial consequences for General Public Utilities Corporation (GPU), the public utility holding company that owns the reactor, and for its three subsidiary operating companies, Metropolitan Edison Company (MetEd), Pennsylvania Electric Company (Penelec), and Jersey Central Power and Light Company (Jersey Central). GPU's financial problems have included an increase in cash requirements, a decrease in net income, and the inability to obtain money for refinancing long-term debt or to make capital improvements.[11] As this report is being written in mid-1982, no long-term financing is available to the company and even the continuation of short-term funding is uncertain.

The accident at TMI-2 has provided a first major test of the adequacy of existing liability and property insurance coverage. While the liability insurance coverage fostered under the Price-Anderson Act[a] has been adequate in paying all claims to date, property damage insurance coverage has proven to be inadequate due to the unanticipated decontamination expense.[b]

(a) The Price-Anderson Act was passed by Congress in 1957 and is in Section 170 of the Atomic Energy Act of 1954. It provides for insurance coverage of up to $560 million for offsite personal and property damage claims resulting from a nuclear accident.

(b) GPU had the TMI units insured for $300 million in onsite property damage insurance the maximum amount available at the time of the accident.

As of 31 December 1980, GPU had spent about $180 million in its accident recovery effort. To complete the cleanup as scheduled, about $100 to $150 million a year has been estimated to be needed over the next 6 years.[11] (This is net of insurance proceeds and also does not include the approximately $150 million for operating and maintaining the plant during the cleanup period.) Cleanup costs have been estimated to represent a significant fraction of GPU's projected major capital funding needs through 1986.[1] As of mid-1982, there is concern whether GPU would be able to borrow the needed funds for other capital requirements, such as bond retirements, as long as the company and its stockholders continue to be solely responsible for TMI-2 cleanup costs. As a result of the potential inability of GPU to both finance the cleanup costs and maintain system reliability, much of the necessary funding for TMI-2 cleanup may have to come from other sources than GPU system earnings.

Several options have been proposed for providing TMI cleanup cost support.[11,12] These include: 1) new ownership for the TMI units, 2) a nuclear fuel enrichment surcharge, 3) a mandated insurance assessment which would increase the amount of available property insurance and make funds from this increased insurance available retroactively to TMI-2, 4) GPU funding from increased rate revenue and stockholder earnings, 5) federal assistance, and 6) industry contributions. These options are not all-inclusive and each one has its limitations. However, they represent a cross-section of the kinds of solutions that have been proposed for funding the cleanup costs. A combination of these options, rather than any single one, may ultimately be employed.

In July 1981, the Governor of Pennsylvania proposed a comprehensive cost-sharing plan for TMI-2 cleanup that called for GPU and its subsidiary companies, other utilities operating nuclear power plants, manufacturers and suppliers of nuclear equipment, the federal government, and the states of Pennsylvania and New Jersey to share the cost of cleanup at the damaged reactor.[13,14] Several potential participants to the Governor's plan, including representatives of the federal government and of industry, have indicated a willingness to contribute to the TMI-2 cleanup effort.[15-17] However, as this is being written in mid-1982, no firm plan for funding the cleanup of TMI-2 has been adopted.

6.4 INSURANCE ALTERNATIVES FOR FUNDING POST-ACCIDENT CLEANUP AND DECOMMISSIONING

Because reactor accidents are unplanned and rare events, insurance appears to be an appropriate mechanism for providing funds for cleanup following an accident (and possibly for post-cleanup decommissioning). Property damage insurance for nuclear power plants is presently available through private insurance companies and utility-organized mutual insurance companies. The status of this insurance coverage as of mid-1982 is described in an NRC-contracted report.[18] As indicated in Section 6.1, the NRC has issued a rule that makes it mandatory for a nuclear power plant licensee to obtain the minimum property damage insurance available or to demonstrate to the statisfaction of the Commission that it possesses an equivalent amount of protection covering the facility.

Private companies providing property insurance are American Nuclear Insurers (ANI) and Mutual Atomic Energy Reinsurance Pool (MAERP). ANI has commitments for property insurance capacity from over 100 conventional stockholder-owned insurance companies. MAERP obtains its commitments for insurance capacity from about 100 mutual insurance companies. Both ANI and MAERP solicit reinsurance capacity from insurers abroad, with about one-half of their insurance capacity derived from foreign insurers. ANI and MAERP have also combined their resources by extensively reinsuring each other to form, in effect, a single, larger ANI-MAERP pool. In 1981, ANI and MAERP insured reactors at thirty-four states.

Nuclear Mutual Limited (NML) is a utility-organized mutual insurance company which, in 1982, insured reactors at twenty-seven sites. In addition to assessing premiums from its member utilities, NML obtains some reinsurance coverage from conventional insurers. Since a major nuclear accident would exhaust NML resources, the company has authority to assess each member utility its proportionate share of the insured balance due the member owner of the disabled nuclear unit. To limit the financial exposure of each member utility, the maximum yearly retrospective premium adjustment is limited to 14 times the annual premium rate paid by the member utility.

Nuclear Electric Insurance Limited (NEIL) is an industry self-insurance corporation organized in 1980 for the purpose of providing protection for power replacement costs when a reactor has suffered an outage caused by an accident. Weekly payments are provided, beginning 6 months after the outage. At the end of 1981, NEIL provided replacement power coverage for 51 reactor units at 34 sites. NEIL has recently initiated a second type of insurance coverage (NEIL-II) that provides property damage excess coverage. The NEIL-II coverage provides a second layer of insurance up to a specified maximum that tracks the primary coverage that a utility has with another insurer. Excess insurance coverage is described in more detail in Section 6.4.3.

At the time of the TMI accident March 1979, the maximum property damage insurance coverage provided by both ANI-MAERP and NML was limited to $300 million per insured unit. Estimates of cleanup costs for TMI-2 have shown that this amount of coverage is inadequate, and subsequently there has been a continuous increase in the amount of property damage insurance available to the nuclear utilities. In April 1981, ANI-MAERP coverage was increased from $300 million to $369 million per site per policy, and in January 1982, coverage was increased to $450 million per site. NML coverage was increased from $300 million to $450 million in August 1981. Both NML and ANI-MAERP expect to raise their maximum limits per insured site to $500 million during 1982.(18) NEIL-II, as of March 1982, offered up to $290 million per insured site in excess of $500 million.

As of mid-1982, several options are potentially available for further increasing the level of property insurance coverage at nuclear generating plants. These options include:

- voluntary increases in insurance coverage
- quota sharing
- multi-layer insurance coverage.

These options are not mutually exclusive. Each has advantages and disadvantages and all are currently being pursued by the industry. The option which appears to be gaining both utility and insurance industry support is multi-layer coverage, described in Section 6.4.3.

6.4.1 Voluntary Increases in Insurance Coverage

The first option for increasing nuclear property insurance would be to rely on the insurance and utility industries to voluntarily increase the coverage to adequate levels. As noted above, since the accident at TMI-2, both ANI-MAERP and NML have increased their levels of coverage. However, financial and regulatory constraints described in the following paragraphs could deter these companies from increasing their coverage beyond what is currently available or planned in the near future.

ANI-MAERP derives its insurance capacity from other insurance companies who commit funds to the ANI-MAERP insurance pool. Two factors limit the insurance provided by these other companies. The first factor is the lack of actuarial knowledge of the risks involved in providing coverage to nuclear units. Insurance companies have good actuarial knowledge of the risks involved in providing conventional coverage for life, auto, commercial, and homeowner insurance. In contrast, the limited experience with nuclear plants provides little actuarial basis for assessing risk, especially since the costs of cleanup and decommissioning following a reactor accident are largely unknown. Many companies prefer to avoid this actuarial uncertainty and to limit their commitment to nuclear assurance pools.

The second factor is an insurance company perception that uncertainties in federal regulatory practices related to nuclear power reactors could threaten future premium flows. Despite the low probability of a major nuclear accident in any given year, insurance losses from the accident could easily exceed total premiums collected by the insurers in that year. While insurers may experience a loss in a particular year, over a period of several years they expect to realize a return on their investment. The accident at TMI-2 exhausted several years of previously accumulated premiums for ANI-MAERP insurers. Many insurers would experience a net loss if another major nuclear accident were to occur in the next few years and the federal government took strong regulatory action to curtail the nuclear industry. Such federal regulatory action could threaten future premium flows, and adversely affect the overall return on investment to insurers.

NML cannot rapidly increase its level of insurance coverage without a corresponding increase in retrospective premium adjustments, reserves, or reinsurance. Otherwise, a series of nuclear accidents could force NML into bankruptcy. Currently, NML member utilities are concerned about increasing retrospective premium adjustments because of uncertainties about whether a state public utility commission would allow the cost of a retrospective premium adjustment to be passed through to the ratepayer. No precedents exist upon which to predict a state commission's rulemaking decision for retrospective premiums.

6.4.2 Quota Sharing

Quota sharing is a common practice in the insurance industry. In quota sharing, each insurer is responsible for its proportionate share of any covered loss. If applied to nuclear insurance, a utility could elect to purchase property insurance coverage from both ANI-MAERP and NML. With quota sharing, coverage from the obtained resources of the two insurers would double the property insurance available if the utility elected to obtain the maximum amount.

A study of nuclear insurance options by the General Accounting Office[11] describes several problems that exist with respect to quota sharing. These include:

- reduced premium flows to both ANI-MAERP and NML
- possible violations of antitrust laws
- differing safety and contract standards between ANI-MAERP and NML
- increased exposure of reinsurers who insure both ANI-MAERP and NML.

Assuming that these difficulties can be resolved, quota sharing represents one possibility for increasing property insurance coverage.

6.4.3 Multi-Layer Insurance Coverage

Multi-layer insurance coverage is the use of one insurance carrier or funding mechanism to provide coverage up to a given level, followed by the use of another carrier or mechanism to provide coverage in excess of the first level up to a second level, etc. The nuclear liability insurance coverage fostered under the Price-Anderson Act is an example of multi-layer coverage.

The Price-Anderson Act divides liability insurance coverage into three layers:

- the maximum amount of liability insurance available from private sources (first layer)
- required utility industry self-insurance (second layer)
- federal government indemnity (third layer).

First layer liability insurance coverage of $160 million is provided by two insurance pools in the private sector--ANI and Mutual Atomic Energy Liability Underwriters (MAELU). The two insurance pools are composed of private insurance companies who have voluntarily pledged funds to either ANI or MAELU to provide this liability insurance coverage. Second layer liability coverage is provided by requiring the utilities to pay a specified retrospective premium assessment (currently $5 million for each operating nuclear unit) to cover losses in excess of those provided for by first layer insurance coverage. The total amount of second layer coverage is equal to the product of the specified premium assessment multiplied by the number of operating reactors. In the event of an accident which exhausts both the first and second layers of liability insurance, the federal government is liable for the third layer under the Price-Anderson Act. (Total offsite liability coverage under Price-Anderson is currently $560 million.)

Proposals made in 1981 by private insurers and the utility industry would provide multi-layer property damage insurance coverage of up to $1 billion without government participation.[11,19,20] A proposal was advanced in late spring of 1981 by ANI-MAERP that would provide three layers of property insurance coverage to the nuclear utilities.[19] The first layer would be the existing $450 million coverage available from either ANI-MAERP or NML. A second layer of $350 million insurance would be utility self-insurance in which a retrospective assessment would be collected if the first layer of coverage was exhausted by an accident. The third layer would be ANI-MAERP pre-paid insurance coverage for losses exceeding $800 million up to a maximum of $1 billion.

A somewhat similar plan was announced in September 1981 by the Edison Electric Institute as an extension of the NEIL industry self-insurance coverage for power replacement costs. The new NEIL program would provide two layers of coverage. The first layer of coverage would consist of property damage insurance currently available, assumed to total $500 million. (As noted above, property damage coverage of $450 million was available as of January 1982.) The second $500 million of insurance coverage would be underwritten by two sources. A portion of this second-layer coverage, $300 million, would be provided by the same sort of mutual arrangement as underwrites the NEIL replacement power cost insurance. The remaining $200 million would be purchased from commercial insurers such as ANI and MAERP.

The accident at TMI-2 in March 1979 showed that the then-available property damage insurance coverage was not adequate to fund the cleanup of a major nuclear accident. Both the insurance industry and the utility industry have taken steps since that accident to increase the insurance coverage. Multi-layer coverage appears to provide a viable option for increasing property damage coverage to adequate levels.

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CHAPTER 7

CHARACTERISTICS OF THE REFERENCE PWR POWER STATION

This chapter contains a brief description of the reference PWR power station. Included are descriptions of the reference site and the reference facility, together with estimates of the radioactivity in reactor components and structural materials as a result of neutron activation during normal reactor operation. Additional information about the reference site and the reference facility is given in Appendices A and B of Volume 2. The reference facility of this study is the same as that used for a previous study[1] of the decommissioning of a pressurized water reactor following normal shutdown. A more complete facility description can be found in the published report of the previous study.

The reference accident scenarios for this study, described in Chapter 8, contain information about radioactive contamination, physical damage to structures and equipment, and radiation exposure rates that result from the postulated accidents at the reference facility.

7.1 THE REFERENCE SITE

A reference site, described briefly in this section, is used in assessing the public safety effects of post-accident cleanup and decommissioning of the reference PWR. Site characteristics are representative of existing and potential nuclear reactor sites in the midwestern or middle southeastern United States. Additional information supporting this site description is found in Appendix A of Volume 2, which is developed from information contained in References 2 and 3.

Individual features of this reference site vary from those of any specific LWR site. However, it is believed that use of a reference site rather than a specific site results in a more meaningful overall analysis of potential impacts associated with decommissioning of nuclear power facilities. Site-specific assessments will be required, prior to actively decommissioning a specific facility, for the safety analysis and the environmental report that must accompany the request for license amendment.[4]

The reference site occupies 4.7 km^2 in a rectangular shape of 2 km by 2.35 km. A river of moderate size, with an average flow rate of 1420 m^3/sec, flows through one corner. Plant facilities are located inside a 0.12 km^2 fenced portion of the site. The minimum distance from the point of plant airborne releases to the outer site boundary is 1 km.

The site is located in a rural area that has relatively low population density. Higher population densities are located at distances of 20 to 60 kilometers, and gradually reducing population densities are encountered out to 180 kilometers. The closest moderately large city, population 40,000, is about 32 kilometers distant. The closest large city, population 1,800,000, is about 48 kilometers away. The total population within a radius of 80 kilometers is 3.52 million.

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

Prior to the postulated reactor accidents, the site is assumed to be slightly contaminated with radioactive material as a result of deposition from normal operating effluents over the 40-year plant operating life, assumed to be equivalent to 30 effective full power years (EFPY) of operation. Site contamination that results from the postulated accidents is discussed in Chapter 8.

7.2 THE REFERENCE PWR

The reference PWR that serves as the basis for post-accident cleanup and decommissioning analyses is described in this section. This brief description is derived from a more complete facility description contained in Reference 1.

The reference PWR is the Trojan nuclear plant at Rainier, Oregon, operated by the Portland General Electric Company. It is a 3500-MWt (1175-MWe) pressurized water reactor of Westinghouse design.

The principal systems, components, and structures of the reference PWR are described briefly in the sections that follow. More detailed information can be found in Appendix B of Volume 2 and in Reference 1. The information presented is based on the Trojan Final Safety Analysis Report,[5] the Westinghouse RESAR-3 Preliminary Safety Analysis Report,[6] the SNUPPS Preliminary Safety Analysis Report,[7] and other data furnished by personnel of the Portland General Electric Company.

7.2.1 Nuclear Power Generation System

The PWR nuclear power generation system is illustrated schematically in Figure 7.2-1.

In a PWR, water, maintained at conditions of pressure and temperature that prevent boiling (typically about 1.5×10^4 kPa and 320°C), circulates through the primary coolant loops of the reactor core and the steam generators. At

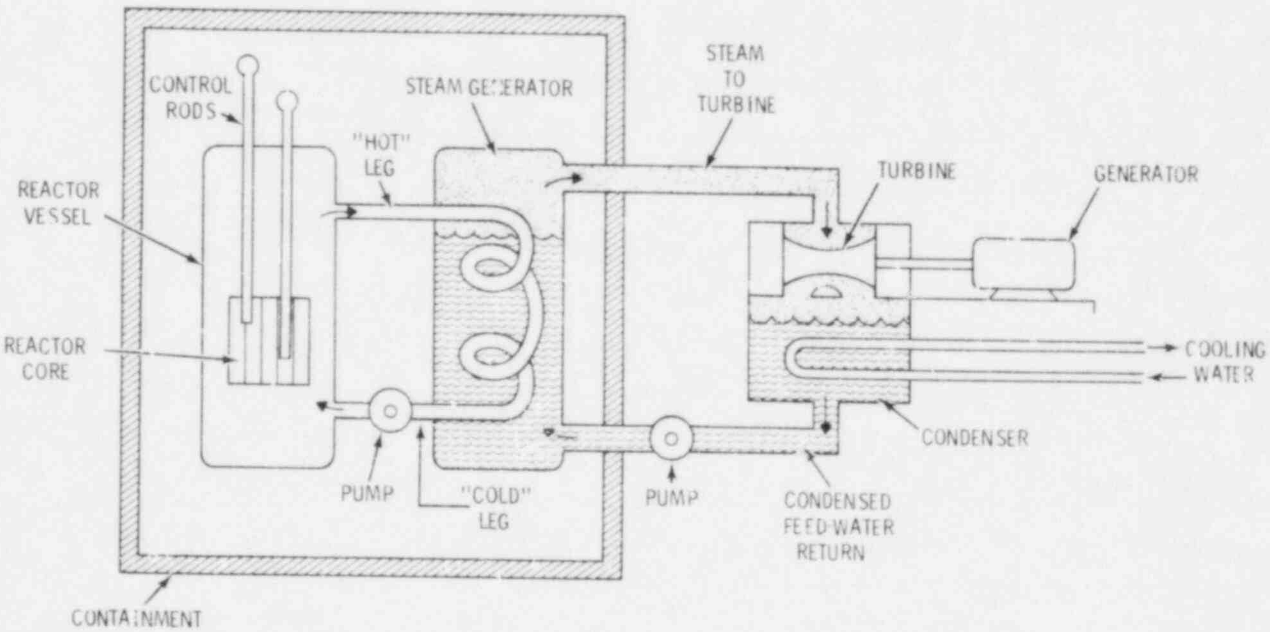


FIGURE 7.2-1. Schematic of PWR Nuclear Power Generation System

relatively lower pressures and temperatures (typically about 0.7×10^4 kPa and 260°C), steam for the turbines is produced in secondary heat exchange loops that are isolated from the nominally radioactive primary coolant system.

The nuclear steam supply system that includes the nuclear reactor, the steam generators, associated piping, valves, and auxiliary fluid systems is housed in a containment building designed to withstand the effects of nuclear accidents and severe natural catastrophies and to prevent the dispersal of radioactivity to the environment in the event of an accident.

7.2.2 Major Systems and Components

The principal systems and components of interest from a decommissioning standpoint are the reactor vessel, which contains the reactor core and coolant, and the reactor coolant system (RCS), which transfers the heat from the core to the secondary coolant system via the steam generator heat exchangers where steam is produced for use in the turbine generator.

Reactor Vessel

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, constructed of carbon steel, is about 0.216 m 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 and 4.6 m outer diameter. The mass of the vessel is nearly 400 Mg, empty.

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 that includes the core barrel and shroud, the lower core plate and support columns, and the neutron shield pads; the upper core support structure that includes the upper core support assembly, the upper core plate, and guide tube assemblies; and the in-core instrumentation support assemblies. The structures are made of 304 stainless steel and have a total mass of about 190 Mg.

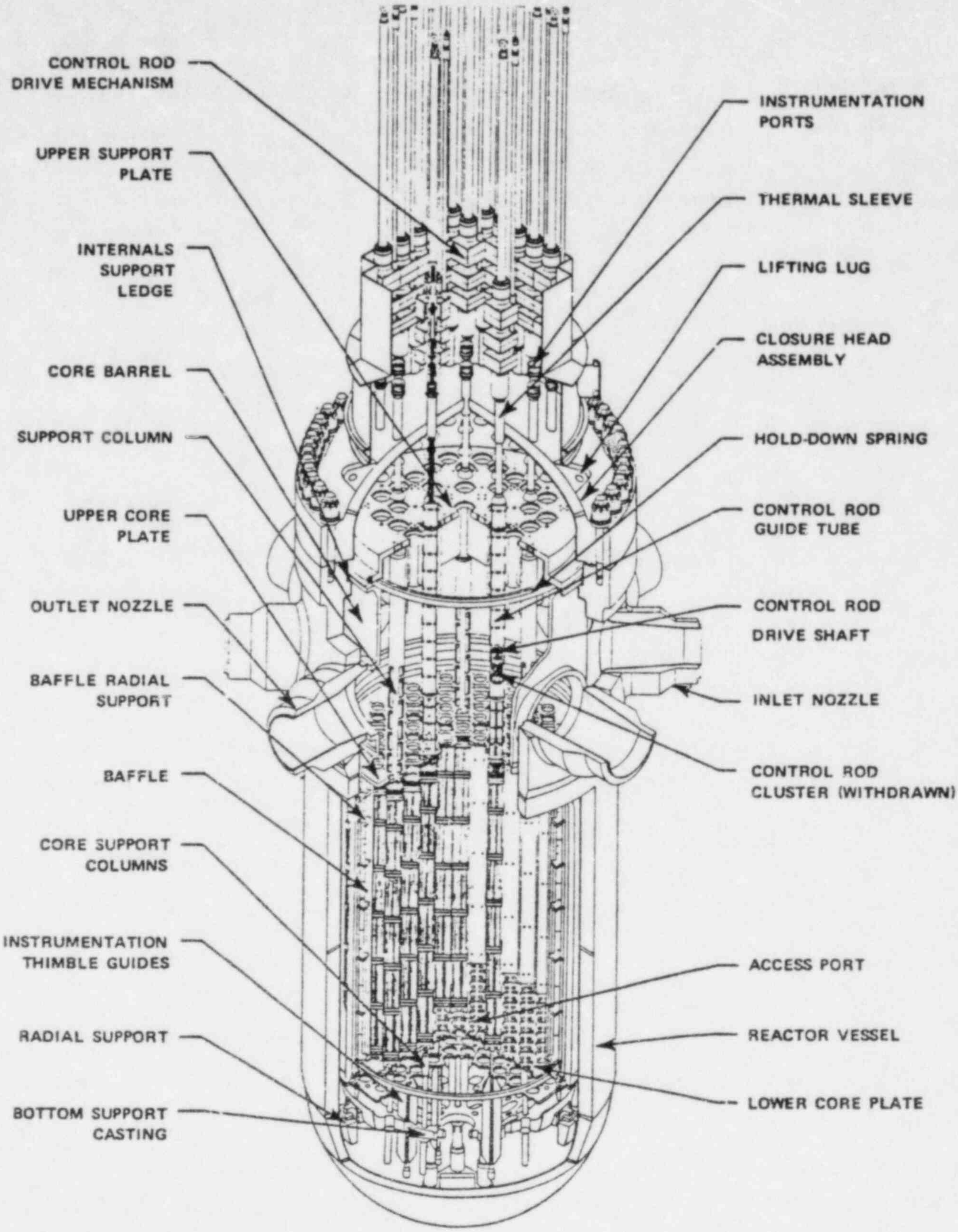


FIGURE 7.2-2. PWR Reactor Vessel and Internals

Reactor Coolant System

The reactor coolant system consists of four loops for transferring heat from the reactor to the secondary coolant system. (Two loops are shown schematically in Figure 7.2-3). Each loop contains a steam generator, a reactor coolant pump, and connecting piping. One loop also contains a pressurizer.

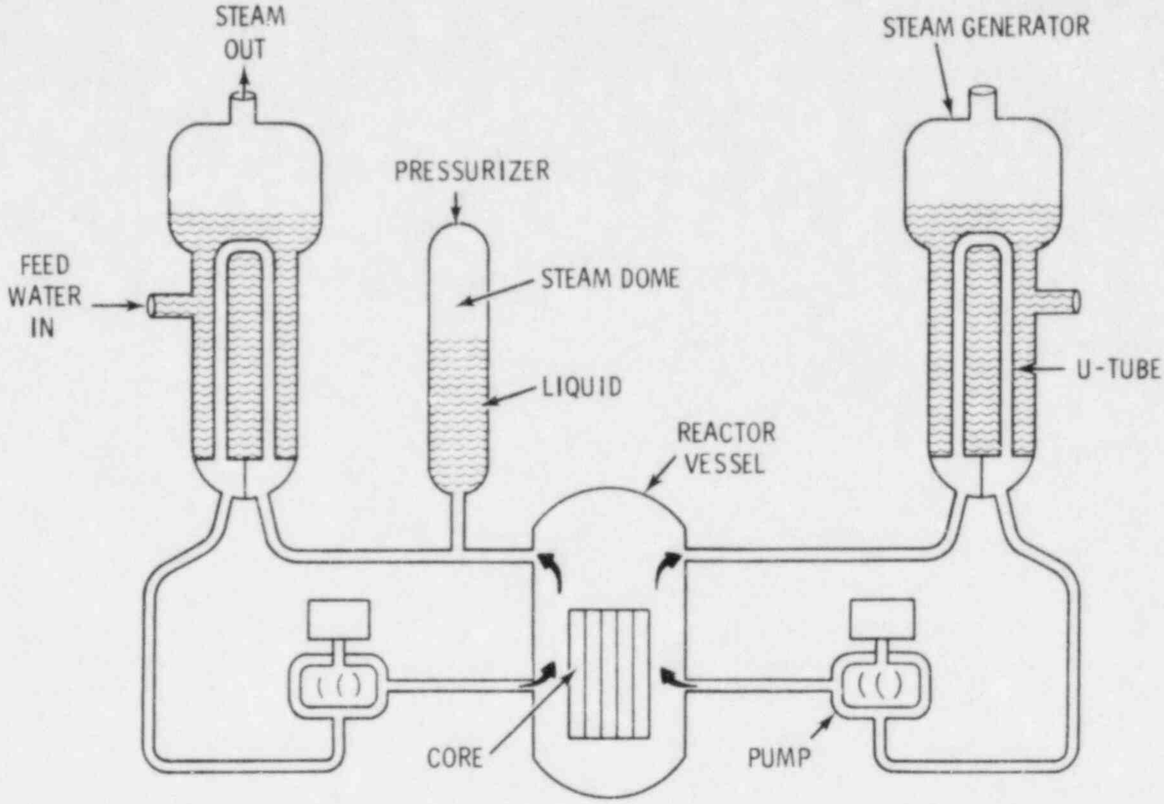


FIGURE 7.2-3. Schematic of PWR Reactor Coolant System (Two loops of four-loop system are shown.)

Each vertical U-tube steam generator 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. Each pump is about 8.7 m in overall height and 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 (∿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 a total mass of slightly over 100 Mg.

PWR steam supply systems are equipped with a pressurizer to maintain constant primary coolant pressure during operation and to limit pressure changes caused by coolant thermal expansion and contraction as plant loads change. The pressurizer 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, 2.3 m in outside diameter, and 88.7 Mg in mass.

7.2.3 Plant Structures

The arrangement of the structures on the reference PWR plant site is illustrated in Figure 7.2-4. The structures of primary interest during cleanup and decommissioning are those which contain radioactive material, i.e., the containment, fuel, auxiliary, and control buildings. All of the structures are discussed briefly in this section, with more detailed descriptions given in Appendix B of Volume 2.

Containment Building

The containment building is designed to house the nuclear steam supply system that includes the reactor vessel and internals, four steam generators, four reactor coolant pumps, the pressurizer, and the reactor coolant piping. A vertical section of the PWR containment building is shown in Figure 7.2-5. The structure is a right circular cylinder with a hemispheric top and a flat base. It is about 64 m in height and about 43 m in diameter. The building is constructed of reinforced concrete, with post-tensioned tendons in the cylindrical walls and dome, and is lined with a welded steel skin. Access to the containment is through a personnel lock and a 6.5-m-diameter equipment hatch at the operating floor (28.4 m) level. There is also an emergency personnel lock at the grade (13.7 m) level.

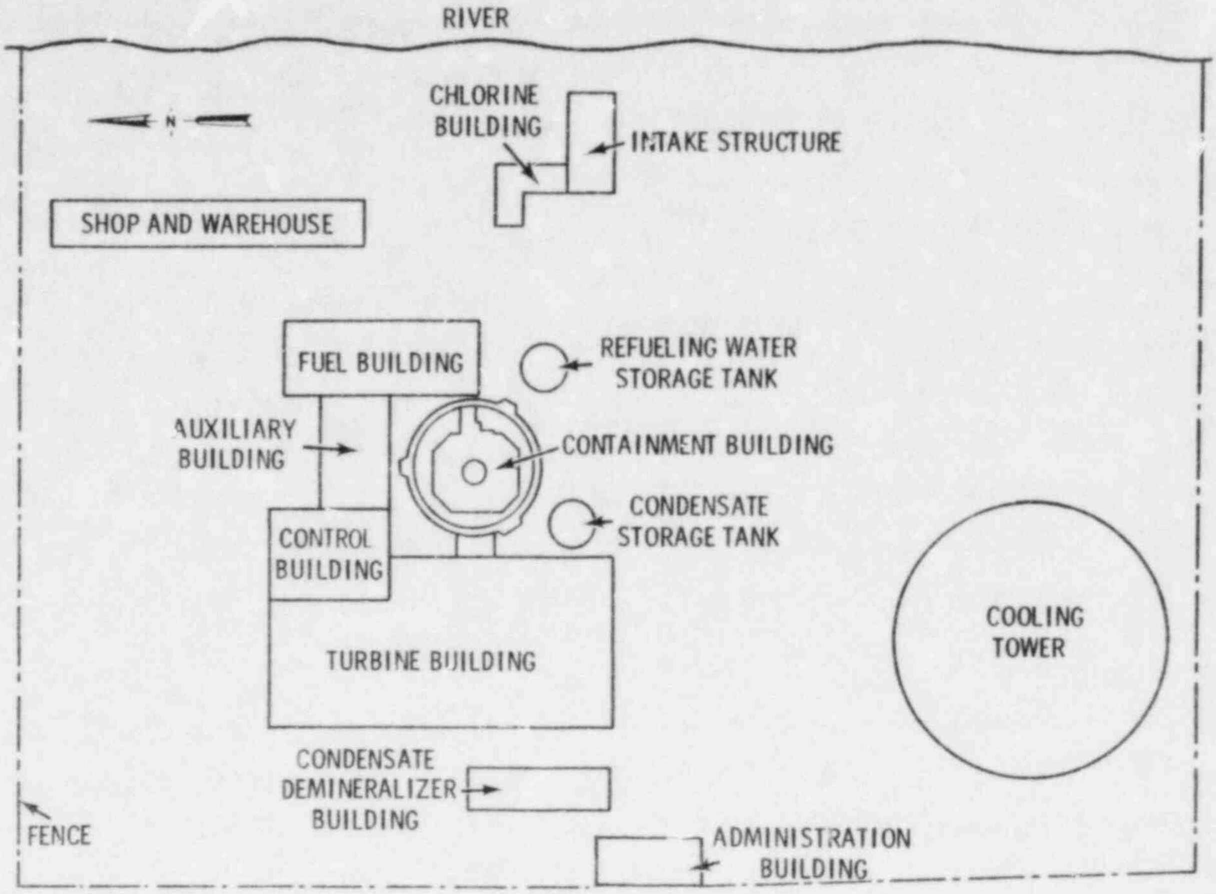


FIGURE 7.2-4. Site Layout of the Reference PWR Power Plant

Major interior structures include the biological shield, the steam generator and pressurizer cubicles, and the refueling cavity. These structural components are shown in Figure 7.2-6.

Fuel Building

The fuel building houses new and spent fuel storage and handling facilities, the makeup water treatment system, the chemical and volume control system, and the solid radioactive waste handling equipment. The building is a steel-frame and reinforced-concrete structure with four floors. It is approximately 27 m in height and has lateral dimensions of about 54 m by 19 m. Additional details of construction and diagrams showing the placement of equipment in this building are given in Appendix B of Volume 2.

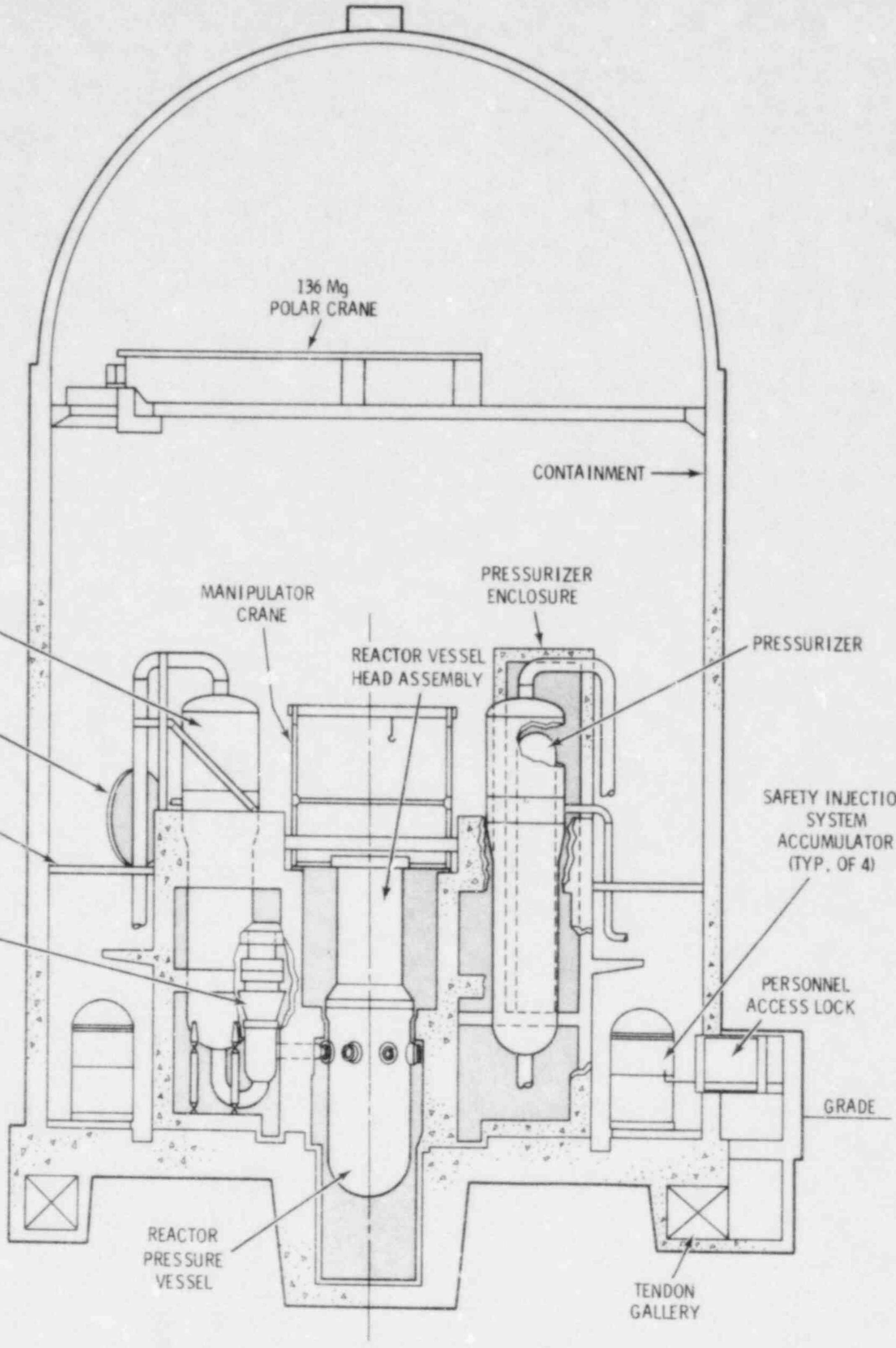
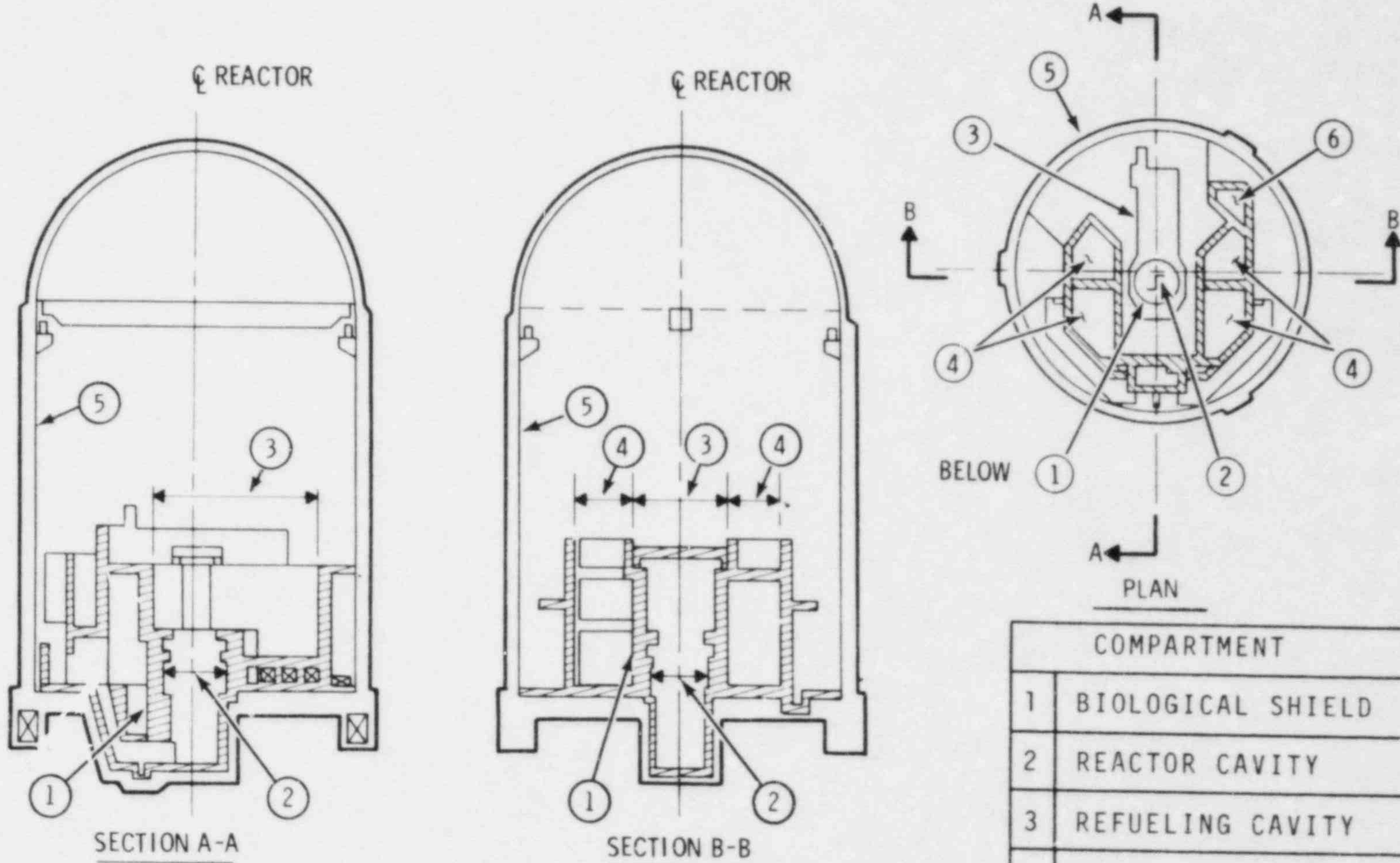


FIGURE 7.2-5. Vertical Section of PWR Containment Building



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

FIGURE 7.2-6. Major Structural Components of PWR Containment Building

Auxiliary Building

The principal systems contained in the auxiliary building include the liquid radioactive waste treatment systems, the filter and ion exchanger vaults, the waste-gas treatment system, and the ventilation equipment for the containment, fuel, and auxiliary buildings. The 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, has lateral dimensions of about 35 m by 19 m, and is structurally connected to the fuel building. Additional details of construction and diagrams showing the placement of equipment in this building are given in Appendix B of Volume 2.

Control Building

The control building houses the reactor control room, the cable spreading room, process control laboratories and counting rooms, and personnel facilities. It is a steel and reinforced-concrete structure with four floors above grade. The building is approximately 18 m in height, 31 m by 24 m in lateral dimensions, and is structurally connected to the auxiliary building. An isometric view showing the relation of the control building to the containment and auxiliary buildings is given in Appendix B of Volume 2.

Turbine Building

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. The building, framed with structural steel, has a reinforced concrete slab floor with the turbine pedestals poured into it at grade level and two operating floors above. The structure has lateral dimensions of about 95 m by 49 m and is about 33 m in height.

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 19,700 m^3 of water are contained in the reservoir beneath the cooling fins.

Other Structures

The remaining structures on the reference PWR site are of conventional construction. These structures are expected to remain uncontaminated under both normal and accident conditions. The chlorine building, a steel-framed structure on a concrete slab, contains the chlorination equipment for treating water coming from and being discharged into the river. The condensate demineralizer building has two levels of reinforced concrete below grade and one level of structural steel above grade; this building contains the condensate demineralizer ion exchangers and facilities for disposal of expended resins. The shop and warehouse is a single-story steel-frame structure on a concrete slab and contains the general machine shop, paint shop, warehouse, offices, lockers, and lunchroom. The administration building is a two-story steel-frame structure that contains the general administrative offices and the plant security control station.

7.3 RADIOACTIVE CONTAMINATION

Radionuclide inventories at the time of accident cleanup and decommissioning following a reactor accident result both from normal reactor operation prior to the accident and from accident-generated fission product contamination. Information about the radioactive contamination from both of these sources present in the reference PWR power plant is needed to assess the level of effort required to clean up and decommission the plant, as well as to evaluate the impacts of these activities on occupational and public safety.

The radioactive contamination present at the time of post-accident cleanup can be characterized as follows:

1) neutron-activated components in and surrounding the reactor core (from normal operation)
2) contamination from fission products and activated corrosion products deposited on inside surfaces of piping and equipment systems (from both normal operation and the accident)

3) fission product contamination on building and equipment surfaces (mostly from the accident)

4) sump water[(a)] and sludge contaminated with fission products (from the accident)

5) fission product contamination of reactor coolant system water (from the accident).

The fission product contamination resulting from the accident may also include uranium and transuranic contamination from fuel materials in the reactor core.

Immediately following a postulated accident, the radiological conditions generated by the accident are of considerably more concern than those resulting from normal operation. Data on the estimated total amount and isotopic composition of the fission product radioactivity released during an accident and on the average exposure rates resulting from this radioactivity are given with the reference accident descriptions in Chapter 8. However, as initial cleanup and decommissioning activities proceed, certain categories of the radioactive contamination resulting from normal operation become important because of: 1) the reductions in accident-generated contamination effected by decontamination efforts, and 2) changes in operations (e.g., removal of the reactor vessel and of the biological shield) being performed by decommissioning workers. Therefore, this section presents data on the estimated total amount and isotopic composition of the radioactive contamination resulting from normal plant operation and on the average exposure rates to decommissioning workers from this radioactivity. Analyses of worker exposure presented in this study take into account exposure contributions from all sources of contamination.

The isotopic compositions of radionuclide inventories from neutron activation, summarized in this section, are taken from Reference 1; additional information on neutron-activated components is contained in the reference.

(a) Sump water is reactor coolant system water that is released during an accident and that collects in the basement and in sumps of the reactor building.

7.3.1 Neutron-Activated Components

Radioactive material is produced in the structural components in and around reactor vessels because of neutron absorption during reactor operation. Three basic types of materials are used in and around reactor vessels: stainless steel (type 304), carbon steel (type SA 533), and reinforced concrete. This section contains summaries of estimated radionuclide inventories for and total radioactivity in neutron-activated components.

Concentrations of radioactivity in the reactor vessel and the surrounding shielding of the reference PWR are estimated assuming 30 effective full-power years (EFPY), equivalent to 40 calendar years at 75% of full-power operation. Neutron flux levels throughout the reference reactor vessel and surrounding shield are calculated using the multi-energy group transport theory code, ANISN.[8] The calculated neutron fluxes are then used in a series of calculations with the ORIGEN code[9] 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. Details of the calculational methods used to estimate the reference neutron activation product inventories are presented in Appendix C of Reference 1.

The calculated buildup of selected activation product radionuclides is illustrated in Figure 7.3-1, with the concentration of each radionuclide normalized to unity at 30 EFPY. The shorter-lived radionuclides, such as ^{55}Fe and ^{60}Co, reach an equilibrium concentration in less than 30 years of operation, while the concentrations of the long-lived radionuclides, such as ^{59}Ni and ^{94}Nb, increase almost linearly with increased irradiation time.

Estimated radionuclide inventories for neutron-activated materials in the reference PWR after 30 EFPY of operation are presented as follows: Table 7.3-1 for stainless steel (reference radionuclide inventory 1), Table 7.3-2 for carbon steel (reference radionuclide inventory 2), and Table 7.3-3 for reinforced concrete (reference radionuclide inventory 3). Only the major isotopic components with half-lives greater than 35 days are listed in the reference inventories. Reference radionuclide inventory 3 accounts for the radionuclides in both the concrete and the carbon-steel reinforcing material.

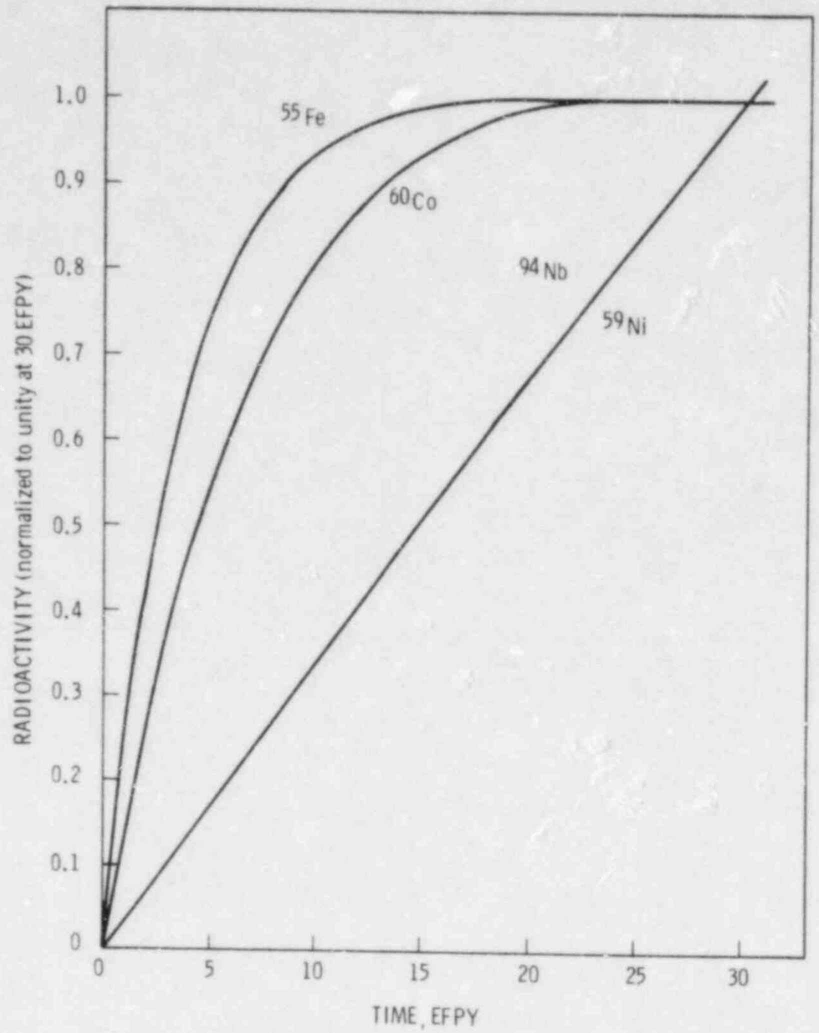


FIGURE 7.3-1. Calculated Buildup of Selected Activation Products in Core Internals as a Function of Time at Full Power

These inventories are estimated using the calculated thermal neutron flux distribution at the axial midplane of the nuclear fuel zone. They therefore represent maximum values of neutron-induced radioactivity after 30 EFPY.

7.3.2 Neutron-Activated Corrosion Products

Corrosion products from structural components of reactor coolant systems become activated by neutron absorption during reactor operation. These activated corrosion products are present in reactor coolant streams and tend to deposit on the inner surfaces of piping systems, creating distributed sources of radiation throughout the facility.

TABLE 7.3-1. Reference Radionuclide Inventory 1 (Stainless Steel Activation Products)[a,b]

Radionuclide	Half-Life	Radioactivity Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at Shutdown
^{14}C	5730 yr	1.5×10^{2}	5.0×10^{-5}
^{54}Mn	303 day	6.8×10^{4}	2.6×10^{-2}
^{55}Fe	2.7 yr	1.3×10^{6}	4.9×10^{-1}
^{59}Fe	45 day	4.6×10^{4}	1.7×10^{-2}
^{58}Co	72 day	1.5×10^{5}	5.7×10^{-2}
^{60}Co	5.27 yr	9.6×10^{5}	3.6×10^{-1}
^{59}Ni	8×10^{4} yr	7.4×10^{2}	2.8×10^{-4}
^{63}Ni	92 yr	1.2×10^{5}	4.5×10^{-2}
^{65}Zn	245 day	1.2×10^{2}	4.5×10^{-5}
^{93}Mo	3.5×10^{3} yr	3.6×10^{-1}	1.4×10^{-7}
^{94}Nb	2×10^{4} yr	5.4×10^{0}	2.0×10^{-6}
Totals		3.0×10^{6}	1.0

(a) Calculated at the inner surface of the stainless steel core shroud, at the axial midplane of the fuel zone, for 30 EFPY of operation.
(b) Summarized from Chapter 7 of Reference 1.

TABLE 7.3-2. Reference Radionuclide Inventory 2 (Carbon Steel Activation Products)[a,b]

Radionuclide	Half-Life	Radioactivity Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at Shutdown
^{14}C	5730 yr	1.9×10^{-2}	2.1×10^{-5}
^{54}Mn	303 day	4.7×10^{1}	5.3×10^{-2}
^{55}Fe	2.7 yr	7.2×10^{2}	8.2×10^{-1}
^{59}Fe	45 day	2.7×10^{1}	3.1×10^{-2}
^{58}Co	72 day	6.6×10^{0}	7.5×10^{-3}
^{60}Co	5.27 yr	7.5×10^{1}	8.5×10^{-2}
^{59}Ni	8×10^{4} yr	3.2×10^{-2}	3.6×10^{-5}
^{63}Ni	92 yr	3.8×10^{0}	4.3×10^{-3}
^{93}Mo	3.5×10^{3} yr	1.3×10^{-3}	1.5×10^{-6}
Totals		9.0×10^{2}	1.0

(a) Calculated at the inner surface of the carbon steel reactor vessel, at the axial midplane of the fuel zone, for 30 EFPY of operation.
(b) Summarized from Chapter 7 of Reference 1.

TABLE 7.3-3. Reference Radionuclide Inventory 3 (Neutron-Activated Concrete)[a,b]

Radionuclide	Half-Life	Radioactivity Concentration at Shutdown (Ci/m^3)	Fractional Radioactivity at Shutdown
^{39}Ar	269 yr	3.96×10^{-2}	1.14×10^{-3}
^{41}Ca	8×10^{4} yr	7.00×10^{-3}	2.01×10^{-4}
^{45}Ca	165 day	3.66×10^{0}	1.05×10^{-1}
^{54}Mn[c]	303 day	1.68×10^{-1}	4.83×10^{-3}
^{55}Fe[c]	2.7 yr	3.01×10^{1}	8.65×10^{-1}
^{60}Co[c]	5.27 yr	6.69×10^{-1}	1.92×10^{-2}
^{59}Ni[c]	8×10^{4} yr	1.19×10^{-3}	3.42×10^{-5}
^{63}Ni[c]	92 yr	1.40×10^{-1}	4.02×10^{-3}
Totals		3.48×10^{1}	1.0

(a) Calculated at the inner surface of the biological shield, at the axial midplane of the fuel zone, for 30 EFPY of operation.
(b) Summarized from Chapter 7 of Reference 1.
(c) Due largely to structural steel in the biological shield.

The isotopic composition of activated corrosion products is derived from information available in the literature. The reference radionuclide inventory of activated corrosion products, taken from Reference 1, is based on radionuclide deposits in a PWR steam generator.[10] This radionuclide inventory (reference radionuclide inventory 4) is shown in Table 7.3-4.

7.3.3 Exposure Rates Due to Contamination Resulting from Normal Operations

Estimated contact exposure rates from neutron-activated components and estimated average radiation exposure rates at selected locations within the reactor facility due to contamination resulting from normal operations are presented in this section. These exposure rates are used in combination with those developed in Chapter 8 to estimate worker dose rates during post-accident cleanup and decommissioning operations.

TABLE 7.3-4. Reference Radionuclide Inventory 4 (Neutron-Activated Corrosion Products)[a]

Radionuclide	Half-Life	Fractional Radioactivity at Shutdown
^{51}Cr	28 day	2.4×10^{-2}
^{54}Mn	303 day	3.6×10^{-2}
^{59}Fe	45 day	8.2×10^{-3}
^{58}Co	72 day	4.6×10^{-1}
^{60}Co	5.27 yr	3.2×10^{-1}
^{95}Zr	65 day	5.6×10^{-2}
^{95}Nb	35 day	5.6×10^{-2}
^{103}Ru	40 day	2.6×10^{-2}
^{137}Cs	30 yr	1.2×10^{-3}
^{141}Ce	284 day	6.6×10^{-2}
Total		1.0

(a) Summarized from Chapter 7 of Reference 1.

7.3.3.1 Exposure Rates for Neutron-Activated Components

The computed concentrations of radionuclides in the highly activated reactor pressure vessel and its components and in the concrete shield that surrounds the pressure vessel were used in calculations to estimate radiation exposure levels that might be encountered during the removal and disposal of these components. Details of exposure rate calculations for the reference PWR are given in Chapter 7 and Appendix C of Reference 1. The results of these calculations are shown in Table 7.3-5.

The decay of ^{60}Co controls the exposure rate from neutron-activated components for about the first 80 years following reactor shutdown. After that time, the exposure rate is increasingly dominated by the decay of ^{94}Nb.

TABLE 7.3-5. Estimated Radiation Exposure Rates from Neutron-Activated Components in the Reference PWR(a-c)

Component	Calculated Exposure Rates (R/hr)			
	^{60}Co (gamma)	^{94}Nb (gamma)	^{55}Fe (IB(d), gamma)	^{59}Ni (IB(d), gamma)
Core Shroud				
Inner Surface	(1.9 to 5.6) x 10^{5}	2.0 x 10^{0}	1.1 x 10^{-1}	9.1 x 10^{-2}
Outer Surface	(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 Surface	(2.6 to 7.9) x 10^{4}	1.7 x 10^{-1}	5.7 x 10^{-3}	2.1 x 10^{-2}
Outer Surface	(2.9 to 9.7) x 10^{3}	5.9 x 10^{-2}	1.3 x 10^{-3}	6.4 x 10^{-3}
Pressure Vessel				
Inner Surface	(2.3 to 5.4) x 10^{2}	--(e)	3.4 x 10^{-4}	1.6 x 10^{-4}
Outer Surface	(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) From Reference 1.
(b) Calculated at the time of final reactor shutdown.
(c) Exposure rates calculated at a distance of 10 mm from the surface of the activated material, at the vertical center line of the reactor core.
(d) IB means "inner bremsstrahlung".
(e) A dash means that radionuclide was not present in that component.

7.3.3.2 Exposure Rates Throughout the Plant Due to Contamination Resulting from Normal Operations

Assumed exposure rates at selected locations in the reference PWR due to contamination resulting from normal operations are given in Appendix C of Reference 1. The exposure rates are a composite of data from six commercial PWRs that had operated for 3 to 6 years. Both contact and general area exposure rate data are presented in Reference 1.

Estimated exposure rates from contamination resulting from normal operations at the reference PWR vary over a range from 0.001 to 30 R/hr, depending on location. The highest exposure rates are contact rates measured at external surfaces of steam generators, heat exchangers, and reactor coolant pumps. General area dose rates vary from about 0.001 to 0.2 R/hr except in the reactor cavity where rates as high as 1 R/hr are postulated.

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CHAPTER 8

REFERENCE ACCIDENT SCENARIOS AND RESULTANT CONTAMINATION LEVELS AT A REFERENCE PWR

Accident scenarios that provide a basis for the post-accident cleanup and decommissioning cost and safety estimates of subsequent chapters are discussed in this chapter. From the viewpoint of this study, which deals with post-accident cleanup and decommissioning, the consequences of an accident (i.e., the radiological and physical condition of the plant following an accident) are much more important than the sequence of events that occur during the accident. Therefore, detailed descriptions of accident sequences are not given. The reference accident scenarios provide information about radioactive contamination, physical damage, and radiation exposure rates that result from the postulated accidents. An effort has been made to postulate consequence scenarios that are credible in terms of initiating circumstances.

Safety Analysis Reports prepared as part of the licensing process for operating reactors were reviewed for information about postulated credible accidents. In addition, NRC-issued safety evaluation reports and licensing criteria were reviewed. These documents include descriptions of a spectrum of loss-of-coolant accidents, including small steam and liquid line breaks, the inadvertent opening of safety or relief valves, and major ruptures of pipes containing reactor coolant. Postulated loss-of-coolant accidents involving the rupture of a main coolant line are generally assumed to impose the most severe demands on emergency core cooling system (ECCS) performance and the most serious consequences in terms of radioactive contamination and physical damage to buildings and equipment.[1-4] In accidents postulated for Safety Analysis Reports, the ECCS is assumed to operate effectively to limit fuel damage that could result in a release of fission product radioactivity. However, the accident at Three Mile Island Unit 2 (TMI-2)[5] illustrates that core damage and radioactive contamination of the containment can occur if emergency core cooling does not function properly either as a result of system malfunction or operator action.

For this post-accident decommissioning study, accident (i.e., consequence) scenarios are chosen that are illustrative of a range of cleanup and decommissioning requirements resulting in occupational radiation doses and cleanup and decommissioning costs that are substantially greater than those estimated in the previous study of PWR decommissioning following normal shutdown.[6] Accident scenarios with minor consequences that do not significantly impact the requirements and costs of cleanup and decommissioning are not considered. Scenarios are not restricted to accidents that have occurred, but the scenarios chosen for this study are believed to be credible with respect to initiating circumstances, and to be in agreement with scenarios currently considered as design basis by the NRC.[a]

The three accident scenarios chosen for analysis are described in the following paragraphs. An accident that results in extensive radioactive contamination and major physical damage inside the containment building (such as might occur as a result of a major coolant pipe rupture) is the most serious accident considered. An accident in which the partially molten core melts through the reactor pressure vessel is not generally considered to be a credible accident and is not included in this report.

The postulated accident scenarios, listed in increasing order of difficulty of the post-accident cleanup and decommissioning, are:

1. A small LOCA (a small steam line break or the inadvertent opening of a safety or relief valve) in which emergency core cooling functions to cool the core and limit the release of radioactivity. Some fuel cladding rupture is postulated, but no fuel melting. The consequence scenario includes a small amount of contaminated water in sumps and on floors and moderate contamination of the containment building.
2. A small LOCA in which emergency core cooling is delayed, resulting in 50% fuel cladding failure and a small amount of fuel melting.

(a) As this report is being completed, the NRC is performing research to assess behavior of nuclear power plant systems under a range of severe accident conditions, including a program to determine the radiological source term under these conditions.[7] As information from this program is developed, it should be used in potential addenda to this report.

The consequence scenario includes extensive radioactive contamination of the containment building, moderate radioactive contamination of the auxiliary and fuel buildings, and minor physical damage to buildings and equipment.

3. A major LOCA in which emergency core cooling is delayed, resulting in 100% fuel cladding failure and significant fuel melting and core damage. The consequence scenario includes severe radioactive contamination of the containment building, moderate radioactive contamination of the auxiliary and fuel buildings, and major physical damage to structures and equipment.

The three reference accident scenarios are summarized in Table 8.0-1 and are described in more detail in the following sections. For the scenario 2 accident, 5% fuel melting is postulated. This assumption is believed to be conservative for the case of a small LOCA with delayed ECCS. For the scenario 3 accident, 50% fuel melting is postulated. This is consistent with the value assumed for fuel melting in the severe core damage accident with delayed ECCS operation analyzed in an NRC study of fission product behavior during an LWR accident.(8,9)

TABLE 8.0-1. Summary of Accident Parameters for the Reference Reactor Accidents

Accident Parameter	Parameter Value Scenario 1	Scenario 2	Scenario 3
Accident description	Small LOCA; ECCS functions	Small LOCA; ECCS delayed	Large LOCA; ECCS delayed
% fuel cladding failure	10	50	100
% fuel melting	0	5	50
Reactor coolant released (m^3)	200	1000	1600
Core damage	Minor	Moderate	Severe
Physical damage to reactor building	None	Minor damage to valves and equipment. Contamination of building ventilation system.	Loss of electrical and other services. Major damage to building components.

The volume of reactor coolant released for the scenario 1 accident is approximately equal to one reactor coolant system volume, exclusive of the reactor vessel. The volume of coolant released for the scenario 3 accident is approximately equal to the volume contained in the refueling water storage tank which also serves as the source of coolant for the ECCS. For the scenario 2 accident, the volume of coolant released is chosen to be intermediate between the volumes for the scenario 1 and scenario 3 accidents.

Site contamination from the reference accidents is assumed to be negligible. Following the accident at TMI-2, thousands of environmental samples were collected and analyzed by various agencies. An NRC Special Inquiry Group review(5) of sampling results from air, water, milk, vegetation, and soil samples in the vicinity of the TMI site showed that only very low levels of radioiodines and radioxenons could be attributed to releases from the accident. The trace quantities of radiocesium and radiostrontium found in a few samples could be attributed to and were consistent with global fallout from previously conducted nuclear weapons tests. It was therefore concluded that radioactive releases to the environment as a result of the accident at TMI-2 were limited to the noble gas radionuclides and a small quantity of radioiodines. An NRC study of fission product behavior during LWR accidents(8) concludes that aerosol removal mechanisms within containment will substantially reduce atmospheric releases if containment failure is delayed. Analysis of a severe core damage accident in which 50% core melt was postulated to result from delayed ECCS operation showed that an attenuation factor greater than 100,000 could be expected for release from containment of all fission products except the noble gases if no loss of containment integrity and operability of containment safety features are assumed. Therefore, in this study, at the time of post-accident cleanup and decommissioning, no contamination of the site is assumed beyond that which would be present as a result of 30 to 40 years of normal reactor operation.

The fission product source term that provides the basis for estimates of radioactive contamination following the postulated accidents is described in Section 8.1. Release fractions and estimated inventories of fission products released from the fuel core as the result of the reference accidents are described in Section 8.2. Details of the fission product contamination, gamma radiation exposure rates, fuel core damage, and damage to the containment building for the reference PWR accident scenarios are given in Section 8.3.

8.1 FISSION PRODUCT SOURCE TERM

The contamination condition of a nuclear power plant following a reactor accident is related to the radioactivity released from the reactor fuel core during the accident. The radioactivity release is dependent on the fission product source inventory (i.e., on the amount of fission product radioactivity in the fuel at the time of a postulated accident) and on the amount of fuel cladding failure and fuel melting that occurs in the accident.

The fission product source inventory used in this study is taken from the Reactor Safety Study (RSS)[3] and is shown in Table 8.1-1. The RSS fission product inventory was chosen because the RSS provides activity values and release fractions for several different categories of radionuclides, including noble gases, halogens, alkali metals, alkaline earths, rare earths, and transuranics. The use of this inventory as the bases for estimating fission product contamination following the accidents postulated for this study is believed to provide radionuclide data of sufficient accuracy to allow reasonable estimates of radiation exposure to be made for the reference accidents. The RSS fission product source inventory was calculated by means of the ORIGEN[10] program for a 1000-MWe (3200-MWt) three-region PWR core. It was assumed in the calculation that the three regions of the core operate at a constant specific power density of 40 kW/kg of uranium charged. The inventory was calculated for an equilibrium core initially charged with 3.3% enriched uranium at a time when the three regions have average burnups of 8800, 17,600, and 24,000 megawatt days per metric ton of uranium charged. This is equivalent to operation of each one-third of the core for 1, 2, and 3 years, respectively, at an assumed load factor of 0.75.

The fission product source inventory in Table 8.1-1 includes 54 isotopes. All other radioisotopes, because of their very short half-lives or the small quantity present, are judged to be insignificant compared to the isotopes considered.

TABLE 8.1-1. Initial Activity of Radionuclides in the Reactor Core at the Time of the Hypothetical Accidents[a]

Radionuclide	Half-Life (days)	Radioactive Inventory[b] (Ci)
^{58}Co	7.10×10^{1}	7.8×10^{5}
^{60}Co	1.92×10^{3}	2.9×10^{5}
^{85}Kr	3.95×10^{3}	5.6×10^{5}
^{85m}Kr	1.83×10^{-1}	2.4×10^{7}
^{87}Kr	5.28×10^{-2}	4.7×10^{7}
^{88}Kr	1.17×10^{-1}	6.8×10^{7}
^{86}Rb	1.87×10^{1}	2.6×10^{4}
^{89}Sr	5.21×10^{1}	9.4×10^{7}
^{90}Sr	1.10×10^{4}	3.7×10^{6}
^{91}Sr	4.03×10^{-1}	1.1×10^{8}
^{90}Y	2.67×10^{0}	3.9×10^{6}
^{91}Y	5.90×10^{1}	1.2×10^{8}
^{95}Zr	6.52×10^{1}	1.5×10^{8}
^{97}Zr	7.10×10^{-1}	1.5×10^{8}
^{95}Nb	3.50×10^{1}	1.5×10^{8}
^{99}Mo	2.80×10^{0}	1.6×10^{8}
^{99m}Tc	2.50×10^{-1}	1.4×10^{8}
^{103}Ru	3.95×10^{1}	1.1×10^{8}
^{105}Ru	1.85×10^{-1}	7.2×10^{7}
^{106}Ru	3.66×10^{2}	2.5×10^{7}
^{105}Rh	1.50×10^{0}	4.9×10^{7}
^{127}Te	3.91×10^{-1}	5.9×10^{6}
^{127m}Te	1.09×10^{2}	1.1×10^{6}
^{129}Te	4.80×10^{-2}	3.1×10^{7}
^{129m}Te	3.40×10^{-1}	5.3×10^{6}
^{131m}Te	1.25×10^{0}	1.3×10^{7}
^{132}Te	3.25×10^{0}	1.2×10^{8}
^{127}Sb	3.88×10^{0}	6.1×10^{6}
^{129}Sb	1.79×10^{-1}	3.3×10^{7}
^{131}I	8.05×10^{0}	8.5×10^{7}

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TABLE 8.1-1. (contd)

Radionuclide	Half-Life (days)	Radioactive Inventory[b] (Ci)
^{132}I	9.58×10^{-2}	1.2×10^{8}
^{133}I	8.75×10^{-1}	1.7×10^{8}
^{134}I	3.66×10^{-2}	1.9×10^{8}
^{135}I	2.80×10^{-1}	1.5×10^{8}
^{133}Xe	5.28×10^{0}	1.7×10^{8}
^{135}Xe	3.84×10^{-1}	3.4×10^{7}
^{134}Cs	7.50×10^{2}	7.5×10^{6}
^{136}Cs	1.30×10^{1}	3.0×10^{6}
^{137}Cs	1.10×10^{4}	4.7×10^{6}
^{140}Ba	1.28×10^{1}	1.6×10^{8}
^{140}La	1.67×10^{0}	1.6×10^{8}
^{141}Ce	3.23×10^{1}	1.5×10^{8}
^{143}Ce	1.38×10^{0}	1.3×10^{8}
^{144}Ce	2.84×10^{2}	8.5×10^{7}
^{143}Pr	1.37×10^{1}	1.3×10^{8}
^{147}Nd	1.11×1^{1}	6.0×10^{7}
^{239}Np	2.35×10^{0}	1.6×10^{9}
^{238}Pu	3.25×10^{4}	5.7×10^{4}
^{239}Pu	8.9×10^{6}	2.1×10^{4}
^{240}Pu	2.4×10^{6}	2.1×10^{4}
^{241}Pu	5.35×10^{3}	3.4×10^{6}
^{241}Am	1.5×10^{5}	1.7×10^{3}
^{242}Cm	1.63×10^{2}	5.0×10^{5}
^{244}Cm	6.63×10^{3}	2.3×10^{4}
Total		5.1×10^{9}

(a) From NUREG 75/014, Table VI 3-1.
(b) Calculated by means of the ORIGEN program for a 1000-MWe (3200-MWe) PWR. Inventories were calculated for an equilibrium core initially charged with 3.3% enriched uranium at a time when the three regions have average burnups of 8.8, 17.6, and 26.4 MWd/kg of uranium charged.

8.2 ESTIMATED FISSION PRODUCT RELEASES FOR REFERENCE ACCIDENTS

Estimated fission product releases from accident-damaged fuel are given in this section. The use of release fractions to estimate fission product releases is described in Section 8.2.1. Inventories of fission products released from the fuel core in the reference accidents are given in Section 8.2.2. These fission product inventories provide the basis for estimates of radioactive contamination and radiation exposure rates presented in Section 8.3.

8.2.1 Release Fractions

Fission product release from core material as a result of accidents considered in this study can be defined in terms of two release components:

1. Gap release - fission product release that occurs when the fuel cladding experiences rupture. It consists mostly of the radioactivity that was released to void spaces within the fuel rods during normal reactor operation, and the rapid depressurization of contained gases provides the driving force for escape.
2. Fuel-melt release - fission product release that occurs when fuel is heated to a temperature at which it becomes molten.

The fractions of gaseous and volatile fission products that escape from the reactor core during an accident depend on the fuel temperature and on specific accident conditions. Thermal analyses of fuel rod behavior during temperature increases that could result in cladding failure and fuel melting provide only generalized data on core temperature profiles, geometry changes, and rod behavior versus time. This fact, plus the uncertainties that exist in gap and meltdown release fractions, argue against construction in this study of a detailed model to calculate the release of fission products from the fuel core during an accident. In this study, the fission product release during an accident is treated as being simply proportional to the fraction of fuel rods in the core that experience cladding failure or fuel melting. This procedure for modeling fission product release is consistent with that used in the Reactor Safety Study.(3)

For a particular fission product, the total release fraction (i.e., the fraction of the total fuel core inventory of that fission product released from the fuel in an accident) is given by

$$RF_i = \left(G_i \times F_c\right) + \left(M_i \times F_m\right) \qquad (8.1)$$

where:

RF_i = total release fraction for the i^{th} fission product

G_i = fraction of fission product activity of the i^{th} fission product present in the gap between the fuel and the cladding and released as a result of cladding failure (the gap release fraction)

F_c = fraction of fuel rods experiencing cladding failure

M_i = fraction of fission product activity of the i^{th} fission product released as a result of fuel rod melting (the rod-melt release fraction)

F_m = fraction of fuel rods experiencing fuel melting.

Gap release fractions and rod-melt release fractions are shown in Table 8.2-1. These release fractions are taken from the Reactor Safety

TABLE 8.2-1. Estimated Core Release Fractions[a]

Fission Product	Gap Release Fraction	Rod-Melt Release Fraction[b]
Noble Gases - Xe, Kr	0.030	0.870
Halogens - I, Br	0.017	0.883
Alkali Metals - Cs, Rb	0.050	0.760
Tellurium - Te, Se, Sb	0.0001	0.150
Alkaline Earths - Sr, Ba	0.000001	0.100
Noble Metals - Ru, Rh, Pd, Mo, Tc	--	0.030
Rare Earths - Y, La, Ce, Pr, Nd, Pm, Sm, Eu, Np, Pu	--	0.003
Refractory Oxides - Zr, Nb	--	0.003

(a) From NUREG-75/014, Table VII 1-6.
(b) To obtain the rod-melt release fraction, account is taken of the fact that the fraction of the fuel rod inventory experiencing gap release is then not available for rod-melt release.

Study.[3] The fractions of fuel rods experiencing cladding failure or fuel melting are shown in Table 8.0-1 for each of the three accident scenarios.

Estimates of rod-melt release fractions are also made in the NRC study of fission product behavior during LWR accidents.[8] Estimated release fractions for Te and Sb are considerably higher (1.0 compared to 0.15) in the NRC fission product behavior study than they are in the Reactor Safety Study. The estimated release fraction for Sr is also higher (0.3 compared to 0.1) in the fission product behavior study than it is in the Reactor Safety Study. Comparable release fraction values are estimated for other radionuclides of interest such as ^{134}Cs, ^{137}Cs, the rare earths, and the noble metals. Because the fission product inventories used in this study are from the Reactor Safety Study and because the NRC fission product behavior study does not provide release fraction data for all of the fission product radionuclides of interest, the decision was made to use RSS release fraction data. The use of RSS data to estimate fission product contamination following the postulated accidents is believed to result in reasonable radiation exposure estimates.

8.2.2 Fission Product Radionuclide Releases for Reference Accidents

Fission product radionuclide releases for the three reference accidents are shown in Tables 8.2-2, 8.2-3, and 8.2-4, and represent the quantities of the various fission product nuclides released from the fuel core during the reference accidents. The quantity of radioactivity released from the fuel core during an accident is estimated by multiplying each fission-product source term, shown in Table 8.1-1, by the appropriate total release fraction, defined by Equation 8.1. Details of fission product radionuclide release calculations are given in Appendix C.

During the reference accidents, nearly complete release of the noble gases and halogens from all fuel that is severely damaged is postulated. Immediately after an accident, these radioisotopes may pose a threat to the population in the vicinity of the damaged reactor. They also present a difficulty for personnel attempting to assess conditions inside the containment. However, cleanup operations inside the containment building will be delayed for several months to a few years while planning and preparation activities, described in

TABLE 8.2-2. Estimated Inventory of Radioactivity Released from Damaged Reactor Fuel During Scenario 1 Accident[a]

Radionuclide	Activity Released From Damaged Fuel During Accident (Ci)	Activity After 1 Year (Ci)
^{58}Co	--	--[b]
^{60}Co	--	--
^{85}Kr	1.7×10^{3}	1.6×10^{3}[c]
^{85m}Kr	7.2×10^{4}	--
^{87}Kr	1.4×10^{5}	--
^{88}Kr	2.0×10^{5}	--
^{86}Rb	1.3×10^{2}	--
^{89}Sr	9.4×10^{0}	7.3×10^{-2}
^{90}Sr	3.7×10^{-1}	3.6×10^{-1}
^{91}Sr	1.1×10^{1}	--
^{90}Y	--	3.6×10^{-1}[d]
^{91}Y	--	--
^{95}Zr	--	--
^{97}Zr	--	--
^{95}Nb	--	--
^{99}Mo	--	--
^{99m}Tc	--	--
^{103}Ru	--	--
^{105}Ru	--	--
^{106}Ru	--	--
^{105}Rh	--	--
^{127}Te	5.9×10^{1}	--
^{127m}Te	1.1×10^{1}	1.1×10^{0}
^{129}Te	3.1×10^{2}	--
^{129m}Te	5.3×10^{1}	3.1×10^{-2}
^{131m}Te	1.3×10^{2}	--
^{132}Te	1.2×10^{3}	--
^{127}Sb	6.1×10^{1}	--
^{129}Sb	3.3×10^{2}	--
^{131}I	1.4×10^{5}	--

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TABLE 8.2-2. (contd)

Radionuclide	Activity Released From Damaged Fuel During Accident (Ci)	Activity After 1 Year (Ci)
^{132}I	2.0×10^5	--
^{133}I	2.9×10^5	--
^{134}I	3.2×10^5	--
^{135}I	2.6×10^5	--
^{133}Xe	5.1×10^5	--
^{135}Xe	1.0×10^5	--
^{134}Cs	3.8×10^4	2.7×10^4
^{136}Cs	1.5×10^4	--
^{137}Cs	2.4×10^4	2.3×10^4
^{140}Ba	1.6×10^1	--
^{140}La	--	--
^{141}Ce	--	--
^{143}Ce	--	--
^{144}Ce	--	--
^{143}Pr	--	--
^{147}Nd	--	--
^{239}Np	--	--
^{238}Pu	--	--
^{239}Pu	--	--
^{240}Pu	--	--
^{241}Pu	--	--
^{241}Am	--	--
^{242}Cm	--	--
^{244}Cm	--	--
Totals	2.3×10^6	5.2×10^4

(a) Scenario 1 assumes 10% fuel cladding failure and no fuel melting.
(b) Less than 1×10^{-3}Ci.
(c) Released to the ouside atmosphere by controlled venting of the containment building prior to the start of accident cleanup.
(d) Daughter of ^{90}Sr.

TABLE 8.2-3. Estimated Inventory of Radioactivity Released from Damaged Reactor Fuel During Scenario 2 Accident[a]

Radionuclide	Activity Released From Damaged Fuel During Accident (Ci)	Activity After 1 Year (Ci)
^{58}Co	1.2×10^3	3.4×10^1
^{60}Co	4.4×10^2	3.8×10^2
^{85}Kr	3.3×10^4	3.1×10^4 [b]
^{85m}Kr	1.4×10^6	-- [c]
^{87}Kr	2.8×10^6	--
^{88}Kr	4.0×10^6	--
^{86}Rb	1.6×10^3	2.1×10^{-3}
^{89}Sr	4.7×10^5	3.7×10^3
^{90}Sr	1.8×10^4	1.8×10^4
^{91}Sr	5.5×10^5	--
^{90}Y	5.8×10^2	1.8×10^4 [d]
^{91}Y	1.8×10^4	2.5×10^2
^{95}Zr	2.2×10^4	4.6×10^2
^{97}Zr	2.2×10^4	--
^{95}Nb	2.2×10^4	4.6×10^2 [e]
^{99}Mo	2.4×10^5	--
^{99m}Tc	2.1×10^5	--
^{103}Ru	1.6×10^5	2.7×10^2
^{105}Ru	1.1×10^5	--
^{106}Ru	3.8×10^4	1.9×10^4
^{105}Rh	7.4×10^4	--
^{127}Te	4.5×10^4	--
^{127m}Te	8.4×10^3	8.2×10^2
^{129}Te	2.4×10^5	--
^{129m}Te	4.0×10^4	2.4×10^1
^{131m}Te	9.9×10^4	--
^{132}Te	9.1×10^5	--
^{127}Sb	4.6×10^4	--
^{129}Sb	2.5×10^5	--
^{131}I	4.4×10^6	--

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TABLE 8.2-3. (contd)

Radionuclide	Activity Released From Damaged Fuel During Accident (Ci)	Activity After 1 Year (Ci)
^{132}I	6.2×10^6	--
^{133}I	8.8×10^6	--
^{134}I	9.9×10^6	--
^{135}I	7.8×10^6	--
^{133}Xe	1.0×10^7	--
^{135}Xe	2.0×10^6	--
^{134}Cs	4.7×10^5	3.4×10^5
^{136}Cs	1.9×10^5	--
^{137}Cs	3.0×10^5	2.9×10^5
^{140}Ba	8.0×10^5	2.1×10^{-3}
^{140}La	2.4×10^4	2.1×10^{-3}(f)
^{141}Ce	2.2×10^4	8.8×10^0
^{143}Ce	2.0×10^4	--
^{144}Ce	1.3×10^4	5.3×10^3
^{143}Pr	2.0×10^4	--
^{147}Nd	9.0×10^3	--
^{239}Np	2.4×10^5	--
^{238}Pu	8.6×10^0	8.6×10^0
^{239}Pu	3.2×10^0	3.2×10^0
^{240}Pu	3.2×10^0	3.2×10^0
^{241}Pu	5.1×10^2	5.1×10^2
^{241}Am	2.6×10^{-1}	2.6×10^{-1}
^{242}Cm	7.5×10^1	1.6×10^1
^{244}Cm	3.4×10^0	3.4×10^0
Totals	6.3×10^7	7.3×10^5

(a) Scenario 2 assumes 50% fuel cladding failure and 5% fuel melting.
(b) Released to the outside atmosphere by controlled venting of the containment building prior to the start of accident cleanup.
(c) Less than 1×10^{-3} Ci.
(d) Daughter of ^{90}
(e) Daughter of ^{95}Zr.
(f) Daughter of ^{140}Ba.

TABLE 8.2-4. Estimated Inventory of Radioactivity Released from Damaged Reactor Fuel During Scenario 3 Accident[a]

Radionuclide	Activity Released From Damaged Fuel During Accident (Ci)	Activity After 1 Year (Ci)
^{58}Co	1.2×10^4	3.4×10^2
^{60}Co	4.4×10^3	3.8×10^3
^{85}Kr	2.6×10^5	2.4×10^5[b]
^{85m}Kr	1.1×10^7	--[c]
^{87}Kr	2.2×10^7	--
^{88}Kr	3.2×10^7	--
^{86}Rb	1.1×10^4	1.5×10^{-2}
^{89}Sr	4.7×10^6	3.7×10^4
^{90}Sr	1.9×10^5	1.8×10^5
^{91}Sr	5.5×10^6	--
^{90}Y	5.8×10^3	1.8×10^5[d]
^{91}Y	1.8×10^5	2.5×10^3
^{95}Zr	2.3×10^5	4.8×10^3
^{97}Zr	2.3×10^5	--
^{95}Nb	2.3×10^5	4.8×10^3[e]
^{99}Mo	2.4×10^6	--
^{99m}Tc	2.1×10^6	--
^{103}Ru	1.7×10^6	2.9×10^3
^{105}Ru	1.1×10^6	--
^{106}Ru	3.8×10^5	1.9×10^5
^{105}Rh	7.4×10^5	--
^{127}Te	4.4×10^5	--
^{127m}Te	8.2×10^4	8.0×10^3
^{129}Te	2.3×10^6	--
^{129m}Te	4.0×10^5	2.4×10^2
^{131m}Te	9.8×10^5	--
^{132}Te	9.0×10^6	--
^{127}Sb	4.6×10^5	--
^{129}Sb	2.5×10^6	--
^{131}I	3.9×10^7	--

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TABLE 8.2-4. (contd)

Radionuclide	Activity Released From Damaged Fuel During Accident (Ci)	Activity After 1 Year (Ci)
^{132}I	5.5×10^7	--
^{133}I	7.8×10^7	--
^{134}I	8.7×10^7	--
^{135}I	6.9×10^7	--
^{133}Xe	7.9×10^7	--
^{135}Xe	1.6×10^7	--
^{134}Cs	3.2×10^6	2.3×10^6
^{136}Cs	1.3×10^6	4.6×10^{-3}
^{137}Cs	2.0×10^6	2.0×10^6
^{140}Ba	8.0×10^6	2.1×10^{-2}
^{140}La	2.4×10^5	2.1×10^{-2}(f)
^{141}Ce	2.2×10^5	8.7×10^1
^{143}Ce	2.0×10^5	--
^{144}Ce	1.3×10^5	5.3×10^4
^{143}Pr	2.0×10^5	--
^{147}Nd	9.0×10^4	--
^{239}Np	2.4×10^6	--
^{238}Pu	8.6×10^1	8.6×10^1
^{239}Pu	3.2×10^1	3.2×10^1
^{240}Pu	$.2 \times 10^1$	3.2×10^1
^{241}Pu	5.1×10^3	5.1×10^3
^{241}Am	2.6×10^0	2.6×10^0
^{242}Cm	7.5×10^2	1.6×10^2
^{244}Cm	3.5×10^1	3.5×10^1
Totals	5.4×10^8	5.2×10^6

(a) Scenario 3 assumes 100% fuel cladding failure and 50% fuel melting.
(b) Released to the outside atmosphere by controlled venting of the containment building prior to the start of accident cleanup.
(c) Less than 1×10^{-3} Ci.
(d) Daughter of ^{90}Sr.
(e) Daughter of ^{95}Zr.
(f) Daughtert of ^{140}Ba.

Appendix E, are completed. (At TMI-2, the processing of sump water from the containment building basement did not begin until approximately 2-1/2 years after the accident.) During this period, the short-lived radioisotopes, including the radioiodines and all of the noble gases except ^{85}Kr, will decay to insignificant levels of radioactivity. The ^{85}Kr can be vented to the outside atmosphere prior to the entry of cleanup personnel into the containment. The principal fission products of concern during cleanup and decommissioning operations are ^{90}Sr, ^{106}Ru, ^{134}Cs, ^{137}Cs, and ^{144}Ce. Information about the radioactive decay of these isotopes is presented in Appendix C. They all have half-lives that are comparable to the time postulated for planning and preparation that precedes the cleanup of containment.

For reference purposes, both the fission product radioactivity released from damaged fuel at the time of the accident and the fission product radioactivity remaining 1 year after the accident are shown in Tables 8.2-2 through 8.2-4. The 1-year time period is chosen as being representative of the time that might be required for planning and preparation. Because most of the accident cleanup and decommissioning activities that require entry into contaminated areas take place after planning and preparation is completed, the radioactivity remaining after 1 year is used to estimate contamination levels and occupational exposure rates. Most of the radioisotopes that are omitted from consideration based on the 1-year time period have half-lives that are very short compared to 1 year, and most of the isotopes that remain have half-lives that are long compared to 1 year. Therefore, the use of shorter (e.g., 6 months) or longer (e.g., 2 or 3 years) reference time periods would not significantly alter the estimates of fission product radioactivity in the containment building at the start of accident cleanup operations.

8.3 REFERENCE PWR ACCIDENT SCENARIOS

The reference PWR accident scenarios are described in this section. Parameters that characterize these scenarios include:

- fission product contamination levels
- volumes of reactor coolant released (sump water)

- gamma radiation exposure levels
- fuel damage
- damage to the reactor building and equipment.

Postulated values of these parameters for each of the three accident scenarios are given in Table 8.3-1. Because it is the only accident at a large power reactor resulting in significant contamination, corresponding accident parameters for the March 28, 1979 accident at TMI-2[11] are also given in the table to facilitate comparisons with the reference accident scenarios. The contamination levels at TMI-2 tend to fall between the values postulated for the scenario 2 and scenario 3 accidents. As described in other chapters of this report, many of the accident cleanup activities for the scenario 2 and scenario 3 accidents are similar to activities planned for cleanup at TMI-2.

The parameter values given in Table 8.3-1 refer to conditions inside the containment building 1 year after the reference accidents. Contamination of other onsite structures as a result of these accidents is generally not considered in this study. However, for the scenario 2 and scenario 3 accidents, radioactive contamination of the auxiliary and fuel buildings is also postulated. This contamination is assumed to be limited to plateout on building and equipment surfaces, small puddles of contaminated water, and internal contamination of ECCS and chemical and volume control system (CVCS) tanks and piping. The amount of radioactivity in tanks and piping in the auxiliary and fuel buildings is assumed to be 20,000 Ci contained in 200 m^3 of liquid. The radioactivity is predominantly ^{90}Sr, ^{134}Cs, and ^{137}Cs. General area radiation exposure levels inside the auxiliary and fuel buildings following the scenario 2 and scenario 3 accidents are assumed to be about 100 mR/hr. Higher readings of up to 100 R/hr occur in cubicles that contain the filters, demineralizers, and holdup tanks for the CVCS.

Decontamination of the auxiliary and fuel buildings would precede cleanup operations inside the containment, since equipment and services located in these buildings are required for maintenance of the reactor in a safe shutdown condition as well as for subsequent cleanup operations inside the containment building. The requirements and costs of accident cleanup in the auxiliary

TABLE 8.3-1. Comparison of TMI-2 Accident Parameters with Reference PWR Accident Parameters

Parameter	Parameter Values[a]			
		Reference PWR Accidents		
	TMI-2[b]	Scenario No. 1	Scenario No. 2	Scenario No. 3
Percent of fuel cladding failure	~50	10	50	100
Percent of fuel melting	0	0	5	50
Volume of sump water (m^3)	2650	200	1000	1600
Depth of sump water (m)	2.4	0.2	1.0	1.6
Total fission product radioactivity in sump water (Ci)[c]	5.3×10^5	2.5×10^4	3.5×10^5	2.5×10^6
Average fission product radioactivity in sump water (Ci/m^3)	200	125	350	1560
Total fission product radioactivity plated out on building surfaces (Ci)[d]	>100	5	70	500
Average fission product radioactivity on building surfaces (Ci/m^2)				
- Floors	0.001-0.100[e]	0.001	0.014	0.1
- Walls		0.00001	0.00014	0.001
Average gamma radiation exposure rate at operating floor level (R/hr)				
- Contribution from plateout		0.01	0.15	1.0
- Contribution from sump water		0.015	0.045	0.20
- Total exposure rate	0.25	0.025	0.20	1.2
Average gamma radiation exposure rate at lowest entry level (R/hr)				
- Contribution from plateout		0.01	0.15	1.0
- Contribution from sump water		8	30	170
- Total exposure rate	0.50[f]	8	30	170
Damage to fuel core	Oxidation of fuel cladding. Melting and fusing together of stainless steel fittings. Cracking and crumbling of some fuel pellets. Probably no fuel melting.[g]	Slight damage to some fuel elements as a result of fuel swelling and cladding rupture.	Oxidation of fuel cladding. Melting and fusing together of stainless steel fittings. Cracking and crumbling of some fuel pellets. Melting of fuel in localized areas of central core.	Cracking, crumbling, and melting of fuel pellets. Melting and fusing together of stainless steel parts on adjacent fuel assemblies. Molten fuel present over much of core radius. Fuel and cladding fragments carried throughout primary coolant system.
Damage to containment building and equipment	Contamination of building ventilation system. Most electrical equipment and some valves inoperable due to water damage and corrosion. Polar crane inoperable. Minor structural damage.	No significant physical damage.	Contamination of building ventilation system. Some electrical equipment and some valves inoperable due to water damage and corrosion. Minor structural damage. Polar crane inoperable.	Ventilation ductwork damaged. Doors, catwalks, pipes, and cable conduits dented or ripped away. Loss of electrical and other services. Erosion of concrete and metal surfaces. Polar crane inoperable.

(a) Values refer to conditions inside the containment building approximately 1 year after the postulated accident.
(b) Summarized from data in Reference 10 unless otherwise noted.
(c) "Sump water" is accident water present in the containment building basement.
(d) Plateout values are after washdown of the walls by condensing moisture.
(e) Limited data exists on plateout at TMI-2.
(f) In the reference PWR, the lowest entry level is 2.7 m above the basement level. At TMI-2, this level is 7.5 m above the basement level. At TMI-2, gamma radiation measurements of 40-45 R/hr were obtained at distances of about 2 m above the top of the sump water.
(g) Condition of fuel core uncertain until reactor vessel head is removed and core is visually inspected.

and fuel buildings following the scenario 2 and scenario 3 accidents are described in Appendices E and F, along with discussions of the requirements and costs of accident cleanup in the containment building.

The models and assumptions used to estimate contamination levels and gamma radiation exposure rates are described in the following subsections. Additional details are presented in Appendix C. Appendix C also contains a discussion of radioactive contamination and gamma exposure level measurements at TMI-2, for comparison with the values of these parameters for the reference scenarios of this study.

8.3.1 Fission Product Contamination

Estimates of the fission product radioactivity plated out on containment building surfaces and equipment or contained in the sump water at time t = 1 year after the postulated accidents are shown in Table 8.3-1. Total plate-out on containment building surfaces and equipment is estimated to range from about 5 Ci for the scenario 1 accident to about 500 Ci for the scenario 3 accident. Plateout values listed in the table assume washdown of the walls by condensing moisture. Total fission product radioactivity in the sump water is estimated to range from about 25,000 Ci for the scenario 1 accident to about 2.5 million Ci for the scenario 3 accident. These values compare with an estimated 100 Ci of plateout and 500,000 Ci of fission product radioactivity in sump water at TMI-2.[11]

Estimates of plateout and of sump water radioactivity are derived from the radionuclide inventories of Tables 8.2.-2 through 8.2-4 by making the assumptions listed below. The rationale for these assumptions is presented in the following paragraphs.

1. Approximately 50% of the non-gaseous fission product radioactivity released from the fuel core is retained within the reactor pressure vessel and the coolant system piping. The remaining 50% is released to the reactor building atmosphere.
2. Approximately 0.1% of the non-gaseous fission product radioactivity released to the reactor building atmosphere plates out on building

and equipment surfaces. The remainder of the non-gaseous radioactivity released to the building atmosphere concentrates in the sump water.

3. The radioactivity plated out on building and equipment surfaces is initially distributed uniformly over a total surface area of 25,000 m^2. However, flushing of the walls that occurs during and after the accident reduces the contamination per unit area on the walls to approximately 1% of the value on the floors and other horizontal surfaces.

Assumptions 2 and 3 are consistent with initial observations of radioactive contamination inside the TMI-2 containment building.

During a reactor accident that includes fuel cladding failure and fuel melting there will be many cooler regions in the primary coolant system (e.g., in the reactor vessel and coolant system piping) that offer suitable plateout surfaces for released non-gaseous fission products. Experiments reported by Levenson and Rahn(12) indicate that cesium plates out on such surfaces when their temperatures are in the range of 500 to 1000°C. No quantitative data are documented that could provide a basis for estimating the amount of fission product radioactivity released from the fuel core that is retained as plateout in the pressure vessel or the primary system piping. The NRC study of fission product behavior during an LWR accident(8) points out that retention of non-gaseous fission products within the reactor coolant system (RCS) can be substantial for certain accident sequences such as small-break LOCAs. Retention in the RCS would likely be less for a large-break LOCA that resulted in the rapid removal of material from the system. In the absence of specific numerical data, in this study it is assumed for all three accident scenarios that 50% of the non-gaseous radioactivity released from damaged fuel is retained in the RCS and the remaining 50% escapes to the containment building atmosphere.

Fission product radionuclides that are dissolved or suspended in the primary coolant may be released to the reactor building atmosphere with the water vapor and steam released during the accident. The distribution and retention of fission products in the containment building are a function of the course of the accident, of propagation and precipitation mechanisms, and of the effect of safety equipment such as building sprinklers and ventilation system filters. Adequate means are not presently available for the theoretical determination of the complex processes that produce the final distribution of fission product radioactivity in the containment building following an accident. Therefore, only very general descriptions of the fission product contamination and radiation dose rates are attempted. For the purposes of this study, worker doses can be estimated from the average dose rates thus defined.

Radionuclides that are originally present as particles or aerosols in the discharge may experience one or more of the following processes:

- they may be released to the environment through leaks in the containment
- they may be collected in filters
- they may fall out of the building atmosphere by the action of gravity
- they may be washed out of the building atmosphere by operation of the building sprinkling system
- they may be precipitated or condensed on building and equipment surfaces
- they may be washed into the sump by gravity flow of moisture that condenses on building and equipment surfaces.

The amount of radioactivity that plates out on building or equipment surfaces is expected to be a small fraction of the total radioactivity present in the building atmosphere. For this study, 0.1% of the non-gaseous fission product radioactivity present in the building atmosphere is assumed to plate out on exposed surfaces. This is consistent with measurements made during pre-cleanup entries into the containment at TMI-2.[11] The plateout is assumed to be initially uniformly distributed over a total surface area of approximately 25,000 m^2.

Results of a limited number of wipe tests made at TMI-2 indicate that levels of removable contamination on the floors of the containment building are much higher than contamination levels on the walls of the building. This is presumably due to washdown of the walls by moisture that condenses on the walls and drains down onto the floors and into the basement. In this study, it is assumed that washdown reduces the level of radioactive contamination on walls to approximately 1% of its original plateout value. This is also consistent with TMI results.[11]

8.3.2 Gamma Radiation Exposure Levels

Estimated average gamma radiation exposure rates from fission product contamination at the operating floor level (28.35 m) and at the emergency airlock floor level (16.6 m) of the reference PWR 1 year after the postulated accidents are shown in Table 8.3-1. Estimated average exposure rates at the operating floor level range from 25 mR/hr for the scenario 1 accident to 1.2 R/hr for the scenario 3 accident. Estimated average exposure rates at the airlock floor level range from 8 R/hr for the scenario 1 accident to 170 R/hr for the scenario 3 accident.

The total gamma radiation exposure rate in any given area includes contributions from surface contamination and from sump water contamination. At the emergency airlock floor level, the gamma exposure results almost entirely from the sump water contamination. At the operating floor level, the contribution from surface contamination predominates.

Exposure rate calculations employ the methodology developed in the Reactor Shielding Design Manual.[13] Details of these calculations are given in Appendix C. A simplified model of the PWR containment building, used to define the geometrical parameters in the equations for exposure rate calculations, is shown in Figure 8.3-1. Points P_1 and P_2 represent the points at the operating floor level and the emergency airlock floor level where exposure rates are calculated. The operating floor level is approximately 14.5 m above the containment building basement. Shielding from the sump water radioactivity provided by structures and equipment at this level is assumed to be equivalent to a 0.45-m-thick concrete floor. The emergency airlock level is approximately 2.7 m above the containment building basement. The floor at this level is a steel grating.

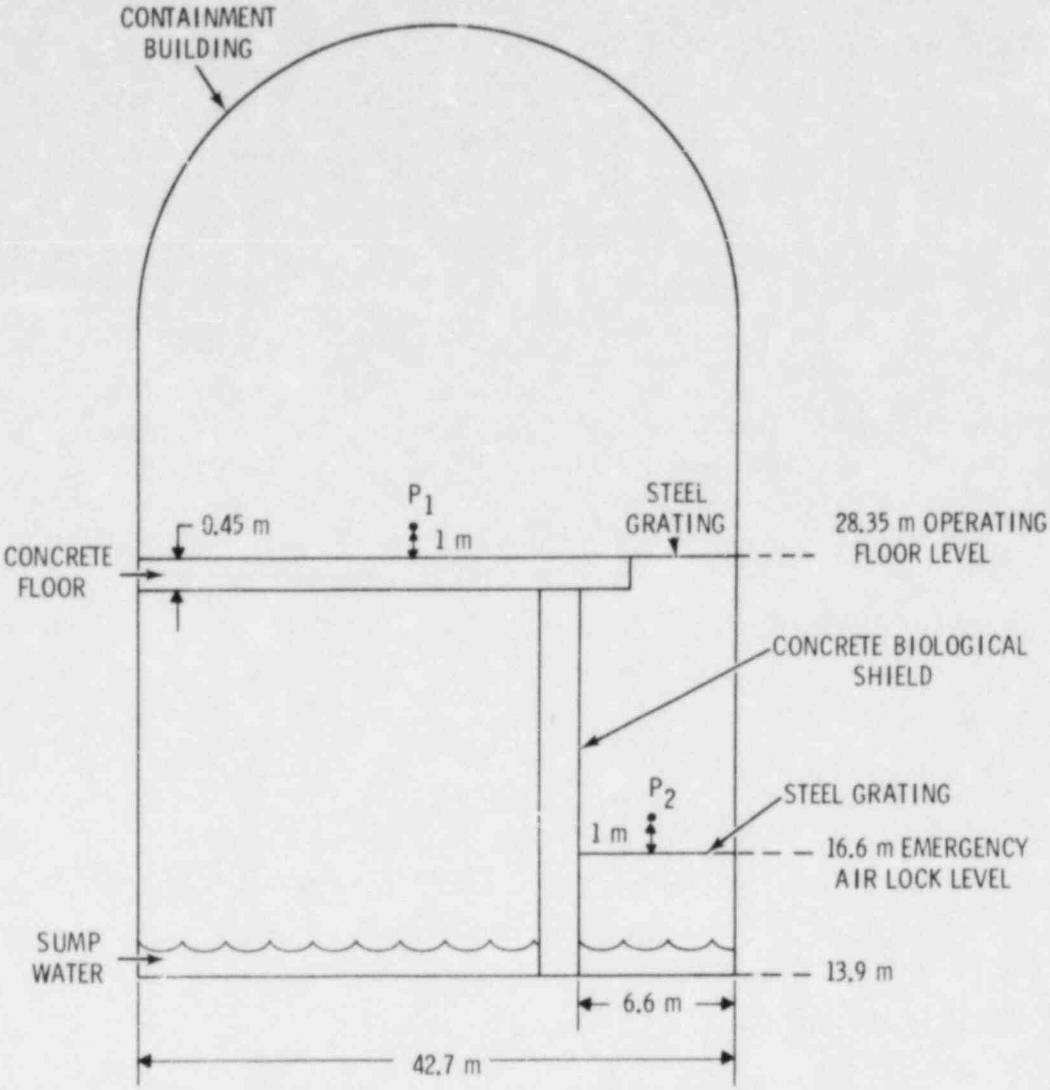

FIGURE 8.3-1. Simplified Model of PWR Containment

To calculate the exposure rate from surface contamination, the postulated contamination on floors and walls is approximated by a uniformly contaminated infinite plane. Results of a sensitivity analysis made for the BWR decommissioning study[(14)] indicate that the calculated gamma exposure rate at 1 m from a uniformly contaminated infinite plane closely approximates calculated exposure rates at 1 m from the floors of rooms with floor areas greater than about 30 m^2, when wall surface contamination levels are much lower than floor contamination levels. This analysis is reproduced in Appendix C. For calculational purposes, the contamination is assumed to be entirely from ^{137}Cs. As shown in Tables 8.2-2 through 8.2-4, the cesium isotopes, ^{134}Cs and ^{137}Cs, are the major contributors to fission product radioactivity 1 year after the postulated accidents. The principal gamma rays from the decay of ^{134}Cs have energies of 0.605 MeV and 0.796 MeV. The decay of ^{137}Cs results in the emission of a 0.662-MeV gamma ray. (See Appendix C for details of the decay schemes of ^{137}Cs and ^{137}Cs.) For ease of calculation, gamma radiation exposure rates are estimated on the basis that each decay of a fission product nucleus results in the emission of a 0.662-MeV gamma ray.

To calculate the gamma radiation exposure rate from the contaminated sump water, the sump water is modeled as a uniformly contaminated disk of finite thickness. At the operating floor level, the disk is assumed to have a diameter of 42.7 m, and 0.45 m of concrete shielding is assumed between the disk and the point P_1. At the emergency airlock floor level, the contaminated disk is assumed to have a diameter of 6.6 m and no shielding is assumed between the disk and the point P_2. The smaller diameter disk is used to model the sump water source at the point P_2 because of the presence of the concrete biological shield surrounding the steam generators. Since the emergency airlock floor is a steel grating, no credit is taken for shielding from the source. Assumed disk thicknesses and radioactivity per unit volume are shown in Table 8.3-1 for the three accident scenarios.

The estimated exposure rates from fission product contamination shown in Table 8.3-1 are average values that are used as a basis for estimating occupational radiation doses to workers engaged in cleanup and decommissioning operations in the reference PWR. Actual exposure rates would vary depending on the location of the worker in the containment building. Localized hot spots (such as floor drains) would result in much higher exposures. At the operating floor level, the exposure rates near stairwells would be much higher than average (possibly as high as 1 to 10 R/hr) because of the absence of concrete shielding for protection from gamma radiation from the contaminated sump water.

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CHAPTER 9

ACCEPTABLE RESIDUAL RADIOACTIVE CONTAMINATION LEVELS FOR A DECOMMISSIONED LWR

Unrestricted release of a nuclear reactor facility following decommissioning requires that the radioactivity remaining in the facility and on the site be reduced to levels which are considered acceptable for unrestricted access and subsequent NRC license termination. This chapter provides a summary of the criteria being developed for determining acceptable residual contamination levels for public use of a decommissioned LWR.

Some guidance currently exists defining the levels of radioactive surface contamination that are acceptable to the NRC for the termination of operating licenses.[1,2] This guidance by itself is not explicit enough in relating contamination levels to annual dose rates or justifying that the values presented are ALARA, and does not sufficiently accommodate the various radionuclide mixtures or site-specific features found at each unique nuclear facility. This has resulted in the need for case-by-case licensing decisions to establish acceptable residual levels. These facts suggest that the methodology used to calculate acceptable levels of residual radionuclide contamination at decommissioned nuclear facilities should be based on a more general concept that is capable of accommodating unique radionuclide mixtures and site-specific features.

The draft generic environmental impact statement (GEIS) on decommissioning of nuclear facilities[3] contains a recommendation that the allowable residual radioactivity levels for facility release be based on the dose anticipated to be received by individuals who use the facility or site after the license is terminated. The GEIS includes a discussion of several basic requirements that must be considered in selecting an acceptable level for facility release. These requirements include:

1) The level must be low enough to comply with the ALARA concept.

2) The level must be verifiable through actual detailed radiological survey measurements of the facility and site.

3) Dose rates and associated contamination levels should be based on realistic dose assessment methodology.

Guidance on establishing residual radioactivity levels, on the costs and methodology of radiological surveys, and on realistic dose methodology is contained in the GEIS[3] and in two reports prepared by Oak Ridge National Laboratory.[4,5]

It is not within the scope of this study to recommend annual dose limits for the exposure of the public to radioactive materials. Under the provisions of the Energy Reorganization Act of 1974, the Environmental Protection Agency (EPA) has responsibility for establishing radiation dose standards for the protection of the public health and safety and for establishing radiation dose limits for decommissioning a nuclear facility to unrestricted access. The EPA has not yet promulgated these limits and is not scheduled to begin this activity until 1984 with standards being set at some later date. In the absence of standards in this area, the NRC, based on the statutory authority contained in the Atomic Energy Act to protect the health and safety of the public, has issued the guidance in the GEIS and announced its intention to promulgate regulations establishing criteria for acceptable residual radioactivity levels for the release of decommissioned facilities and sites.[6]

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CHAPTER 10

ACTIVITIES AND MANPOWER REQUIREMENTS FOR ACCIDENT CLEANUP AT A REFERENCE PWR

The first activities following reactor accidents that result in severe radioactive contamination and possible physical damage of the containment building are designed to bring the accident under control and to stabilize the facility to prevent further releases of radioactivity. Initial stabilization is followed by accident cleanup, which is followed by either refurbishment or decommissioning of the reactor facility. This chapter provides information about the activities and manpower requirements for accident cleanup. Activities and manpower requirements for decommissioning following accident cleanup are discussed in Chapter 12. This study does not analyze the detailed activities related to refurbishment of the facility.

The information presented in this chapter is a summary of the detailed discussion of accident cleanup presented in Appendix E of Volume 2. General information on decontamination, liquid waste treatment, and packaging and disposing of wastes is given in Appendix D of Volume 2. The purpose of accident cleanup is discussed in Section 10.1. Preparations for accident cleanup are discussed in Section 10.2. Accident cleanup in the fuel and auxiliary buildings, postulated to be required for cleanup following the scenario 3 accident, is discussed in Section 10.3. Accident cleanup in the containment building is discussed in Section 10.4. The costs of accident cleanup are summarized in Chapter 11.

10.1 PURPOSE OF ACCIDENT CLEANUP

The reactor accidents postulated in this report (see Chapter 8) result in severe contamination of the containment building, damage to the fuel core, and the accumulation of contaminated water on floors and in building sumps. Either decommissioning or refurbishment of a nuclear power reactor following these postulated accidents would begin with an accident cleanup campaign with three principal goals:

1) to reduce the initial high levels of radioactive contamination present on building surfaces and in water released during the accident, thereby reducing the radiation dose received by workers engaged in cleanup and subsequent decommissioning or refurbishment operations

2) to safely defuel the reactor, placing the fuel in a configuration that is safe from nuclear criticality and/or fuel meltdown

3) to collect, package for disposal, and dispose of the large quantities of water-soluble and otherwise readily dispersible radioactivity present in the plant.

To achieve these goals, the accident cleanup campaign is postulated to include the following tasks:

- processing of the contaminated water generated by the accident (and by decontamination operations) to remove and immobilize radioactive contaminants
- initial decontamination of building surfaces and decontamination or disposal of some equipment
- removal of spent fuel (damaged and undamaged) from the reactor vessel and storage of the fuel in the spent fuel pool
- cleanup of the reactor coolant system
- solidification and packaging of wastes from accident cleanup operations.

The rationale for accident cleanup is discussed in detail in Section E.1 of Appendix E. The processing of accident water and the defueling of the reactor are considered major steps in the cleanup process following a reactor accident and must be performed regardless of whether the reactor is to be restarted or decommissioned. Accident cleanup activities must establish conditions within the containment building that permit reasonable occupancy times without excessive radiation exposure to workers engaged in defueling operations. Thus, accident cleanup activities include, in addition to the processing of accident water, the venting of any radioactive gases present in the containment building and some decontamination of containment building surfaces to reduce the levels of radiation exposure in selected work areas. In keeping with ALARA principles, decontamination during accident cleanup is restricted to operations that result in the greatest reduction in residual radioactive contamination with the least radiation dose to decommissioning workers.

As a result of fuel damage during an accident, many small fuel fragments can be carried out to portions of the reactor coolant system that are external to the reactor vessel. In this report, the decontamination of reactor coolant system components following an accident in which fuel damage has occurred is considered part of the accident cleanup activities. Decontamination includes draining and processing the contaminated water; removing fuel debris from the system by flushing, chemical treatment, or other means; and flushing with appropriate solutions to remove most of the remaining contamination adhering to inside surfaces.

The tasks included in accident cleanup are necessary and the procedures used to accomplish these tasks are assumed to be essentially independent of whether the facility is ultimately restarted or decommissioned, and if decommissioned, of the alternative (DECON, SAFSTOR, or ENTOMB) chosen. (The rationale for this assumption is discussed in Section E.1 of Appendix E.) Because accident cleanup activities are similar, the requirements and tasks presented in this section are considered to be a good representation of the activities carried out during accident cleanup independent of the ultimate use of the plant. The work required to accomplish each cleanup task is, of course, affected by the severity of the accident. Technical requirements for accident cleanup are discussed in the following sections.

As defined in this study, accident cleanup does not include the extensive hands-on decontamination operations required to reduce surface contamination inside the plant to levels suitable for release of the facility for unrestricted use. That additional decontamination would take place during the decommissioning activities. In addition, as defined in this study, accident cleanup does not include the additional hands-on decontamination to reduce radiation levels to low enough values to permit the extensive work necessary during refurbishment of the facility. Accident cleanup also does not include the decontamination or disposal of large, permanently installed equipment items, such as the reactor vessel, reactor coolant pumps, steam generators, pressurizer, regenerative heat exchangers, and associated piping, that would not be removed or refurbished until the decommissioning or refurbishment of the facility after completion of the accident cleanup activities.

10.2 PREPARATIONS FOR ACCIDENT CLEANUP

Since accidents are unplanned events, preparations for accident cleanup and for the subsequent decommissioning of the reactor must begin after the accident has occurred and the plant is shut down. Several planning and preparation tasks, needed to ensure the proper performance and the success of accident cleanup, must be completed before cleanup operations begin. Planning and preparation activities and the manpower requirements for their performance are described in this section.

10.2.1 Planning and Preparation Activities

Planning and preparation activities that must be performed prior to the operational phase of containment building cleanup include (not necessarily or exactly in this order):

- venting of radioactive gases (e.g., krypton-85)
- containment entry and data acquisition
- preparation of documentation for regulatory agencies
- design, fabrication, and installation of special equipment
- development of detailed work plans and procedures
- selection and training of accident cleanup staff
- removal of spent fuel stored in the spent-fuel storage pool from prior reactor refuelings.

10.2.1.1 Venting of Radioactive Gases

Significant quantities of radioactive fission products and particulates are released to the containment building atmosphere as a result of the reactor accidents postulated in this study. The fission products include noble gases, iodine, and volatile and semivolatile radionuclides such as ^{137}Cs and ^{90}Sr. Most of the fission product noble gases have short half-lives (e.g., ^{133}Xe with a 5.3-day half-life) and decay to insignificant levels prior to the start of building decontamination. The iodine isotopes which are generally of concern also have short half-lives. (The principal iodine isotope of concern is ^{131}I with an 8-day half-life.) The major contributors to radiation exposures inside the containment building at times greater than 1 year following the accident are the relatively long-lived cesium and strontium isotopes, which plate out on building surfaces or are retained in the accident water, and the noble gas ^{85}Kr, which has a 10.7-year half-life. The ^{85}Kr must be removed from the containment building atmosphere before workers enter the building for data gathering and accident cleanup operations.

Several alternatives are available for krypton removal from the containment building atmosphere. These alternatives include:

- controlled venting (purging)
- selective absorption
- charcoal adsorption
- gas compression and storage
- cryogenic processing.

A discussion of these alternatives is given in Section E.2.1.2 of Appendix E. An environmental assessment evaluating the alternatives for ^{85}Kr removal from the TMI-2 reactor building(1) concluded that the potential health impact on the public from using any of these alternatives, including purging, is negligible.

Based on considerations described in detail in Section E.2.1.2 concerning the safety of alternatives for krypton removal, in this study it is postulated that ^{85}Kr is removed from the containment building by controlled venting of the building atmosphere. This involves the controlled release of air from inside the building by way of filtering and monitoring equipment that leads to the building ventilation stack. A description of the containment building ventilation system is given in Section B.1.5. Costs of other alternatives are included for completeness in the discussion of cost sensitivity in Section 11.6.

10.2.1.2 Containment Entry and Data Acquisition

Data on the post-accident radiological and physical condition of the plant are obtained and analyzed during the planning and preparations phase. These data include measurements of contamination levels and radiation exposure rates inside the containment, estimates of physical damage to structures and equipment, and information about the operational status of plant systems and services. The data provide a basis for planning accident cleanup operations and also provide some of the information needed to prepare documentation for regulatory agencies. Some information about the status of the fuel core can be obtained from an analysis of the accident and from measurements of containment building contamination. However, detailed information about the status of the damaged core may be available only after initial decontamination of the containment building is completed and the reactor vessel head is removed to permit inspection of the core.

Data on post-accident conditions inside the containment building are obtained by teams of workers (generally 2 to 20 persons) who spend short periods (generally a few minutes to a few hours) inside the building. The numbers and durations of entries into containment and the information obtainable during these entries depends on radiation exposure levels inside the building.

10.2.1.3 Documentation for Regulatory Agencies

A major planning task is the preparation by the licensee of the necessary documentation to amend the facility operating license to maintain the reactor in a safe shutdown condition and to obtain regulatory approvals to proceed with cleanup operations. A discussion of existing regulations, guides, and standards that apply to nuclear power reactor accident cleanup and decommissioning is given in Chapter 5. Documentation that must be provided by the licensee prior to the start of cleanup operations is summarized in Section E.2.1.3 of Appendix E. The time needed to furnish information to regulatory agencies, issue environmental statements and assessments, and secure regulatory approvals to proceed with specific cleanup tasks is a critical factor in determining the length of the planning and preparations period and hence could delay the start of actual cleanup operations.

10.2.1.4 Design, Fabrication, and Installation of Special Equipment

Major facilities and equipment items required for accident cleanup include:

- a filter/demineralizer system for processing contaminated water. A new filter/demineralizer system is necessary because the existing radwaste system cannot handle the larger volumes and higher activity of the accident-generated water.
- processed-water storage tanks and associated piping and controls. This additional tankage is necessary because the existing tankage cannot handle the large volumes of accident-generated water.
- special tools for the removal and handling of damaged fuel elements. These tools are necessary because the existing grappling devices for normal defueling may not be able to remove the damaged fuel.
- a mockup of a section of the reactor vessel for use in testing fuel removal equipment and in training personnel to use this equipment.
- stainless steel canisters for overpacking damaged fuel assemblies and modified fuel storage racks designed to accommodate the canistered fuel.
- an evaporator/solidification facility to process the decontamination solutions generated. The existing radwaste system cannot handle the large volumes of accident-generated wastes to be processed.

- a volume reduction incinerator to reduce the total quantities of waste that would need to be disposed of.
- shielded and unshielded storage facilities for interim storage of radioactive wastes. This is necessary because the existing building storage space for processed and solidified radwaste is not large enough for the wastes that will be generated during the accident cleanup. This is especially true if there is difficulty in disposing of wastes because of regulatory or political constraints.
- a laundry facility.

Some items such as evaporators, incinerators, and laundry facilities are commercially available. Other items must be designed and fabricated during preparations for cleanup.

This study assumes that for the reference facility, except for the storage facilities indicated, sufficient space is available for the addition of necessary special equipment for accident cleanup. Because of the potential that at other facilities sufficient space may not be available and hence it might be necessary to construct other new buildings, Section 11.6 includes an analysis of the sensitivity of the costs of accident cleanup to the need to construct a new building to house certain special equipment items.

Designs and specifications are prepared for each special equipment item required. When the item is procured, it is inspected to verify that it meets specifications and complies with applicable quality assurance and safety requirements. It is then tested to ensure that it performs as required. The testing also serves to train personnel in the use of the equipment and to provide pertinent data on its operation. The time requirement for purchasing, installing, and testing this special equipment is a critical factor in determining when actual cleanup activities can begin. Delays in any step in making the equipment ready could cause delays in accident cleanup.

10.2.1.5 Development of Work Plans and Procedures

Detailed work plans and procedures are developed based on an evaluation of the condition of the plant following an accident and on the requirements for accident cleanup. Work plans are included in documentation provided to the NRC with the request for license amendment. The detailed plans and procedures contain all the information required to actually carry out the accident cleanup tasks. They address the following items:

- regulatory requirements and constraints
- decontamination methods and procedures
- schedules and sequences of events
- manpower requirements
- equipment requirements
- contamination control
- radiological and industrial safety
- packaging and disposal of radioactive wastes
- quality assurance.

Physical security and environmental constraints are also considered. Plans are updated as the accident cleanup work proceeds and additional data on the physical and radiological status of the facility becomes available.

10.2.1.6 Selection and Training of Accident Cleanup Staff

The selection and training of operations staff for accident cleanup begins during preparations for cleanup. Because detailed knowledge of and familiarity with the facility being decontaminated increase the effectiveness of cleanup workers, key staff positions are filled, whenever possible, with utility personnel familiar with the construction and operation of the plant. Additional training in specific cleanup tasks is provided, with emphasis on the use of new or unique equipment and procedures.

Training of cleanup staff continues throughout the accident cleanup period. Because of the high exposure rates encountered and the need to limit individual radiation doses, large numbers of persons are involved in accident cleanup operations. Many of these persons are unfamiliar with the plant, and some are unfamiliar with the basic principles of radiation protection. These persons require an orientation in the layout of the plant and in basic radiation protection procedures as well as specific instruction in the tasks to be performed.

10.2.1.7 Removal of Accumulated Spent Fuel

It is postulated in this study that the filter/demineralizer system used to process contaminated water is installed in the spent fuel pool. Reasons for making this assumption are that the pool provides shielding and cooling for this equipment and that use of the pool avoids the necessity of constructing a new building to house the equipment. Hence, because space in the pool is needed for the filter/demineralizer system and for temporary storage of fuel from reactor defueling that takes place during accident cleanup, it is necessary to remove the spent fuel already stored in the pool from prior plant refuelings. (One and one-third fuel cores are assumed to be stored in the spent fuel pool at the time of the postulated accidents.) This fuel is assumed to be transported to an independent spent fuel storage installation (ISFSI) for interim storage. Shipment and storage costs for this fuel are assumed to be charged to reactor operations since the fuel would eventually be shipped offsite if the reactor continued to operate. Because of potential problems that could prevent transport of this fuel, Section 11.6 includes an analysis of the sensitivity of the costs of accident cleanup to the potential need for construction of a new building to house the filter/demineralizer system.

10.2.2 Time Requirements for Preparations for Accident Cleanup

Time requirements for planning and preparation depend on several factors including accident severity, time needed to design, fabricate and test special equipment and facilities, and time required to secure regulatory approvals to proceed with specific cleanup tasks such as the venting of ^{85}Kr, the processing of accident water, or defueling the reactor. The time required to secure regulatory approvals for specific cleanup operations is a critical factor in determining when these operations can begin.

At TMI-2 a period of 2-1/2 years elapsed between the accident and the start of accident water processing operations in the containment building. (However, some accident cleanup in the auxiliary and fuel handling building did the take place during this period.) To provide a basis for computing staff labor costs, preparations for accident cleanup are assumed to require 1.5 years following the scenario 1 accident, 2.5 years following the scenario 2 accident, and 3 years following the scenario 3 accident. To enable the user of this study to compute planning and preparations costs for time periods other than those assumed in this study, utility staff labor costs for preparations for accident cleanup are given on an annual basis in Table F.1-2 of Appendix F.

10.2.3 Staff Requirements for Preparations for Accident Cleanup

A postulated staff organization for preparations for accident cleanup is shown in Figure 10.2-1. This staff includes a cleanup planning branch, a plant operations branch, and several site support branches.

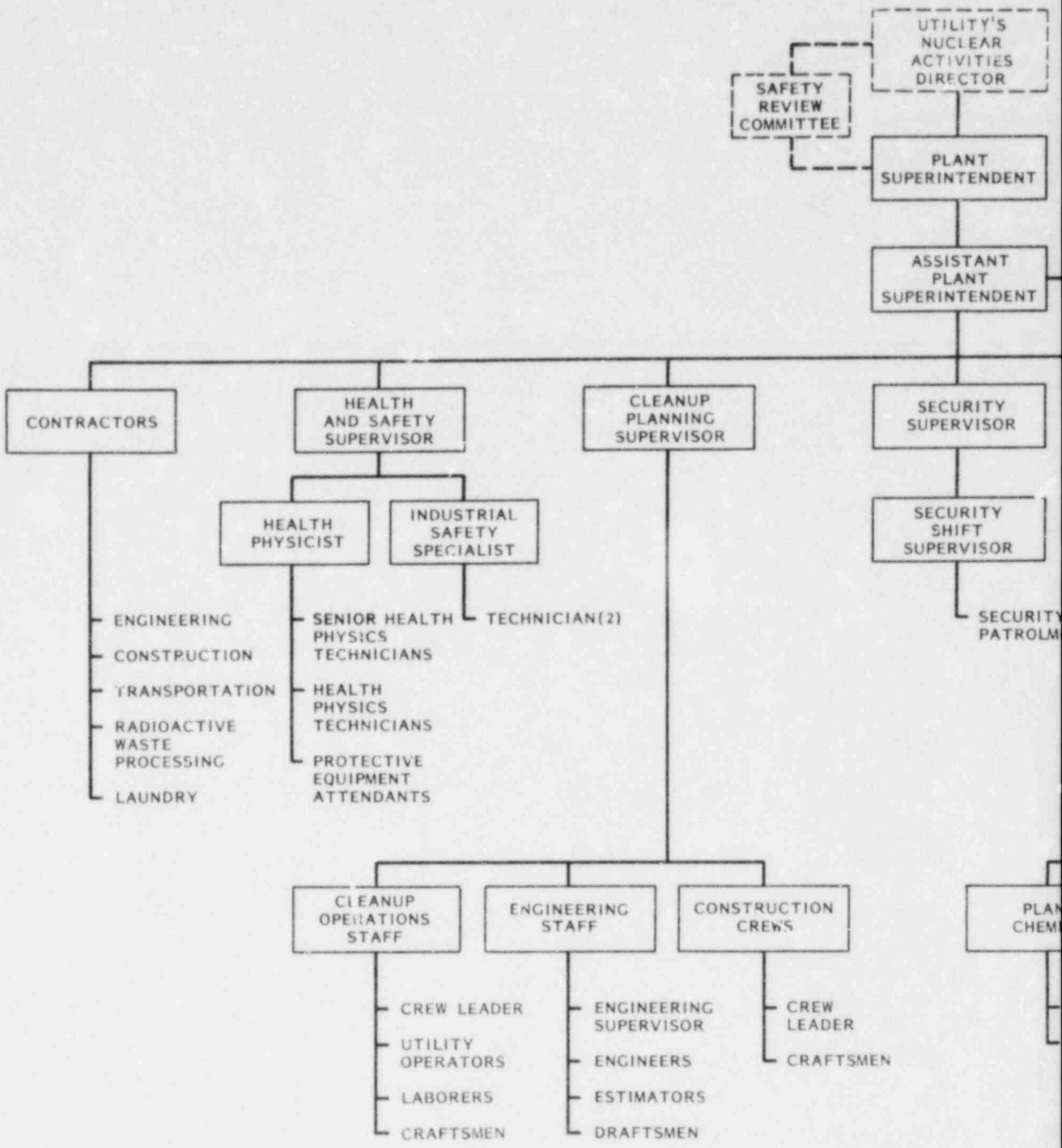
Major activities of the cleanup planning branch include:

- preparation of documentation for regulatory agencies
- preparation of design specifications for special facilities and equipment
- preparation of detailed work plans and work schedules
- venting of radioactive gases present in the containment building following the accident
- acquisition of data on the radiological and physical condition of the plant
- testing of equipment and procedures to be used in cleanup operations
- installation or repair of systems required for accident cleanup (e.g., reroute piping connections, install systems for remote monitoring, etc.).

The plant operations staff has the responsibility to maintain the reactor in a safe shutdown condition. Plant operations staff members also assist the cleanup planning staff in the acquisition of data on the radiological and physical condition of the containment building and in the installation and testing of equipment and systems required for accident cleanup.

Site support includes radiological health, industrial safety, plant security, procurement and accounting, and quality assurance services.

Estimated utility staff labor requirements for preparations for accident cleanup are shown in Table 10.2-1. These labor requirements include 368 man-years for preparations for accident cleanup following the scenario 1



NOTES:

1) NUMBERS IN PARENTHESES REFER TO THE NUMBER OF PERSONNEL PER SHIFT. WHEN THE NUMBER OF PERSONS IS NOT INDICATED, IT IS EITHER UNITY OR VARIES WITH ACCIDENT SCENARIO. TOTAL PERSONNEL REQUIREMENTS FOR EACH ACCIDENT SCENARIO ARE GIVEN IN TABLE E.2-2.

2) THE SECURITY FORCE, THE HEALTH PHYSICS STAFF, THE REACTOR OPERATIONS STAFF, AND THE PLANT MAINTENANCE STAFF FUNCTION ON A 24-HOUR-PER-DAY, 7-DAY-WEEK BASIS.

FIGURE 10.2-1. Postulated Staff Organization for

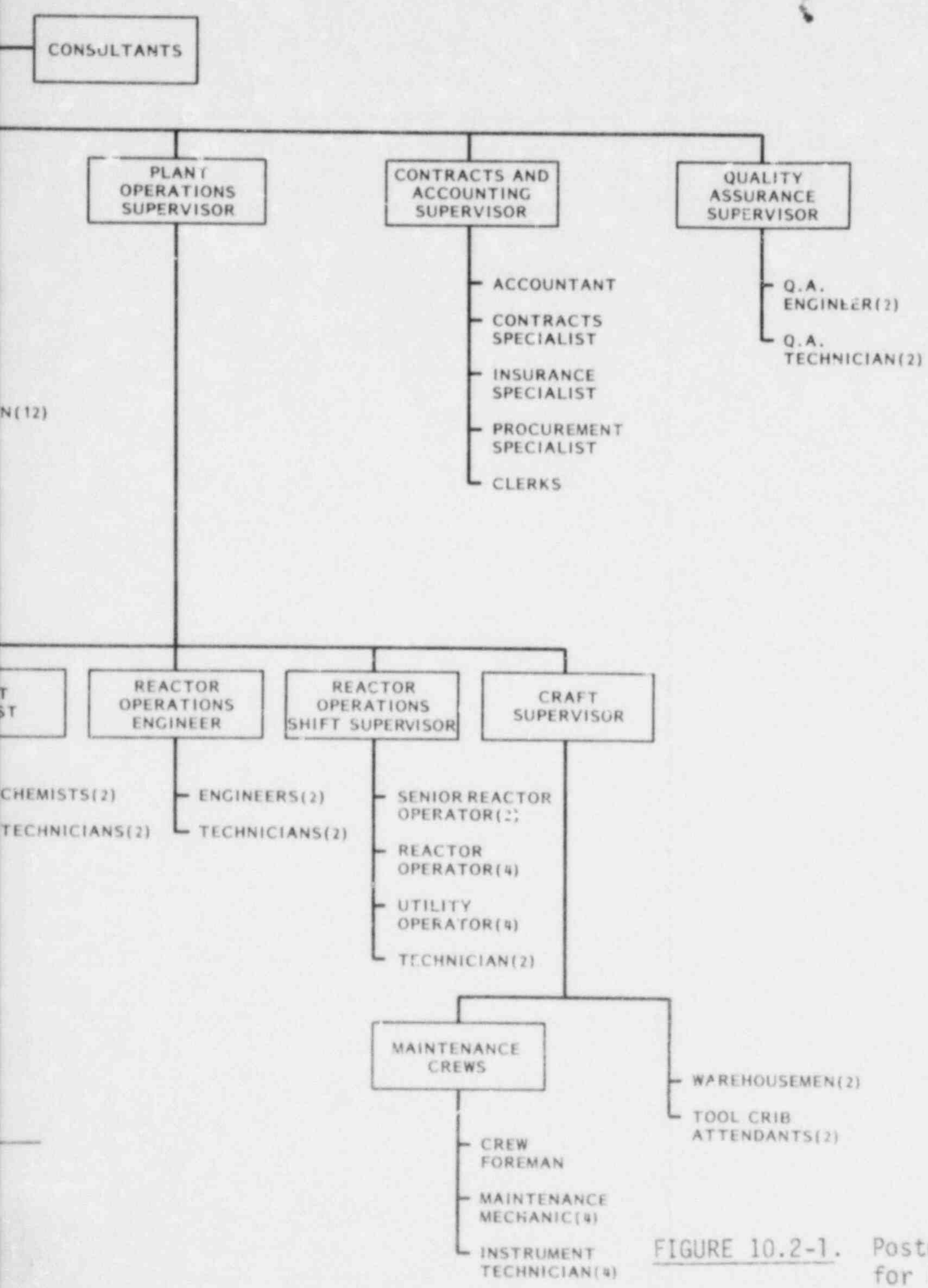


FIGURE 10.2-1. Postulated Staff Organization for Preparations for Accident Cleanup

reparations for Accident Cleanup

TABLE 10.2-1. Estimated Utility Staff Labor Requirements for Preparations for Accident Cleanup

Position	Utility Staff Labor Requirements (man-years) for Preparations for Accident Cleanup		
	Scenario 1 Accident[(a)]	Scenario 2 Accident[(b)]	Scenario 3 Accident[(c)]
Plant Superintendent	1.5	2.5	3.0
Assistant Plant Superintendent	1.5	2.5	3.0
Consultants	4.5	15.0	30.0
Secretaries and Word Processors	12.0	25.0	36.0
Site Support Staff			
Health and Safety Supervisor	1.5	2.5	3.0
Health Physicist	1.5	2.5	3.0
Senior Health Physics Technician	12.0	20.0	36.0
Health Physics Technician	24.0	40.0	72.0
Protective Equipment Attendant	6.0	20.0	24.0
Industrial Safety Specialist	1.5	2.5	3.0
Industrial Safety Technician	3.0	5.0	6.0
Security Supervisor	1.5	2.5	3.0
Security Shift Supervisor	6.0	10.0	12.0
Security Patrolman	72.0	120.0	144.0
Contracts and Accounting Supervisor	1.5	2.5	3.0
Accountant	1.5	2.5	6.0
Contracts Specialist	1.5	2.5	3.0
Insurance Specialist	1.5	2.5	3.0
Procurement Specialist	1.5	2.5	3.0
Clerk	3.0	10.0	12.0
Quality Assurance Supervisor	1.5	2.5	3.0
Quality Assurance Engineer	1.5	5.0	6.0
Quality Assurance Technician	1.5	5.0	6.0
Subtotals	144.0	260.0	351.0
Plant Operations Staff			
Plant Operations Supervisor	1.5	2.5	3.0
Plant Chemist	1.5	2.5	3.0
Chemist	3.0	5.0	6.0
Reactor Operations Engineer	1.5	2.5	3.0
Engineer	3.0	5.0	6.0
Reactor Operations Shift Supervisor	6.0	10.0	12.0
Senior Reactor Operator	12.0	20.0	24.0
Reactor Operator	24.0	40.0	48.0
Utility Operator	24.0	40.0	48.0
Technician	18.0	30.0	36.0
Craft Supervisor	1.5	2.5	3.0

(contd on next page)

TABLE 10.2-1. (contd)

Position	Utility Staff Labor Requirements (man-years) for Preparations for Accident Cleanup		
	Scenario 1 Accident[(a)]	Scenario 2 Accident[(b)]	Scenario 3 Accident[(c)]
Crew Foreman	6.0	10.0	12.0
Maintenance Mechanic	24.0	40.0	48.0
Instrument Technician	24.0	40.0	48.0
Warehouseman	6.0	10.0	12.0
Tool Crib Attendant	6.0	10.0	12.0
Subtotals	162.0	270.0	324.0
Cleanup Planning Staff			
Cleanup Planning Supervisor	1.5	2.5	3.0
Engineering Supervisor	1.5	2.5	6.0
Engineer	9.0	20.0	36.0
Estimator	1.5	5.0	12.0
Draftsman	3.0	10.0	18.0
Crew Leader	1.5	5.0	6.0
Utility Operator	6.0	20.0	48.0
Craftsman	12.0	30.0	48.0
Laborer	6.0	20.0	48.0
Subtotals	42.0	115.0	225.0
Totals	368	690	972

(a) Based on a preparations for cleanup period of 1.5 years.
(b) Based on a preparations for cleanup period of 2.5 years.
(c) Based on a preparations for cleanup period of 3 years.

accident, 690 man-years for preparations for accident cleanup following the scenario 2 accident, and 972 man-years for preparations for accident cleanup following the scenario 3 accident.

The staff labor requirements shown in Table 10.2-1 do not include contractor labor to provide engineering support services during preparations for cleanup. An engineering contractor is assumed to provide assistance to the utility staff in preparing documentation for regulatory agencies, developing detailed work plans and work schedules, and preparing design specifications for the special equipment and facilities required for accident cleanup. Contractor labor for engineering services during preparations for cleanup is estimated to be 20 man-years per year (30 total man-years) following the scenario 1 accident, 40 man-years per year ([illegible] total man-years) following the scenario 2 accident, and 60 man-years per year (180 total man-years) following the scenario 3 accident.

10.3 ACCIDENT CLEANUP IN THE AUXILIARY AND FUEL BUILDINGS

The auxiliary and fuel buildings contain many components of safety-related systems (e.g., the tanks, pumps, piping, and filter and ion exchanger vaults for the chemical and volume control system and the liquid radioactive waste treatment systems) as well as the spent fuel storage pool and fuel handling equipment. Reliable operation of this equipment is necessary to ensure that the reactor is maintained in a safe shutdown condition until it is defueled and to allow the defueling to be accomplished safely and efficiently. If the accident results in substantial fission-product contamination in the auxiliary and fuel buildings, decontamination of these buildings is necessary to permit routine access by plant personnel to perform required operational and maintenance tasks without the need for elaborate protective clothing and respiratory protection devices.

Fission product contamination of the auxiliary and fuel buildings is postulated for the scenario 2 and scenario 3 accidents. (See Section 8.3 of Chapter 8.) Cleanup operations in these buildings are assumed to proceed concurrently with planning and preparation for containment building cleanup. Accident cleanup procedures, cleanup schedules, and cleanup worker manpower requirements for accident cleanup in the auxiliary and fuel buildings are summarized in the following subsections. Details of accident cleanup in these buildings are presented in Section E.3 of Appendix E.

10.3.1 Procedures for Accident Cleanup in the Auxiliary and Fuel Buildings

Decontamination of the auxiliary and fuel buildings following the scenario 2 and scenario 3 accidents has as goals the reduction of general area radiation exposure levels to about 1 to 3 mR/hr and the reduction of exposure levels in work areas (e.g., areas that contain coolant water treatment and radwaste treatment system components) to about 10 to 30 mR/hr. Reduction of radiation dose rates to these levels is desirable to permit routine worker access to these areas for necessary operations and maintenance activities during accident cleanup and decommissioning of the containment building.

The tasks which must be completed for accident cleanup in the auxiliary and fuel buildings include the following:

1. Decontaminate the fuel building to permit access to the spent fuel pool.
2. Remove accumulated spent fuel to provide space in the spent fuel pool for the filter/demineralizer system used to process contaminated water.
3. Install the filter/demineralizer system in the spent fuel pool.
4. Flush contaminated tanks and pipes and process the contaminated liquid through the filter/demineralizer system.
5. Continue the decontamination of auxiliary and fuel building surfaces and equipment.
6. Replace contaminated filters and ion exchange resins.
7. Perform maintenance or repair of systems or equipment items needed for processing accident water, defueling the reactor, or cleanup of the reactor coolant system.
8. Solidify and package wastes from accident cleanup in the auxiliary and fuel buildings.
9. Perform radiation survey to determine the extent of residual contamination following accident cleanup and to verify the effectiveness of cleanup operations.

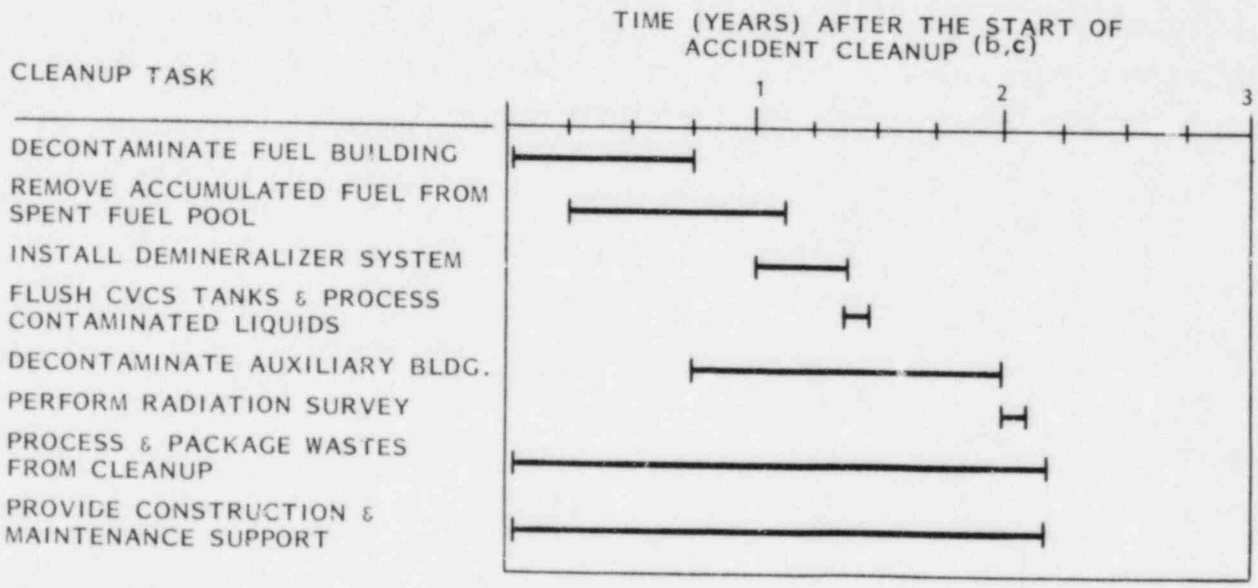
The demineralizer system is described in Section E.4.1 of Appendix E. One leg of the system is used for processing contaminated CVCS liquids. Details of procedures used to decontaminate building surfaces and equipment are given in Section D.1 of Appendix D.

10.3.2 Schedule for Accident Cleanup in the Auxiliary and Fuel Buildings

A sequence and schedule for accident cleanup in the auxiliary and fuel buildings is shown in Figure 10.3-1. Accident cleanup in these buildings is postulated to begin during preparations for cleanup in the containment building and to be substantially completed before containment building cleanup begins. The total time requirement for cleanup in these buildings is estimated to be about 2.2 years.

10.3.3 Cleanup Worker Requirements for Accident Cleanup in the Auxiliary and Fuel Buildings

Accident cleanup in the auxiliary and fuel buildings is accomplished by a staff of cleanup workers that is added to the staff for preparations for cleanup shown in Figure 10.2-1. The cleanup staff includes decontamination crews, a crew that provides construction and maintenance support, and waste processing and waste packaging crews. As discussed in Section E.3.2 of



(a) SCHEDULE DETAILS ARE GIVEN IN TABLE E.3-1 OF APPENDIX E.

(b) CLEANUP IN THE AUXILIARY AND FUEL BUILDINGS IS ASSUMED TO TAKE PLACE DURING PREPARATIONS FOR CLEANUP IN THE CONTAINMENT BUILDING.

(c) THE TOTAL TIME REQUIREMENT FOR CLEANUP OF THESE BUILDINGS IS ESTIMATED TO BE ABOUT 2.2 YEARS.

FIGURE 10.3-1. Sequence and Schedule for Accident Cleanup in the Auxiliary and Fuel Buildings(a)

Appendix E, the labor requirement for completion of accident cleanup in the auxiliary and fuel buildings, based on the schedule shown in Figure 10.3-1, is 129 man-years. This estimate includes only the labor to actually complete the tasks shown in the figure and does not include operations and site support personnel who are included in the staff labor requirements for preparations for cleanup. The cleanup labor requirement of 129 man-years also does not take account of the extra labor needed to maintain compliance with the occupational radiation dose limit, taken in this study as 5 rem/year.[2]

The total estimated occupational radiation dose for accident cleanup in the auxiliary and fuel buildings is 1613 man-rem. (See Section E.3.3 of Appendix E.) As explained in Section E.3.4, an upward adjustment factor of about 2.6 must be applied to the various cleanup worker categories to bring the estimated occupational radiation dose for individual workers down to 5 rem/year. The adjusted manpower requirement for accident cleanup in the auxiliary and fuel buildings is shown in Table 10.3-1. This total adjusted cleanup worker requirement of 334 man-years is used in computing staff labor costs for accident cleanup in the auxiliary and fuel buildings following the scenario 3 accident. (See Section F.2.1 of Appendix F.)

TABLE 10.3-1. Estimated Cleanup Worker Requirements for Accident Cleanup in the Auxiliary and Fuel Buildings

Worker Category	Adjusted Worker Requirement[a] (man-yr)
Cleanup Operations Supervisor	2.2
Crew Leader	42.5
Utility Operator	65.0
Laborer	65.8
Craftsman	115.7
Health Physics Technician	43.0
Total	334

(a) Adjusted worker requirement to comply with an occupational radiation dose limit of 5 rem/yr.[2] Details are given in Table E.3-3 of Appendix E.

10.4 ACCIDENT CLEANUP IN THE CONTAINMENT BUILDING

Accident cleanup procedures, cleanup schedules, and staff labor requirements for accident cleanup in the containment building following the reference accidents are summarized in the following subsections. Details of accident cleanup in the containment building are given in Section E.4 of Appendix E.

The reference accidents are described in Chapter 8. Accident cleanup operations are assumed to reduce general area radiation exposure rates in the containment building to the values shown in Table 10.4-1. As discussed in Section E.1, decontamination activities during accident cleanup are not designed to reduce exposure rates to levels permitting unrestricted use of the facility, but only to limit the doses to workers engaged in accident cleanup. Additional decontamination would be required during decommissioning (or refurbishment) to limit the doses to workers engaged in these activities. Because contamination levels in the containment building at the beginning of accident cleanup are different for the three accident scenarios, and because only selective decontamination of surfaces and equipment takes place during cleanup operations, the average general area exposure rates at the conclusion of accident cleanup are postulated to be different for the three reference accident scenarios.

TABLE 10.4-1. Average General Area Exposure Rates in the PWR Containment Building at the Completion of Accident Cleanup Operations

	Average Exposure Rate (mR/hr)		
Location	Cleanup Following Scenario 1 Accident	Cleanup Following Scenario 2 Accident	Cleanup Following Scenario 3 Accident
Operating Floor Level	3	10	30
Mezzanine Level	5	20	50
Ground Floor Level	10	50	100

10.4.1 Procedures for Accident Cleanup in the Containment Building

Accident cleanup in the PWR containment building is postulated to include the following tasks:

- processing of contaminated liquids
- initial decontamination of the containment building
- defueling of the reactor
- cleanup of the primary coolant system
- treatment and disposal of wastes from cleanup operations.

10.4.1.1 Processing of Contaminated Liquids

Contaminated liquids that must be processed during accident cleanup include accident water (radioactively contaminated water that is released to the containment building during an accident and that collects in sumps or in the containment building basement), contaminated water and chemical decontamination solutions generated during decontamination of containment building surfaces, reactor coolant system water, and reactor coolant system chemical decontamination and flush solutions. Estimated volumes of contaminated liquids from accident cleanup following the postulated accidents and the curies of radioactivity removed from these liquids are given in Section E.4.2 of Appendix E.

Contaminated water is assumed to be treated by filtration and demineralization. The existing liquid waste treatment system in the reference PWR is not adequate for the treatment of accident water or of most water-based decontamination solutions from accident cleanup, since it was designed to process water with significantly lower concentrations of radioactivity than exist in these accident liquids. Contaminated water is treated by use of the filter/demineralizer system that is designed and installed in the spent fuel pool in the fuel building during preparations for accident cleanup. The postulated demineralizer system is similar to the submerged demineralizer system (SDS) used to process accident water at TMI-2.[3]

Details of the demineralizer system are given in Section E.4.1.1 of Appendix E. The process train consists of a prefilter, final filter, two parallel trains of three ion exchange vessels each that employ zeolite ion exchange media, two downstream cation exchange vessels used to polish the effluent from the zeolite vessels, and a post filter. The design flow rate through each train is 0.02 m^3/min and the design flow rate through both trains is 0.04 m^3/min. The zeolite vessels are capable of radioactivity loadings of up to 60,000 Ci. The design objective of the system is a radionuclide concentration of less than 0.0001 Ci/m^3 in the processed water.

Water that has been processed in the filter/demineralizer system is stored in the 1000-m^3-capacity storage tanks constructed onsite during preparations for accident cleanup. The processed water can be reused for building decontamination and reprocessed. In this study, it is assumed that if the processed water is not needed for reuse it can be discharged to the river under controlled conditions. Prior to discharge of the water to the river, processing would have reduced the contamination levels to values that comply with the limits discussed in Section 5.3.1 of Chapter 5. The processed water would also be treated by evaporation to remove the boron. The clean radioactive waste evaporator located in the auxiliary building and described in Section B.2 of Appendix B could be used for this operation. It is recognized in this study that following an accident there could be restrictions against the discharge of processed water. Other alternatives for water disposition are discussed in Section 5.3.1 of Chapter 5. These other alternatives are not treated in detail in this study; however, a discussion of their relative costs is included in the discussion of the sensitivity of costs to various factors in Section 11.6.

Chemical decontamination solutions from initial cleanup operations have radionuclide concentrations in the range from 1 to 100 Ci/m^3. Evaporation is a suitable alternative for treatment of these wastes. However, the existing clean radioactive waste evaporator system located in the auxiliary building does not have the capacity to handle the volumes or radioactivity concentrations of the decontamination liquids from initial cleanup. An evaporator/solidification facility is postulated to be obtained from a commercial supplier and installed in the auxiliary building during preparations for cleanup. This evaporator is assumed to process chemical decontamination solutions at a rate of approximately 0.06 m^3/min. The evaporator bottom liquids are solidified with vinyl ester styrene and packaged in stainless steel liners for interim onsite storage in the shielded storage facility that is constructed during preparations for accident cleanup.

10.4.1.2 Initial Decontamination of the Containment Building

The objective of initial decontamination of the PWR containment building is to reduce surface contamination levels and resultant radiation exposure levels to permit reasonable occupancy times for workers engaged in reactor defueling and reactor coolant system cleanup operations. In addition to surface decontamination procedures, reduction of general area radiation exposure rates requires the removal and processing of reactor building sump water and the removal or shielding of contaminated "crud" or sludge deposits that remain on the walls and floors of the reactor building basement after the sump water is removed. The reduction of general area radiation exposure rates at the defueling location requires that "hot spots" be shielded by using lead sheet or lead bricks, high-density concrete blocks, or containers filled with water.

A general discussion of procedures for the decontamination of surfaces and equipment is given in Appendix D. Initial decontamination of the containment building includes the following steps:

1. Utilize the containment building spray system for a remote wash of building surfaces.
2. Remove and package debris and small items of contaminated equipment that are easily disposed of.
3. Employ high-pressure hose wash techniques for semi-remote decontamination of building surfaces and equipment.
4. Decontaminate and refurbish or replace essential support systems.
5. Perform hands-on decontamination of selected areas where significant reductions in radiation exposure can be achieved with modest effort. Decontaminate floors by scrubbing.
6. Provide local shielding of "hot spots."

For the scenario 2 and scenario 3 accidents, the contribution of sump water contamination to the average background dose rate is so high (see Table 8.3-1) that it is deemed advisable to process the sump water through the demineralizer system before entry of personnel into the containment building to begin surface decontamination operations. The processed water is then postulated to be returned to the building basement to provide shielding from the contamination on basement surfaces.

Water from remote spray and hose-wash decontamination operations is collected in the building basement. Remote spray and hose-wash operations are carefully coordinated with sump water processing operations to maintain an approximately constant water level in the building basement until the hosing of surfaces above the basement level is completed. After the hosing of these surfaces is completed, as the water level in the basement is lowered, basement surfaces are washed with high-pressure hoses to remove surface contamination and "crud" deposits.

10.4.1.3 Defueling the Reactor

The difficulty of the reactor defueling operation is determined by the amount of damage to the core and to the reactor vessel during the accident. Damage to the fuel, to the reactor vessel head, and to internal support structures is postulated to be different for each accident scenario evaluated in this report.

Defueling the reactor following an accident includes the following steps:

- preparations for defueling
- removal of the reactor pressure vessel head and inspection of the core
- removal of structural components above the fuel
- removal of intact fuel assemblies and removal and packaging of damaged fuel assemblies
- removal of fuel element debris.

These steps are discussed in the following paragraphs.

Preparations for Defueling. Preparations for defueling include the following operations:

- install work platforms and equipment needed for defueling
- install temporary radiation shielding
- remove insulation from reactor vessel head
- decontaminate external surfaces of reactor pressure vessel (RPV)
- disconnect electrical cables and cooling water lines
- cleanup the primary system water
- prepare refueling cavity for flooding.

The polar crane which is needed for removal of the reactor vessel head and vessel internals may be badly contaminated or damaged as a result of the accident. The crane must be decontaminated and refurbished prior to the start of defueling operations.

Prior to removal of the reactor vessel head and filling of the refueling cavity with water, the amount of radioactivity in the primary system water must be reduced to minimize the effect of this water as a source of radiation exposure to workers engaged in defueling operations. The cleanup of primary system water is accomplished through a "feed and bleed" process whereby water is removed from the system, processed by the filter/demineralizer system, and replaced by clean borated water from the refueling water storage tank. This process is continued until the amount of radionuclides removed is the same as that being produced in the water by the damaged core.

Removal of RPV Head and Inspection of the Core. Under best-case conditions (assumed for the scenario 1 and scenario 2 accidents) the RPV head closure nuts can be removed from the studs using normal procedures, the control rod cluster assemblies can be readily disconnected from the control rod drives, and the RPV head can be lifted using the normal head-lifting fixture and polar crane.

For the scenario 3 accident, it is postulated that stud removal requires splitting or stripping jammed nuts and cutting off or machining out some of the difficult studs. Disconnection of control rods that are stuck in the reactor core or plenum grid requires cutting the control rod drive shafts and/or lead screws. To remove the pressure vessel head, rigging is attached to the head and secured to the polar crane. Jacking equipment is installed and used to separate the head from the pressure vessel. The head is then lifted using the polar crane.

After removal of the RPV head, the insides of the reactor vessel and the fuel core are inspected using periscopes and television cameras. The purpose of this inspection is to determine the extent of damage so as to define the special procedures or special tools needed for the removal of structural components and fuel assemblies.

Removal of Structural Components. If the upper core support assembly is not damaged, it can be removed as a unit, using the support assembly handling fixture and the polar crane. For the scenario 3 accident, the upper core support assembly is assumed to be stuck in place, making it necessary to cut it out in pieces that are packaged in canisters for interim storage.

Removal of Fuel Assemblies. For the scenario 1 accident, most of the fuel assemblies can be removed from the core by the normal extraction method that involves lifting the fuel assembly from the top with a handling device that attaches to the top end-fitting. Assemblies that have experienced structural damage that might cause them to break apart as they are lifted from the top are removed from the core by using a special handling device that provides support to the bottom and sides of the assembly. Conceptual handling tools for the support and removal of damaged fuel are described in Section E.4.1 of Appendix E.

For the scenario 2 accident, some of the central fuel assemblies are bound or fused together at the spacer grid elevations and cannot be individually removed. Peripheral fuel assemblies are assumed not to have been damaged to an extent that prevents extraction of at least one complete assembly using the normal fuel handling equipment. The cavity created by removal of one peripheral fuel assembly permits a sequential extraction of adjacent assemblies radially toward the center of the core by the use of equipment that supports the assembly at the bottom and/or along the length of the assembly.

At some point the sequential removal activity reaches those fuel assemblies near the center of the core that have sustained the most damage and/or are fused together. These assemblies are either removed as a unit or equipment is used to cut them apart at the points where they are fused together.

For the scenario 3 accident, core damage is assumed to be so extensive that all of the fuel assemblies have sustained some damage and none of the peripheral assemblies can be removed by lifting the assembly by the top end-fitting. The opening of a full-length cavity on the periphery of the core is required to remove the first fuel assembly. This initial cavity is formed by cutting and removing the baffle plates in a segment of the core support structure that provides access to the selected peripheral fuel assembly. Removal of adjacent fuel assemblies then progresses as for the scenario 2 accident. A specially designed hydraulic jack is used to aid in releasing fuel assemblies from their pockets in the lower grid plate supporting the core assembly.

Damaged fuel assemblies removed from the core require packaging in canisters prior to storage in the spent fuel pool. Canning of these assemblies is performed in the refueling cavity. A conceptual fixture used as an aid in the packaging of damaged fuel assemblies is shown in Figure E.4-4 of Appendix E.

10.4.1.4 Cleanup of the Primary Coolant System

Primary coolant system components to be decontaminated include the reactor coolant system (RCS), the charging, letdown, and seal water portion of the chemical and volume control system (CVCS), and associated piping and intertied systems. Decontamination includes the removal of fuel debris (scenario 2 and scenario 3 accidents) and the removal of fission product plateout (all three accident scenarios).

To dissolve the fuel debris located in pumps, piping, and other components of the RCS, an oxalic-peroxide-gluconic (OPG) solution is postulated to be used. The properties and use of OPG solution are described in

Section D.1.1 of Appendix D. One system volume of OPG solution is assumed to be required for fuel dissolution following the scenario 2 accident and two system volumes are assumed to be required following the scenario 3 accident.

To remove fission product plateout from internal surfaces of coolant system components, the chelating agent ethylenediamenetraacetic acid (EDTA) is postulated to be used in combination with citric and oxalic acid in a weak (5%) solution at controlled pH. (The use of concentrated decontamination solutions that are highly corrosive is not considered in this study for reasons that are discussed in Section E.1 of Appendix E.) The properties and use of EDTA/oxalic/citric acid solutions are also described in Section D.1.1 of Appendix D. One system flush with EDTA solution following the scenario 1 and scenario 2 accidents, and two flushes with EDTA solution following the scenario 3 accident are postulated for cleanup. Because of the incompatibility of the OPG and the EDTA/oxalic/citric acid solutions, a system flush with processed water is interposed between use of the two solutions. A second system flush with processed water completes the primary coolant system cleanup.

The reactor coolant system pumps are assumed to be operable and are used for circulation of the decontamination solutions following the scenario 1 and scenario 2 accidents. Extensive repairs to pump motors are assumed to be required before use of these pumps following the scenario 3 accident. In-plant tanks are used to mix the decontamination solutions which are then pumped to the primary coolant system. The regenerative heat exchanger or the letdown heat exchanger are used to heat the solutions. Processing of contaminated solutions is performed in the evaporator/solidification system installed in the auxiliary building during preparations for cleanup.

10.4.1.5 Waste Treatment and Disposal

Radioactive wastes from accident cleanup operations can be divided into four categories, as follows:

1. Solid Materials. Dry radioactive wastes generated from decontamination and defueling operations. These materials consist of trash, contaminated equipment and material, and irradiated, activated hardware.

2. Process Solids. Contaminated sludges and process solid wastes that arise from the treatment of accident water and decontamination liquids. These solid wastes include filter cartridge assemblies, ion exchange media (inorganic zeolites and organic resins), and evaporator bottoms.

3. Chemical Decontamination Solutions. Liquid decontamination wastes that have not been treated to generate process solids. These wastes are immobilized by incorporation in cement or in vinyl ester styrene.

4. Fuel Assemblies and Core Debris. Damaged and undamaged fuel assemblies and the core debris (fuel, cladding, and hardware) removed from the reactor vessel during defueling operations.

The alternatives assumed in this study for the packaging and disposal of these wastes and the waste volumes generated during accident cleanup in the containment building are given in Table E.4-2 of Appendix E.

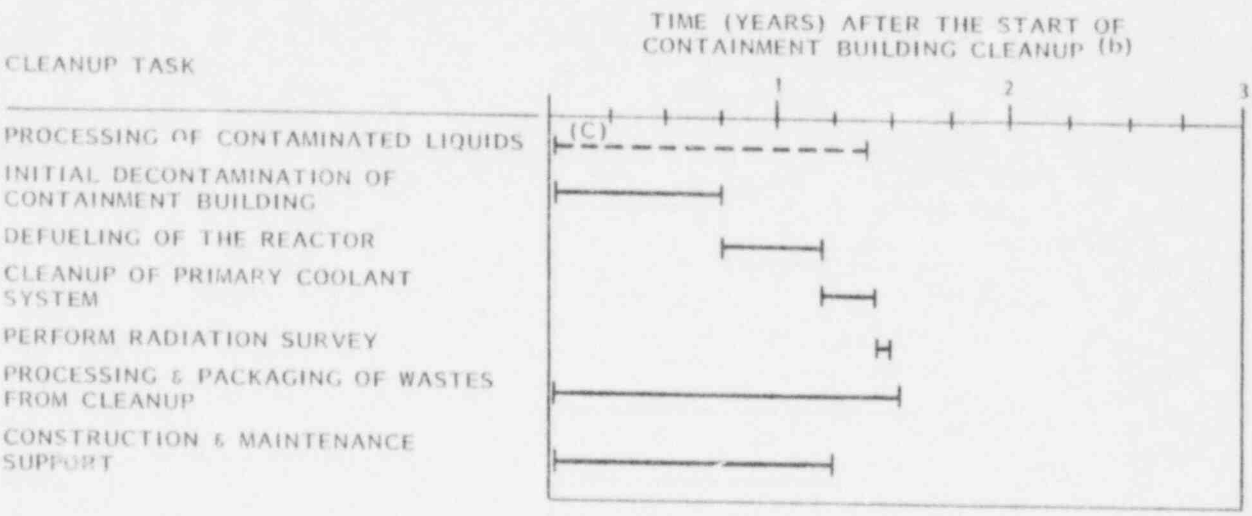
Based on the criteria of 10 CFR, Part 61,[4] all of the wastes from containment building cleanup, except the process solids, the fuel assemblies and core debris, are assumed to be transported to a shallow-land burial ground for disposal. Because of their high curie content, process solids (filters, ion exchange media, and evaporator bottoms) are postulated to be unacceptable for near-surface burial according to 10 CFR 61 criteria. Hence, they are assumed to be transported to a federal repository for storage or disposal. Damaged fuel and fuel debris is also shipped to a federal repository. Intact fuel assemblies from cleanup following the scenario 1 accident are shipped to an ISFSI. The estimated disposal volumes and waste management costs for radioactive wastes shipped to a shallow-land burial ground, a federal repository, or an ISFSI are shown in Table 11.3-2 of Chapter 11.

Because of the potential that a reactor involved in an accident may be unable to dispose of its wastes, either because of lack of disposal capacity or regulatory or political constraints, this study also analyzes extended onsite storage of low-level wastes, highly radioactive wastes, and spent fuel assemblies. This analysis of the impacts of temporary onsite storage of radioactive wastes is presented in Chapter 15.

10.4.2 Schedules for Accident Cleanup in the Containment Building

Schedules for accident cleanup in the containment building following the reference PWR accidents are shown in Figures 10.4-1, 10.4-2, and 10.4-3. Time requirements for accident cleanup depend on accident severity and are based on the cleanup procedures summarized in Section 10.4.1. Accident cleanup in the containment building is estimated to require approximately 1.5 years following the scenario 1 accident, 2.8 years following the scenario 2 accident, and 5.0 years following the scenario 3 accident.

Details of accident cleanup schedules and the bases and assumptions used to prepare these schedules and to estimate cleanup worker requirements are given in Section E.4.2 of Appendix E.

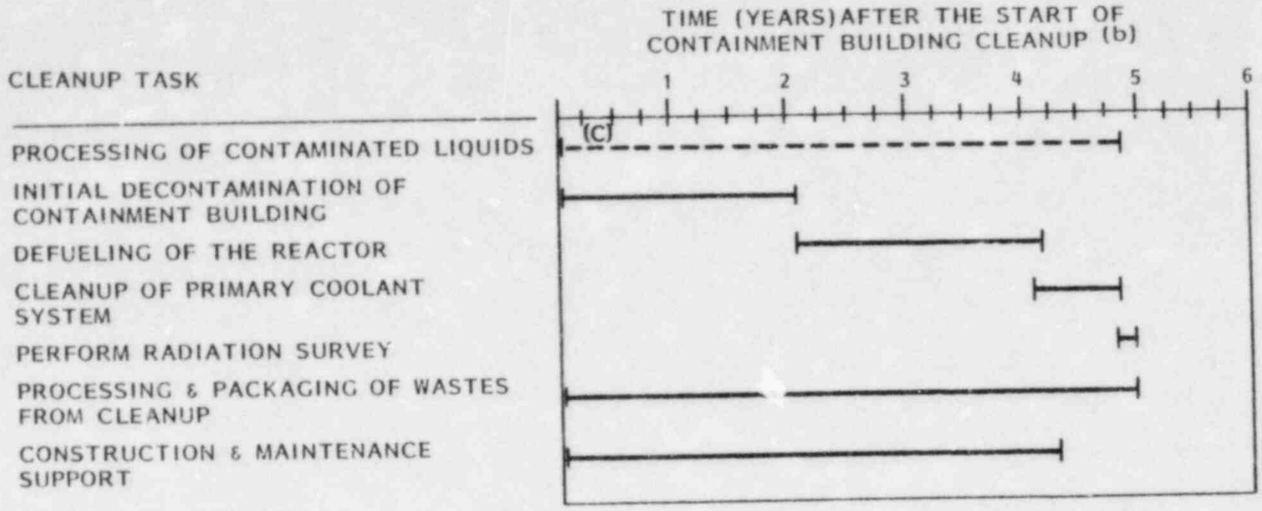


(a) SCHEDULE DETAILS ARE GIVEN IN FIGURE E.4-5 OF APPENDIX E.

(b) THE TOTAL TIME REQUIREMENT FOR ACCIDENT CLEANUP IN THE CONTAINMENT BUILDING FOLLOWING THE SCENARIO 1 ACCIDENT IS 1.5 YEARS.

(c) AS REQUIRED DURING THIS TIME PERIOD.

FIGURE 10.4-1. Sequence and Schedule for Accident Cleanup in the Containment Building Following the Scenario 1 Accident(a)

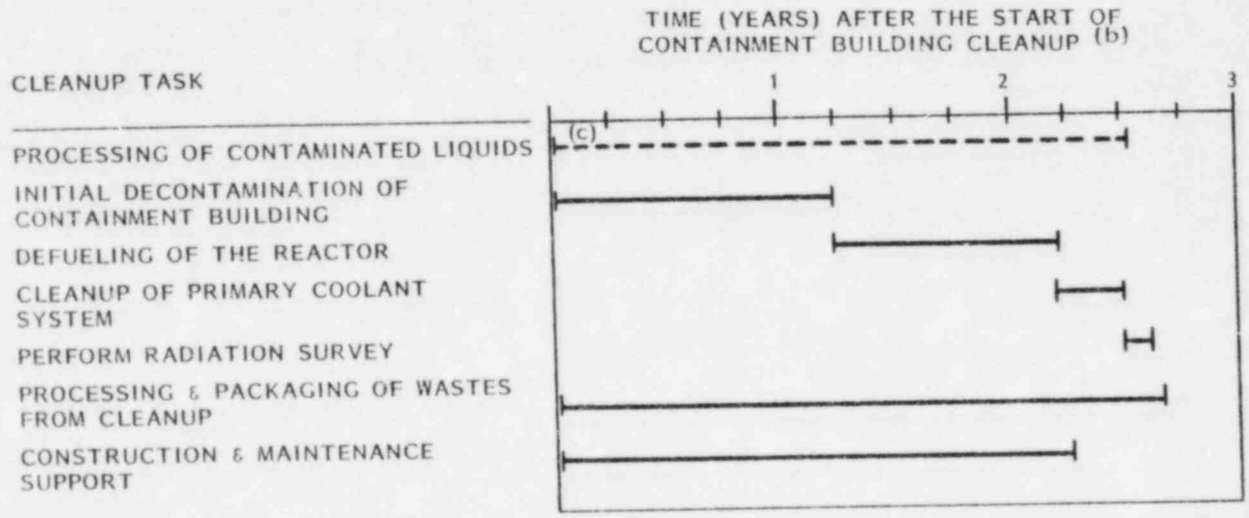


(a) SCHEDULE DETAILS ARE GIVEN IN FIGURE E.4-7 OF APPENDIX E.

(b) THE TOTAL TIME REQUIREMENT FOR ACCIDENT CLEANUP IN THE CONTAINMENT BUILDING FOLLOWING THE SCENARIO 3 ACCIDENT IS 5 YEARS.

(c) AS REQUIRED DURING THIS TIME PERIOD.

FIGURE 10.4-2. Sequence and Schedule for Accident Cleanup in the Containment Building Following the Scenario 2 Accident[a]



(a) SCHEDULE DETAILS ARE GIVEN IN FIGURE E.4-6 OF APPENDIX E.

(b) THE TOTAL TIME REQUIREMENT FOR ACCIDENT CLEANUP IN THE CONTAINMENT BUILDING FOLLOWING THE SCENARIO 2 ACCIDENT IS 2.8 YEARS.

(c) AS REQUIRED DURING THIS TIME PERIOD.

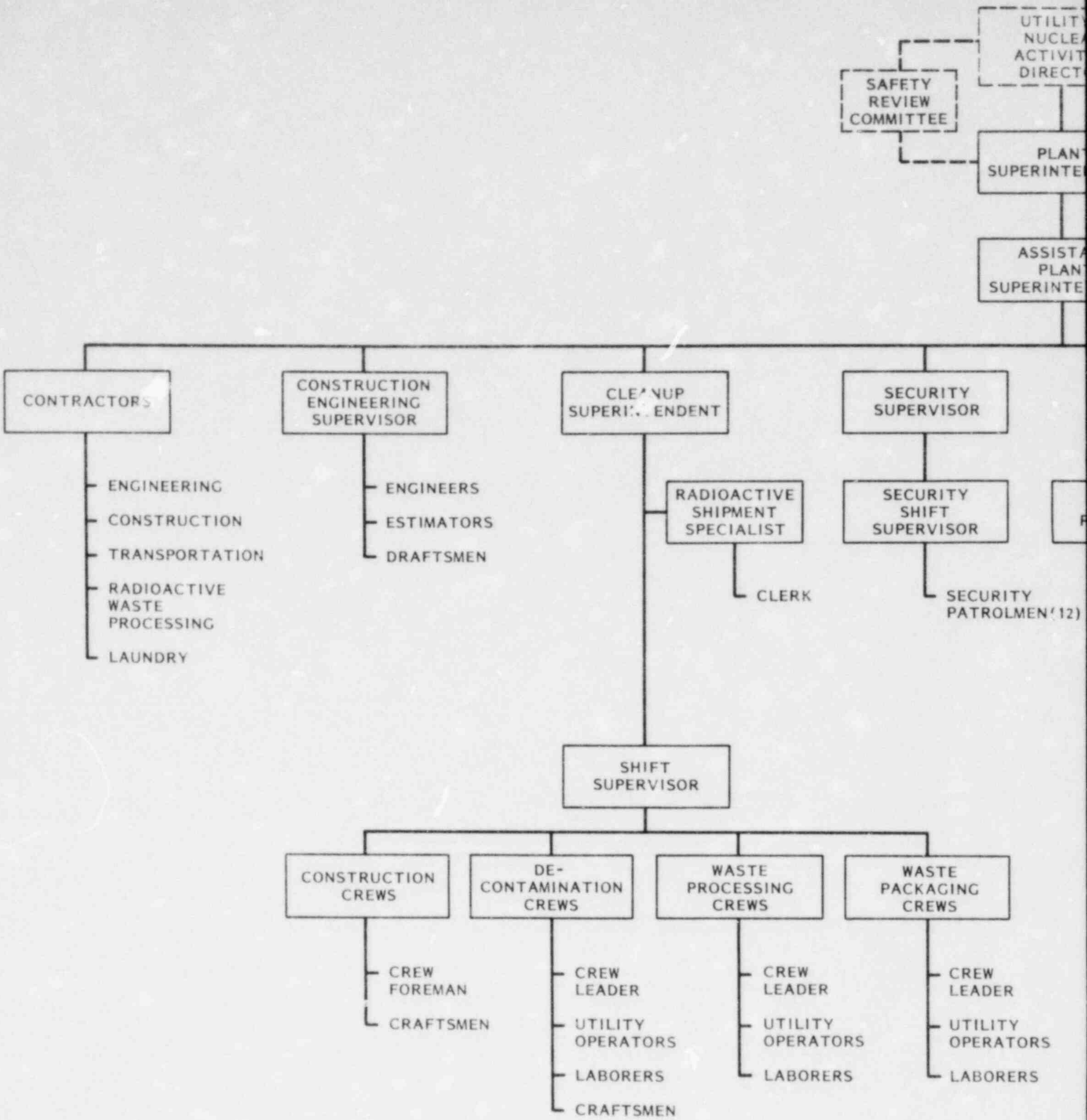
FIGURE 10.4-3. Sequence and Schedule for Accident Cleanup in the Containment Building Following the Scenario 3 Accident[a]

10.4.3 Staff Labor Requirements for Accident Cleanup in the Containment Building

A postulated staff organization for accident cleanup in the containment building is shown in Figure 10.4-4. The utility staff for accident cleanup includes a plant operations branch and several site support branches (e.g., engineering, health and safety, security, contracts and accounting, and quality assurance) as well as the staff actually involved in cleanup in the containment building.

Estimated utility staff labor requirements for accident cleanup in the containment building are shown in Table 10.4-2. These labor requirements are 450 man-years for cleanup following the scenario 1 accident, 1268 man-years for cleanup following the scenario 2 accident, and 3414 man-years for cleanup following the scenario 3 accident. The accident cleanup staff labor requirements (the man-hours for personnel engaged in cleanup operations inside the containment building) shown in Table 10.4-2 have been adjusted upward by appropriate factors to ensure that the estimated occupational radiation dose for individual workers does not exceed 5 rem/year.(2) An explanation of the adjustment factors used to obtain man-years for accident cleanup staff labor is given in Section E.4.4 of Appendix E. The utility staff labor requirements shown in Table 10.4-2 are used in computing utility staff labor costs for accident cleanup in the containment building. (See Section F.3.1 of Appendix F.)

The accident cleanup staff labor contribution to the total utility staff labor requirement for accident cleanup in the containment building increases from 33% for cleanup following the scenario 1 accident to 61% for cleanup following the scenario 3 accident. The management, plant operations, and site support labor contributions to the total utility staff labor requirement decrease from 67% for cleanup following the scenario 1 accident to 39% for cleanup following the scenario 3 accident. Management, plant operations, and site support labor requirements are unaffected by accident severity except that the total labor requirement is a function of the duration of the accident cleanup period. Labor requirements for accident cleanup personnel are



NOTES:

1) NUMBERS IN PARENTHESES REFER TO THE NUMBER OF PERSONNEL PER SHIFT. WHEN THE NUMBER OF PERSONS IS NOT INDICATED, IT IS EITHER UNITY OR VARIES WITH ACCIDENT SCENARIO. TOTAL PERSONNEL REQUIREMENTS FOR EACH ACCIDENT SCENARIO ARE GIVEN IN TABLE E.4-10.

2) THE SECURITY FORCE, THE HEALTH PHYSICS STAFF, THE REACTOR OPERATIONS STAFF, AND THE PLANT MAINTENANCE STAFF FUNCTION ON A 24-HOUR-PER-DAY, 7-DAY-WEEK BASIS.

FIGURE 10.4-4. Postulated Staff Organization for Acc

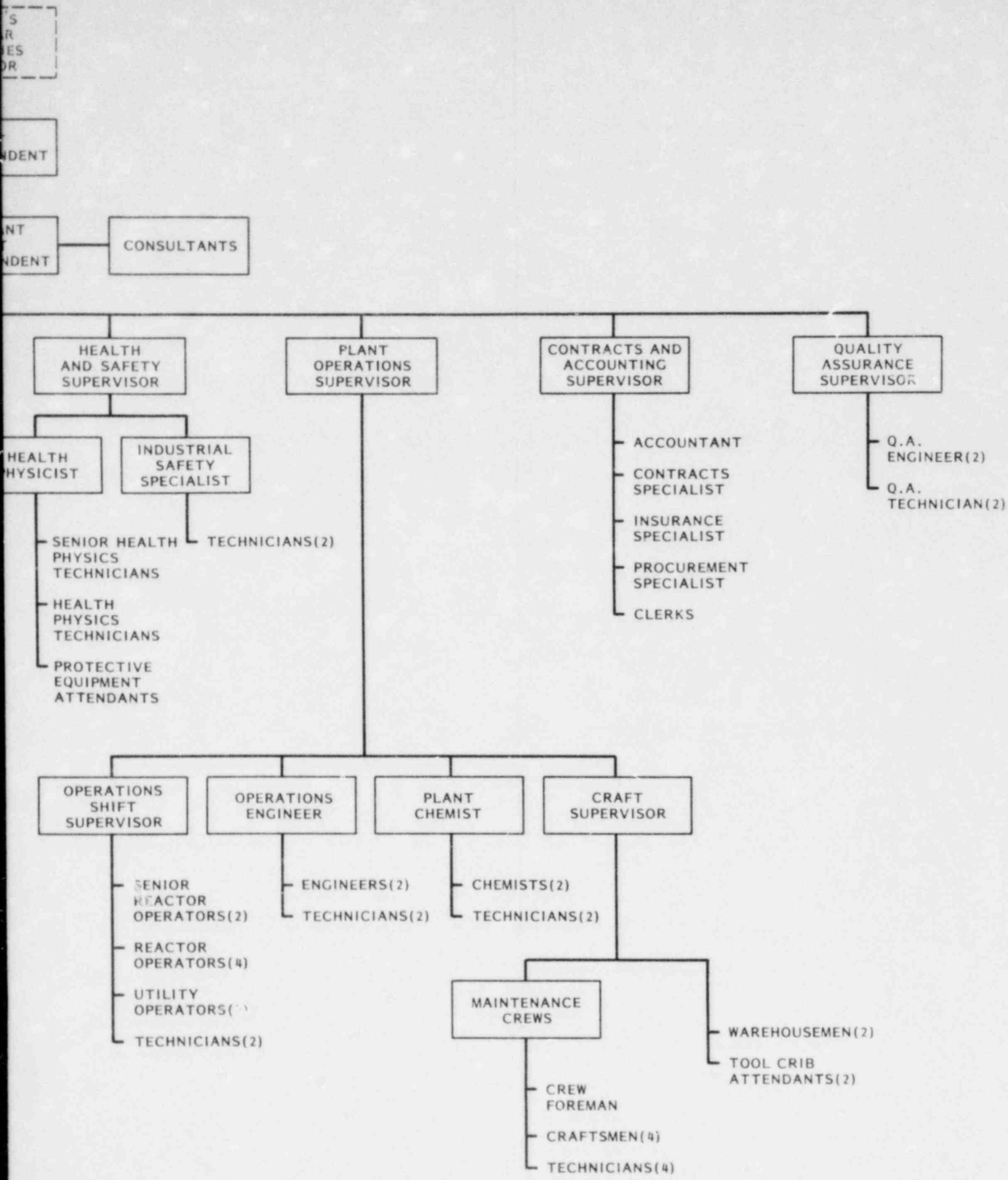


FIGURE 10.4-4. Postulated Staff Organization for Accident Cleanup in the Containment Building

ident Cleanup in the Containment Building

TABLE 10.4-2. Estimated Utility Staff Labor Requirements for Accident Cleanup in the Containment Building

Position	Utility Staff Labor Requirements (man-years) for Cleanup Following: Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
Plant Superintendent	1.5	2.8	5.0
Assistant Plant Superintendent	1.5	2.8	5.0
Consultants	4.5	16.8	50.0
Secretaries and Word Processors	12.0	28.0	60.0
Site Support Staff			
Health and Safety Supervisor	1.5	2.8	5.0
Health Physicist	1.5	2.8	5.0
Senior Health Physics Technician	12.0	22.4	60.0
Health Physics Technician[a]	12.0	22.4	60.0
Protective Equipment Attendant	6.0	22.4	60.0
Industrial Safety Specialist	1.5	2.8	5.0
Industrial Safety Technician	3.0	5.6	10.0
Security Supervisor	1.5	2.8	5.0
Security Shift Supervisor	6.0	11.2	20.0
Security Patrolman	72.0	134.4	240.0
Contracts and Accounting Supervisor	1.5	2.8	5.0
Accountant	1.5	2.8	10.0
Contracts Specialist	1.5	2.8	5.0
Insurance Specialist	1.5	2.8	10.0
Procurement Specialist	1.5	2.8	5.0
Clerk	3.0	11.2	30.0
Quality Assurance Supervisor	1.5	2.8	5.0
Quality Assurance Engineer	3.0	5.6	10.0
Quality Assurance Technician	3.0	5.6	10.0
Construction Engineering Supervisor	1.5	2.8	5.0
Engineer	9.0	22.4	60.0
Estimator	1.5	5.6	20.0
Draftsman	3.0	11.2	30.0
Subtotals	150.0	310.8	675.0
Plant Operations Staff			
Plant Operations Supervisor	1.5	2.8	5.0
Plant Chemist	1.5	2.8	5.0
Chemist	3.0	5.6	10.0
Reactor Operations Engineer	1.5	2.8	5.0
Engineer	3.0	5.6	10.0

(contd on next page)

TABLE 10.4-2. (contd)

Position	Utility Staff Labor Requirements (man-years) for Cleanup Following: Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
Reactor Operations Shift Supervisor	6.0	11.2	20.0
Senior Reactor Operator	12.0	22.4	40.0
Reactor Operator	24.0	44.8	80.0
Utility Operator	24.0	44.8	80.0
Technicians	24.0	56.0	120.0
Craft Supervisor	1.5	2.8	5.0
Crew Foreman	6.0	11.2	20.0
Craftsman[b]	12.0	33.6	60.0
Warehouseman	6.0	22.4	40.0
Tool Crib Attendant	6.0	22.4	40.0
Subtotals	132.0	291.2	540.0
Accident Cleanup Staff			
Cleanup Superintendent	1.5	2.8	5.0
Radioactive Shipment Specialist	1.5	2.8	5.0
Clerk	1.5	2.8	10.0
Shift Supervisor	6.0	11.2	20.0
Crew Leader[c]	15.2	62.4	195.5
Utility Operator[c]	50.3	203.6	605.2
Laborer[c]	29.4	111.0	321.0
Craftsman[c]	24.5	140.3	669.4
Health Physics Technician[c]	18.5	78.2	247.6
Subtotals	148.4	615.1	2078.7
Totals	449.9	1267.5	3413.7

(a) Additional health physics technicians counted as part of accident cleanup staff.
(b) Additional craftsmen counted as part of accident cleanup staff.
(c) Cleanup staff labor requirements are adjusted to limit individual radiation doses to 5 rem/yr.[2]

strongly affected by accident severity. For example, the increased difficulty of defueling the reactor and the greater radiation exposure levels experienced by cleanup workers following the scenario 3 accident result in a substantially greater cleanup worker labor requirement for scenario 1 accident cleanup than for scenario 1 accident cleanup.

The staff labor requirements shown in Table 10.4-2 do not include contractor labor to provide engineering support services during accident cleanup in the containment building. The contractor labor requirement for engineering support services is estimated to be 15 man-years (10 man-years per year) during cleanup following the scenario 1 accident, 56 man-years (20 man-years per year) during cleanup following the scenario 2 accident, and 150 man-years (30 man-years per year) during cleanup following the scenario 3 accident.

REFERENCES

1. Final Environmental Assessment for Decontamination of the Three Mile Island Unit 2 Reactor Building Atmosphere, NUREG-0662, U.S. Nuclear Regulatory Commission, Washington, D.C., May 1980.

2. U.S. Code of Federal Regulations, Title 10, Part 20.101, "Radiation Dose Standards for Individuals in Restricted Areas." (10 CFR 20.101), U.S. Government Printing Office, Washington, D.C., 1980.

3. Safety Evaluation Report Related to the Operation of the Submerged Demineralizer System at Three Mile Island Nuclear Station, Unit 2, NUREG-0796, U.S. Nuclear Regulatory Commission, Washington, D.C., June 1981.

4. "Licensing Requirements for Land Disposal of Radioactive Waste," Federal Register, Vol. 46, No. 142, pp. 38081-38100, July 24, 1981.

CHAPTER 11

COSTS OF ACCIDENT CLEANUP AT A REFERENCE PWR

The costs of accident cleanup activities at the reference PWR following the three postulated accidents are summarized in this chapter. These costs are developed in detail in Appendices F and I of Volume 2. Costs are in early-1981 dollars.

As discussed in earlier chapters of this study, accident cleanup activities would be similar whether the reactor is refurbished for restart or decommissioned. Hence the costs of accident cleanup presented here are considered to be a good representation independent of the ultimate use of the plant. Costs of stabilization activities and of activities related to refurbishment and restart of a reactor, beyond the accident cleanup activities, are not included in this study. Costs of decommissioning following accident cleanup are summarized in Chapter 13.

The costs of preparations for accident cleanup are presented in Section 11.1. The costs of accident cleanup in the auxiliary and fuel buildings are presented in Section 11.2. The costs of accident cleanup in the containment building are presented in Section 11.3. Section 11.4 gives a summary of the estimated costs of accident cleanup activities at the reference PWR. Section 11.5 presents a comparison of the estimated costs of accident cleanup in the reference PWR with estimated costs of cleanup and recovery at TMI-2, which is the only power reactor that has experienced a major accident. Section 11.6 presents a brief analysis of the sensitivity of cost estimates to various factors that can influence costs. These factors include such items as potential delays in the accident cleanup, uncertainties in the plant condition following an accident, alternatives for the processing and disposal of radioactive wastes, and use of contractor labor.

11.1 COSTS OF PREPARATIONS FOR ACCIDENT CLEANUP

The estimated costs of preparations for accident cleanup at the reference PWR following the three postulated accidents are summarized in Table 11.1-1.

TABLE 11.1-1. Summary of Estimated Costs of Preparations for Accident Cleanup at the Reference PWR

Cost Category	Preparations for Cleanup Following Scenario 1 Accident: Estimated Costs[a,b] ($ millions)		Percent of Total	Preparations for Cleanup Following Scenario 2 Accident: Estimated Costs[a,c] ($ millions)		Percent of Total	Preparations for Cleanup Following Scenario 3 Accident: Estimated Costs[a,d] ($ millions)		Percent of Total
Utility Staff Labor		12.808	47.5		24.382	45.3		35.016	44.7
Waste Management		0.125	0.5		0.371	0.7		0.471	0.6
Energy		7.227	26.8		12.045	22.4		14.454	18.4
Special Equipment and Facilities[e]									
Demineralizer System	1.000			1.000			1.000		
Fuel Racks for Canistered Fuel				0.310			0.310		
Processed Water Storage Tanks	0.135			0.270			0.405		
Facilities for Interim Storage of Wastes[f]	0.208			0.364			0.815		
Mock-up of Reactor Vessel				1.000			3.000		
Total Equipment and Facilities Costs		1.343	5.0		2.944	5.5		5.530	7.1
Miscellaneous Supplies		0.075	0.3		0.125	0.2		0.150	0.2
Specialty Contractors									
Engineering	3.000			10.000			18.000		
Environmental Surveillance	0.053			0.085			0.127		
Laundry	0.050			0.100			0.150		
Total Specialty Contractor Costs		3.103	11.5		10.185	18.9		18.277	23.3
Nuclear Insurance and License Fees		2.257	8.4		3.744	7.0		4.488	5.7
Subtotals		26.938	100.0		53.796	100.0		78.386	100.0
Contingency (25%)		6.735			13.449			19.597	
Total Costs		33.673			67.245			97.983	
Disposal of Accumulated Spent Fuel[g]		13.602			13.602			13.602	

(a) Costs are in early-1981 dollars. Number of significant figures shown is for computational accuracy only.
(b) Total costs are based on an assumed time period of 1.5 years for preparations for accident cleanup following the scenario 1 accident.
(c) Total costs are based on an assumed time period of 2.5 years for preparations for accident cleanup following the scenario 2 accident.
(d) Total costs are based on an assumed time period of 3 years for preparations for accident cleanup following the scenario 3 accident.
(e) Costs include contractor labor, materials, and overhead costs for the design and construction of the indicated items.
(f) Facilities include a warehouse-type building for onsite storage of drummed and boxed wastes and a facility for shielded storage of liners containing high-activity wastes.
(g) Costs of transportation to and 10-year storage at an ISFSI of accumulated spent fuel that is removed from the spent fuel pool during preparations for accident cleanup. These costs are assumed to be part of operating costs but are shown here for completeness. The fuel must be removed to make space available in the spent fuel pool for the filter/demineralizer system and for fuel from defueling the reactor following the accident.

These costs, including a 25% contingency, are \$33.7 million for preparations for cleanup following the scenario 1 accident, \$67.2 million for preparations for cleanup following the scenario 2 accident, and \$98.0 million for preparations for cleanup following the scenario 3 accident.

Costs of preparations for accident cleanup include the costs of keeping the reactor in a safe shutdown condition and of providing support services, as well as the costs of completing the activities summarized in Section 10.2 of Chapter 10. The major cost items in preparations for accident cleanup (staff labor and contractor labor costs, energy costs, and nuclear insurance and license fees) vary approximately linearly with the time required to complete the planning and preparations phase. Therefore, the total costs of preparations for cleanup following a particular accident are expected to vary approximately linearly with time.

As discussed in Chapter 10, the spent fuel pool is used to house the filter/demineralizer system that is used to process contaminated water and store fuel removed from the reactor during defueling operations following an accident. Therefore, the spent fuel that was accumulated in the pool during normal operations must be shipped offsite during preparations for cleanup to provide space in the pool for the filter/demineralizer system. The cost of shipment and offsite storage of this fuel is assumed to be an operating cost rather than an accident cleanup cost, since the fuel would eventually be shipped offsite if the reactor continued to operate. The cost of transportation and of 10-year storage of this fuel at an independent spent fuel storage installation (ISFSI), estimated to be about \$13.6 million, is shown as a line item in Table 11.1-1. The sensitivity of accident cleanup costs to the inability to ship this fuel is discussed in Section 11.6 and in Chapter 15.

11.1.1 Cost of Staff Labor

The cost of utility staff labor during preparations for accident cleanup is shown in detail in Table F.1-2 of Appendix F. Labor costs are based on utility staff labor requirements described in Section E.2 of Appendix E and include labor costs for keeping the reactor in a safe shutdown condition and for site support activities, as well as the costs of completing the planning and preparations activities described in Section E.2. Utility staff labor costs comprise approximately 50% of the total costs of preparations for accident cleanup.

Contractor labor costs to provide engineering support (i.e., to assist in the preparation of documentation for regulatory agencies, to prepare work plans and work schedules, and to design special tools and equipment) are shown as a separate line item under specialty contractor costs. Depending on the accident scenario, contractor labor costs contribute an estimated 10 to 20% to the total costs of preparations for accident cleanup.

11.1.2 Cost of Waste Management

The cost of waste management during preparations for accident cleanup includes the costs of packaging, shipment, and disposal of the radioactive wastes generated during this period and of fuel racks removed from the spent fuel pool. As discussed in Section 11.1, it is postulated that the removal of some or all of the existing fuel racks is necessary to make space available for the filter/demineralizer system used to process accident water and to provide space for new fuel racks that can accommodate canistered fuel. Waste management costs represent less than 1% of the total costs of preparations for accident cleanup. Details of these costs are given in Section F.1.2 of Appendix F.

11.1.3 Cost of Energy

Significant quantities of electrical energy are required to operate the essential systems and services that must remain in place to keep the reactor in a safe shutdown condition and to operate necessary support services during preparations for accident cleanup. The cold shutdown plant load at the reference PWR is about 22 MW.[1] This electricity usage rate is the basis for computing energy costs that represent between 20 and 30% of the total costs of preparations for accident cleanup.

11.1.4 Costs of Special Equipment and Facilities

Special equipment and facilities needed for accident cleanup in the containment building include:

- filter/demineralizer system
- fuel racks for canistered fuel (scenario 2 and scenario 3 accidents)
- storage tanks for processed water
- facilities for interim storage of wastes
- reactor vessel mockup (scenario 2 and scenario 3 accidents).

These special items are postulated to be designed, fabricated, and installed during preparations for accident cleanup. The bases and assumptions used to estimate the costs of these items are given in Section F.1.4 of Appendix F.

11.1.5 Costs of Miscellaneous Supplies

Miscellaneous supplies include small tools, protective clothing, replacement filters, clerical supplies, etc. A cost of $50,000 per year is used as the basis for estimating this cost item. Costs of miscellaneous supplies represent less than 1% of the total costs of preparations for accident cleanup.

11.1.6 Costs of Specialty Contractors

Major specialty contractor costs include the costs of engineering support, environmental monitoring, and laundry of protective clothing. The bases and assumptions used to estimate these costs are given in Section F.1.6 of Appendix F. Contractor costs for providing engineering support represent 11.1% of the costs of preparations for cleanup following the scenario 1 accident, 18.6% of the costs of preparations for cleanup following the scenario 2 accident, and 23.0% of the costs of preparations for cleanup following the scenario 3 accident.

11.1.7 Costs of Nuclear Insurance and License Fees

The costs of nuclear liability and property damage insurance and of license fees during preparations for accident cleanup are shown in detail in Table F.1-4 of Appendix F. These costs represent about 6 to 8% of the total costs of preparations for accident cleanup.

11.2 COST OF ACCIDENT CLEANUP IN THE AUXILIARY AND FUEL BUILDINGS

The estimated cost of accident cleanup in the auxiliary and fuel buildings of the reference PWR following the scenario 2 and scenario 3 accidents, including a 25% contingency, is approximately $19.5 million. Cost details are summarized in Table 11.2-1.

TABLE 11.2-1. Summary of Estimated Costs of Accident Cleanup in the Auxiliary and Fuel Buildings at the Reference PWR[a]

Cost Category		Estimated Cost[b] ($ millions)	Percent of Total
Cleanup Worker Labor		11.252	72.2
Waste Management		1.292	8.3
Special Tools and Equipment		0.285	1.8
Miscellaneous Supplies		1.435	9.2
Specialty Contractors			
Engineering	1.000		
Laundry	0.310		
Total Specialty Contractor Costs		1.310	8.5
Subtotal		15.574	100.0
Contingency (25%)		3.894	
Total Cost		19.468	

(a) Accident cleanup in the auxiliary and fuel buildings is assumed to be accomplished during preparations for cleanup in the containment building. Management and support staff costs and incidental costs are included in the costs of preparations for cleanup.
(b) Costs are in early-1981 dollars. Number of significant figures is for computational accuracy only.

Accident cleanup in the auxiliary and fuel buildings is postulated to take place during preparations for cleanup in the containment building. The accident cleanup costs shown in Table 11.2-1 include cleanup worker labor costs, waste management costs, costs of equipment and supplies, and specialty contractor costs specifically associated with accident cleanup in the auxiliary and fuel buildings. Management and support staff costs, costs of maintaining the reactor in a safe shutdown condition during this period, and incidental costs of energy, environmental surveillance, insurance, and taxes are included with the costs of preparations for accident cleanup shown in Table 11.1-1.

11.2.1 Cost of Staff Labor

Estimated cleanup worker labor costs for accident cleanup in the auxiliary and fuel buildings following the scenario 2 and scenario 3 accidents are shown in detail in Table F.2-2 of Appendix F. These costs are based on cleanup worker requirements described in Section E.3 of Appendix E. Cleanup workers are defined as those persons who actually complete the cleanup tasks in the auxiliary and fuel buildings and include staff members assigned to decontamination crews, a crew that provides construction and maintenance support, and waste processing and waste packaging crews. Cleanup worker labor costs represent about 72% of accident cleanup costs in the auxiliary and fuel buildings.

Contractor labor costs to provide engineering support are shown separately from cleanup worker labor costs in Table 11.2-1. Engineering support costs represent approximately 6.5% of accident cleanup costs.

11.2.2 Cost of Waste Management

The cost of management of the radioactive wastes from accident cleanup in the auxiliary and fuel buildings is shown in detail in Table F.2-3 of Appendix F. The waste disposal assumptions upon which these costs are based are discussed in Section 10.4.1.5 of Chapter 10. Waste management costs include packaging costs, transportation charges, and waste disposal charges at a shallow-land burial ground or a federal repository. These costs represent about 8.3% of the costs of auxiliary and fuel building cleanup.

High-activity wastes (filter cartridges and ion exchange materials) from processing contaminated liquids to remove the radioactivity are assumed to be transported to a federal repository for disposal. All other radioactive wastes are assumed to be transported to a shallow-land burial ground for disposal. Both the federal repository and the shallow-land burial ground are assumed to be located 1,600 km from the reactor site. All waste shipments are made by truck.

11.2.3 Costs of Special Tools and Equipment

Estimated costs of the special tools and equipment used during accident cleanup in the auxiliary and fuel buildings are shown in detail in Table F.2-4 of Appendix F. These costs represent about 2% of the total costs of auxiliary and fuel building cleanup.

11.2.4 Costs of Miscellaneous Supplies

Estimated costs of miscellaneous supplies for accident cleanup in the auxiliary and fuel buildings are shown in detail in Table F.2-5 of Appendix F. These costs represent about 9% of the total costs of auxiliary and fuel building cleanup.

11.2.5 Costs of Specialty Contractors

Major specialty contractor costs for accident cleanup in the auxiliary and fuel buildings include the costs of engineering support and laundry services. (The costs of transportation of radioactive wastes are included in waste management costs.) Specialty contractor costs are about 8.5% of the total costs of auxiliary and fuel building cleanup. The bases and assumptions used to estimate these costs are given in Section F.1.6 of Appendix F.

11.3 COST OF ACCIDENT CLEANUP IN THE CONTAINMENT BUILDING

The estimated costs of accident cleanup in the containment building at the reference PWR following the three postulated accidents are summarized in Table 11.3-1. These costs, including a 25% contingency, are $71.5 million following the scenario 1 accident, $137.2 million following the scenario 2 accident, and $287.0 million following the scenario 3 accident.

Costs of accident cleanup in the containment building include the costs of completing the cleanup activities described in Section E.4 of Appendix E as well as reactor operations and site support costs during the accident cleanup period.

11.3.1 Cost of Staff Labor

The cost of labor is the major cost item for accident cleanup in the containment building. Utility staff labor is estimated to account for 28 to 54%

TABLE 11.3-1. Summary of Estimated Costs of Accident Cleanup in the Containment Building at the Reference PWR

Cost Category	Accident Cleanup Following Scenario 1 Accident			Accident Cleanup Following Scenario 2 Accident			Accident Cleanup Following Scenario 3 Accident		
	Estimated Costs[a] ($ millions)		Percent of Total	Estimated Costs[a] ($ millions)		Percent of Total	Estimated Costs[a] ($ millions)		Percent of Total
Utility Staff Labor									
Management and Support Staff	5.880			12.992			29.847		
Plant Operations Staff	4.828			10.344			19.090		
Accident Cleanup Staff	5.085			20.715			69.413		
Per Diem During Defueling[b]	0.360			1.500			5.380		
Total Staff Labor Costs		16.153	28.2		45.551	41.5		123.730	53.9
Waste Management Costs									
Disposal by Shallow-Land Burial	0.864			1.655			6.276		
Disposal at Federal Repository	0.573			1.225			2.911		
Fuel and Fuel Core Debris	23.312			26.038			26.443		
Total Waste Management Costs		24.749	43.3		28.918	26.4		35.630	15.5
Energy		7.740	13.5		14.516	13.2		25.802	11.2
Special Tools and Equipment		3.025	5.3		6.250	5.7		13.650	5.9
Miscellaneous Supplies		1.486	2.6		3.753	3.4		6.950	3.0
Specialty Contractors									
Engineering	1.500			5.600			15.000		
Environmental Surveillance	0.063			0.118			0.212		
Waste Evaporator System	0.050			0.100			0.200		
Laundry	0.225			0.450			0.950		
Total Specialty Contractor Costs		1.838	3.2		6.268	5.7		16.362	7.1
Nuclear Insurance and License Fees		2.231	3.9		4.462	4.1		7.438	3.2
Subtotals		57.222	100.0		109.718	100.0		229.562	99.8[c]
Contingency (25%)		14.306			27.430			57.391	
Total Costs		71.528			137.148			286.953	

(a) Costs are in early-1981 dollars. Number of significant figures shown is for computational accuracy only.
(b) Per diem paid to crew leaders and utility operators temporarily assigned from other plants during defueling operations. See explanation in Section E.4.2 of Appendix E.
(c) Total does not equal 100% because individual percentages are rounded to the nearest one-tenth.

of the total cost of containment building cleanup, depending on accident scenario. Staff labor costs are based on the requirements for accident cleanup in the containment building that are described in Section E.4 of Appendix E and are shown in detail in Table F.3-2 of Appendix F. Staff labor requirements include site support and plant operations staff as well as personnel actually involved in cleanup of the containment building.

In this study, it is assumed that all cleanup activities except engineering support are performed by utility staff labor. Contractor costs for providing engineering support during accident cleanup are shown separately from utility staff labor costs in Table 11.3-1. These engineering support staff costs are estimated to represent approximately 3 to 6% of accident cleanup costs, depending on accident scenario. The sensitivity of cost estimates to having a contractor perform all of the accident cleanup activities in the containment building (exclusive of management, site support, and plant operations activities that would be performed by utility staff) is discussed in Section 11.6.

A staff labor cost not shown in Table F.3-2 but included in the total labor costs in Table 11.3-1 is the living allowance paid to crew leaders and utility operators temporarily assigned from other reactor stations to assist in defueling operations at the damaged reactor. As explained in Section E.4.2 of Appendix E, a large number of trained personnel not normally available at the accident-damaged reactor station are needed for defueling the reactor. Personnel on temporary assignment from other stations to assist in defueling operations are assumed to be paid a living allowance of $2000 per month in addition to their regular salaries.

For accident cleanup following the scenario 1 accident, cleanup personnel account for only about 34% of the total staff labor costs, with site support and plant operations personnel accounting for the remainder of the staff labor costs. For cleanup following the scenario 2 accident, cleanup personnel account for about 49% of the total staff labor costs; and for cleanup following the scenario 3 accident, cleanup personnel account for over 60% of the total staff labor costs. Major factors that affect the increasing contribution of cleanup staff to total utility staff labor costs with increasing accident severity are: 1) the increase in the labor requirement for defueling the reactor, and 2) the additional cleanup manpower required to assure compliance with occupational dose limitations.

11.3.2 Cost of Waste Management

Based on the waste management disposal assumptions discussed in Appendix E, the costs of management of the radioactive wastes from accident cleanup in the containment building are shown in detail in Tables F.3-3, F.3-4, and F.3-5 of Appendix F. These costs include container costs, transportation, and disposal costs. Waste management costs represent about 43% of accident cleanup costs following the scenario 1 accident, about 26% of accident cleanup costs following the scenario 2 accident, and about 16% of accident cleanup costs following the scenario 3 accident.

High-activity wastes (filter cartridges, ion exchange resin liners, and solidified evaporator bottoms) from processing radioactive liquids and fuel assemblies removed from the reactor during defueling operations are postulated in this study to be transported to a federal repository. (Undamaged assemblies from defueling following the scenario 1 accident are postulated to be transported to an ISFSI.) All other radioactive wastes are postulated to be shipped to a shallow-land burial ground for disposal. The federal repository, the ISFSI, and the shallow-land burial ground are all assumed to be located 1600 km from the reactor site.

Volumes and costs of management of the radioactive wastes from accident cleanup in the containment building are summarized in Table 11.3-2. A comparison of these volumes and costs shows that most of the cost of waste management is for the relatively small volume of waste that is shipped to a federal repository. The major cost item for wastes shipped to a federal repository is for the disposal of the reactor fuel from defueling following an accident.

11.3.3 Cost of Energy

Significant quantities of electrical energy are required to operate the essential systems and services and the pumps and motors needed during accident

TABLE 11.3-2. Summary of Waste Management Costs for Accident Cleanup in the Containment Building[(a,b)]

Disposal Option	Cleanup Following Scenario 1 Accident: Waste Volumes (m^3)	Scenario 1: Waste Volumes Percent of Total	Scenario 1: Waste Management Costs[(c)] ($ million)	Scenario 1: Costs Percent of Total	Cleanup Following Scenario 2 Accident: Waste Volumes (m^3)	Scenario 2: Waste Volumes Percent of Total	Scenario 2: Waste Management Costs[(c)] ($ millions)	Scenario 2: Costs Percent of Total	Cleanup Following Scenario 3 Accident: Waste Volumes (m^3)	Scenario 3: Waste Volumes Percent of Total	Scenario 3: Waste Management Costs[(c)] ($ millions)	Scenario 3: Costs Percent of Total
Shallow-Land Burial	931	90.3	1.080	3.5	2005	92.4	2.069	5.7	4737	94.2	7.845	17.6
Federal Repository	56	5.4	4.055	13.1	164	7.6	34.079	94.3	292	5.8	36.693	82.4
ISFSI[(d)]	44	4.3	25.800	83.4								
Totals	1031	100.0	30.935	100.0	2169	100.0	36.148	100.0	5029	100.0	44.538	100.0

(a) Based on waste management assumptions discussed in Appendix E.
(b) Total costs include packaging costs, transportation charges, and disposal costs.
(c) Costs are in early-1981 dollars and include 25% contingency.
(d) Undamaged fuel from reactor defueling following the scenario 1 accident is shipped to an ISFSI.

cleanup in the containment building. Costs of electrical energy represent about 12% of the costs of containment building cleanup following the postulated accidents.

The bases and assumptions used to calculate electric energy costs for accident cleanup in the containment building are given in Section F.3.3 of Appendix F.

11.3.4 Costs of Special Tools and Equipment

The estimated costs of the special tools and equipment used during accident cleanup in the containment building are shown in detail in Table F.3-4 of Appendix F. For accident cleanup following the scenario 2 and scenario 3 accidents, these costs include research and development and fabrication costs for equipment to remove damaged fuel from the reactor and package the fuel in canisters prior to storage in the spent fuel storage pool. The costs of special tools and equipment represent about 5 to 6% of the costs of accident cleanup, depending on accident scenario.

11.3.5 Costs of Miscellaneous Supplies

The estimated costs of miscellaneous supplies for accident cleanup in the containment building are shown in detail in Table F.3-7 of Appendix F. These costs represent about 3% of the costs of accident cleanup following the postulated accidents.

11.3.6 Costs of Specialty Contractors

Major specialty contractor costs for accident cleanup in the containment building include the costs of engineering support, environmental surveillance, rental of an evaporator system for processing decontamination solutions, and laundry services. (The cost of transportation of radioactive wastes is included in waste management costs.) Specialty contractor costs represent about 3% of the costs of accident cleanup following the scenario 1 accident, about 6% of the costs of accident cleanup following the scenario 2 accident, and about 7% of the costs of accident cleanup following the scenario 3 accident. The bases and assumptions used to estimate these costs are given in Section F.1.6 of Appendix F.

11.3.7 Costs of Nuclear Insurance and License Fees

The costs of nuclear liability insurance and of license fees during accident cleanup in the containment building are shown in detail in Table F.3-8 of Appendix F. These costs represent about 4% of the total cost of accident cleanup in the containment building.

11.4 SUMMARY OF ACCIDENT CLEANUP COSTS FOR THE REFERENCE PWR

Based on the assumptions listed as key study bases in Section 2.2 of Chapter 2, the cleanup activities described in Chapter 10, and the costs of accident cleanup summarized in Sections 11.1 through 11.3, the total estimated costs and estimated time requirements for accident cleanup at the reference PWR are shown in Table 11.4-1. Accident cleanup following the scenario 1 accident is estimated to cost about $105 million and to require 3.0 years for completion. Accident cleanup following the scenario 2 accident is estimated to cost about $224 million and to require 5.3 years for completion. Accident cleanup following the scenario 3 accident is estimated to cost about $404 million and to require 8.0 years for completion. These costs and times include those for planning and preparation as well as the actual costs and times for cleanup operations.

TABLE 11.4-1. Summary of Time and Cost Estimates for Accident Cleanup at the Reference PWR Following the Postulated Accidents

	Cleanup Following Scenario 1 Accident		Cleanup Following Scenario 2 Accident		Cleanup Following Scenario 3 Accident	
	Time (years)	Cost ($ millions)[a]	Time (years)	Cost ($ millions)[a]	Time (years)	Cost ($ millions)[a]
Preparations for Accident Cleanup	1.5	33.7	2.5	67.2	3.0	98.0
Accident Cleanup in Auxiliary and Fuel Buildings	--[b]	--[b]	--[c]	19.5[d]	--[c]	19.5[d]
Accident Cleanup in Containment Building	1.5	71.5	2.8	137.1	5.0	287.0
Totals	3.0	105.2	5.3	223.8	8.0	404.5

(a) Costs are in early-1981 dollars and include 25% contingency.
(b) Accident cleanup in the auxiliary and fuel buildings is not postulated following the scenario 1 accident.
(c) Accident cleanup in the auxiliary and fuel buildings is postulated to be completed during preparations for cleanup in the containment building.
(d) Includes the costs of cleanup worker labor, waste management, equipment, supplies, and services for accident cleanup in the auxiliary and fuel buildings. Management and support staff costs and incidental costs (e.g., energy, insurance, etc.) are included in the costs of preparations for accident cleanup.

11.5 COMPARISON OF ESTIMATED COSTS OF ACCIDENT CLEANUP AT THE REFERENCE PWR WITH CLEANUP COSTS AT TMI-2

The accident at Three Mile Island Unit 2 (TMI-2) on 28 March 1979 is the only major accident involving extensive cleanup that has occurred at a large power reactor. Hence comparisons between the estimated costs for accident cleanup of the reference PWR presented in this study and the costs estimated for cleanup of TMI-2 are useful. The comparisons also illustrate the sensitivity of accident cleanup costs to certain plant- and situation-specific factors discussed in more detail in Section 11.6. This section is not intended to be a critique of this study or of estimates made of the costs of cleanup at TMI-2. Rather it is intended to demonstrate the reasonableness of the cost estimates of this study of a hypothetical reference situation compared to a real situation.

Considerations of contamination levels and cleanup activities to be used during accident cleanup at TMI-2 indicate that TMI-2 contains some characteristics of the scenario 2 and scenario 3 accident cases of this study.

The estimated costs of accident cleanup at the reference PWR are summarized in Section 11.4. Various estimates of the costs of cleanup and recovery at TMI-2 have been made. The TMI-2 recovery program cost estimate made by General Public Utilities (GPU) Corporation in July 1981[2] puts this cost at about $1 billion over 8 years of cleanup. The differences in the accident cleanup cost estimates for the reference PWR and for TMI-2 are discussed in the following paragraphs.

Estimated costs for accident cleanup at the reference PWR and for cleanup and recovery at TMI-2 are shown by cost category in Table 11.5-1. The cost categories in the table are those used in Reference 2 to summarize TMI-2 cleanup costs. Different cost categories have been used in this study to develop the accident cleanup cost estimates for the reference PWR (see, for example, Tables 11.1-1 and 11.3-1). To facilitate comparisons with TMI-2 costs, the accident cleanup costs for the reference PWR are shown in the TMI-2 cost format. Uncertainties about how some costs should be assigned may have resulted in some reference PWR costs being shown in a different cost category than that used for similar TMI-2 costs, but, in general, this table is considered to be a good comparison of the two cost estimates.

There are several reasons why the cost estimate for cleanup at TMI-2 is larger than cost estimates made in this study for accident cleanup at the reference PWR. Some of these reasons are discussed below.

1. Certain items are included in the TMI-2 estimate that are not included in the reference PWR cleanup estimate. These items include the cost of base operations and maintenance (Item E in Table 11.5-1) and an allowance for escalation of costs due to inflation during the cleanup period (Item F in Table 11.5-1). The costs presented in this study are in constant, early-1981 dollars, and hence would have to be adjusted to include the legitimate effects of cost escalation during the cleanup period. Taken together, the costs of Item E (approximately \$124 million) and Item F (approximately \$209 million) amount to about \$333 million. The net estimated cost in constant dollars of the cleanup activities at TMI-2, not including base operations and maintenance, is about \$700 million.

2. The TMI-2 cost estimate includes the legitimate cost of several facilities necessary to facilitate the decontamination of TMI-2 but which, based on the design and plant layout of the reference PWR, are not assumed to be needed for the decontamination of the reference plant. Hence costs of these proposed facilities for TMI-2 are not included in the cost estimates for accident cleanup at the reference PWR. They are shown, however, as part of the "additional facilities for RCB decontamination" (Item C.2 in Table 11.5-1) and include a hot chemistry laboratory, containment recovery service building, and command center/temporary personal access facility costing approximately \$84 million.

3. The TMI-2 accident cleanup cost estimate includes a cost for additional decontamination of the containment building, some of which takes place after the defueling of the reactor (Item C.6 in Table 11.5-1). This decontamination would be beyond the gross

TABLE 11.5-1. Comparison of TMI-2 Cleanup Costs with Estimated Costs of Accident Cleanup at the Reference PWR

Cost Item	TMI-2 Cleanup Costs[a]		Accident Cleanup in Reference PWR Following Scenario 2 Accident[b]		Accident Cleanup in Reference PWR Following Scenario 3 Accident[b]	
	Costs ($ thousands)					
A. Maintain Plant in Safe Condition						
1. Operation - Fueled Plant	41 101		58 194		88 555	
2. Site Support Services - Fueled Plant	21 015		27 921		44 692	
3. Operation - Defueled Plant	30 166		4 841		10 222	
4. Site Support Services - Defueled Plant	12 535		2 609		5 859	
Subtotals		104 817		93 565		149 328
B. Auxiliary Building Decontamination		15 766		19 468		19 468
C. Defuel Reactor & Decontamination of Containment Building						
1. Containment Building & RCS Water Cleanup	25 203		15 369		35 841	
2. Additional Facilities for RCB Decontamination	83 985		5 688		12 062	
3. Gross Decontamination of RCB	62 712		22 839		49 256	
4. RPV Head and Core Removal	63 147		66 411		137 500	
5. Facilities to House Contaminated Equipment & Material	8 816		455		1 019	
6. Additional Decontamination of RCB	110 451		0		0	
Subtotals		354 314		110 762		235 678
D. Costs Expended During First 2-1/2 Years Following TMI-2 Accident		226 000				
E. Base Operations and Maintenance						
1. Expended to Date	49 104					
2. Estimated - Future Years	75 000					
Subtotals		124 104				
F. Cost Escalation		209 325				
Total Estimated Costs		1 034 326		223 795		404 474

(a) Costs are from Reference 2 and are in 1980 dollars.
(b) Costs are from Appendix F, are in early-1981 dollars, and include a 25% contingency.

decontamination of the containment building (Item C.3 in Table 11.5-1) necessary to permit reactor defueling activities to proceed with modest radiation exposure to workers. This study defines accident cleanup to include those activities leading to and including the reactor defueling and reactor coolant system decontamination (see Section 10.1). Beyond that, this study defines further decontamination activities to fall under the category of decommissioning or of further cleanup leading to refurbishment (which is not covered in detail in this study). Hence the portion of the legitimate costs of decontamination following defueling listed in Item C.6, approximately $100 million, are not included as accident cleanup costs in this study. As defined in this study, they would either be included as part of the decommissioning costs discussed in Chapter 13 or be considered part of the costs of further cleanup leading to refurbishment. This discussion illustrates the need to carefully consider what activities are being included in a particular cost analysis.

4. Defueling costs are a function of the assumed damage to the fuel core and the difficulty of removing the fuel and packaging it for interim storage in the spent fuel pool. As shown in Table 11.5-1, Item C.4, the estimated cost of defueling following the scenario 2 accident is approximately the same as the estimated cost of defueling at TMI-2, whereas the estimated cost of defueling following the scenario 3 accident is about twice the TMI-2 cost. This illustrates the importance of uncertainties in plant condition in estimates of the cost of cleanup.

5. An additional item of difference in Table 11.5-1 is Item D, which is defined in Reference 2 as those costs incurred at TMI-2 between the time of the accident and the end of 1981, a period of about 2-1/2 years. Approximately $226 million in costs were incurred at TMI-2 during this period for activities such as stabilization of the plant, preparations for accident cleanup, and maintenance of the plant in a safe shutdown condition. This study does not estimate costs for

stabilization of the plant or for maintenance of the plant in a safe shutdown condition during delays. In this study, the costs of preparations for accident cleanup are estimated to be about $67 million during a 2-1/2-year period following the scenario 2 accident and about $98 million during a 3-year period following the scenario 3 accident. The costs of facility stabilization are not included in this study. This cost item illustrates that there may be legitimately high costs arising from unexpected delays. In a report prepared by the General Accounting Office[3] it was pointed out that significant costs are incurred simply to maintain the status quo during delays. It is indicated in that report that delays in proceeding with the cleanup of TMI-2 have resulted from financial difficulties and regulatory concerns.

6. There are various other cost differences in Table 11.5-1, each of which by itself may not be significant, but which can be large when totalled together. These differences in the estimated costs of completing the various technical tasks illustrate that there can be differences in the methodology of cleanup with resulting differences in cost.

When these above differences in cost are considered, the cost estimates made in this study of the reference plant appear to be reasonable estimates of the costs of accident cleanup.

11.6 SENSITIVITY OF ACCIDENT CLEANUP COSTS TO VARIOUS FACTORS

Accident cleanup is an activity that takes place at a time when conditions in a plant are uncertain and when social, political, financial, and regulatory constraints can affect the progress and costs of cleanup activities. In addition, the processing of accident-generated wastes and the defueling of the reactor may require the use of specialized procedures and techniques. The sensitivity of accident cleanup costs to various factors is addressed in this section. Some factors that can influence accident cleanup costs include:

- the potential for delays in accident cleanup due to various causes such as greater core damage or contamination than expected, requirements for the design and construction of specialized systems for processing wastes or defueling the reactor, social or political constraints, regulatory concerns, financial difficulties, etc.

- the need to use more complicated and expensive processing systems, if they are required, for such activities as containment purge or waste solidification

- the temporary inability to dispose of radioactive wastes offsite due to technical, political, or regulatory constraints

- the need to construct buildings to house special equipment items such as the filter/demineralizer systems

- a requirement to use outside contractors to complete certain cleanup tasks because sufficient utility staff are not available.

In addition, considerations related to plant design may result in a need for special buildings and equipment for accident cleanup that may be different for different reactor facilities. As discussed in Section 11.5, TMI-2 required approximately $80 million of additional facilities that were not employed in this study due to differences in design characteristics between TMI-2 and the reference PWR.

Estimates of the sensitivity of accident cleanup costs to the various factors listed above are given in the following subsections.

11.6.1 Sensitivity of Costs to Delays in Completing Accident Cleanup

Delays in accident cleanup can result in increased costs in several ways. One is the added costs that are incurred in paying staff for additional time onsite. Another is the added costs of such items as energy and insurance. A third is the cost of additional processing equipment that may be required. It is beyond the scope of this report to analyze all of the social or political constraints or other regulatory or financial problems that could cause delays. The study assumes the existence of delays and estimates the associated costs. Estimates of the added costs resulting from a 1-year delay in either preparations for accident cleanup or the actual accident cleanup activities at the reference PWR are given in Table 11.6-1. Since the added costs of delays are expected to vary approximately linearly with length of the delay period, the added costs of delays for shorter or longer time periods can be inferred from the table.

11.6.2 Sensitivity of Costs to the Need to Use Special Processing Systems or Equipment

This study uses as bases certain assumptions concerning the capability of the licensee to employ specific methods for disposing of gaseous, liquid, and solid accident wastes. For example, it is assumed for purposes of estimating the costs of accident cleanup that krypton can be removed from the containment building atmosphere by purging. It is also assumed that processed water can

TABLE 11.6-1. Estimated Added Costs of Delays in Completing Accident Cleanup at the Reference PWR

Parameter	Value Following Scenario 1 Accident	Value Following Scenario 2 Accident	Value Following Scenario 3 Accident
Preparations for Accident Cleanup			
Reference Time[(a)]	1.5 years	2.5 years	3.0 years
Reference Cost[(a)]	\$33.7 million	\$67.2 million	\$98.0 million
Added Cost of 1-yr Delay	\$20 million	\$22 million	\$25 million
Accident Cleanup in the Containment Building			
Reference Time[(a)]	1.5 years	2.8 years	5.0 years
Reference Cost[(a)]	\$71.5 million	\$137.1 million	\$287.0 million
Added Cost of 1-yr Delay	\$21 million	\$26 million	\$32 million

(a) Reference times and costs are taken from Table 11.4-1.

be discharged to the river when processing has reduced contamination levels below the limits set by regulatory requirements. If, due to technical, regulatory, or political constraints, alternative, more complicated treatment or disposal methods are required, these can also add to the cost.

Alternatives for the removal of krypton from the containment building atmosphere are described in Section E.2 of Appendix E. These alternatives include:

- selective absorption
- charcoal adsorption
- gas compression and storage
- cryogenic processing.

Each of these alternatives requires the construction of special equipment or facilities that could add significantly to the time and cost requirements of preparations for accident cleanup. Estimates of the time required to implement each alternative, had an alternative other than purging been chosen for the removal of radioactive krypton from the containment building atmosphere at TMI-2, are given in an environmental assessment evaluating the alternative for ^{85}Kr removal from the TMI-2 reactor building.[4]

Estimates of the times and added costs of using an alternative other than purging for krypton removal at the reference PWR are shown in Table 11.6-2. Time estimates are based on information from the TMI-2 environmental assessment.[4] Costs include the costs of delays in preparations for cleanup as well as the estimated costs of additional processing equipment.

TABLE 11.6-2. Estimated Added Times and Added Costs of Preparations for Accident Cleanup at the Reference PWR Due to Use of an Alternative for ^{85}Kr Removal from the Containment Building

^{85}Kr Removal Alternative	Estimated Added Time (years)	Estimated Added Cost ($ millions)
Selective Absorption	1.0	25
Charcoal Adsorption	2.5	60
Gas Compression & Storage	2.5	60
Cryogenic Processing	2.0	50

Alternatives for the disposal of processed water are discussed in Section D.4.3 of Appendix D. These alternatives include onsite storage in tanks and immobilization of the processed water in cement followed by disposal at a shallow-land burial facility.

Onsite storage would be a temporary measure to allow tritium to decay to concentrations compatible with primary drinking water standards. A 1000-fold reduction in the tritium radioactivity would require storage for a period of approximately 130 years. The estimated cost of construction of a 1000-m^3 carbon steel tank is about $150,000. The cost of surveillance and maintenance of the tank is not estimated in this study. It should be noted, however, that a requirement for onsite storage of the processed water could affect the choice of decommissioning alternative and the ability to release the facility and site for unrestricted use.

The offsite disposal alternative would require that processed water be immobilized in cement, packaged, and transported to a low-level waste disposal facility. Packaging of this water in 0.21-m^3 steel drums could result in a very large number of drums. For example, immobilization of the effluent from treatment of the accident water and the water-based decontamination solutions from the scenario 3 PWR accident could result in a requirement for immobilization of about 2500 m^3 of processed water. This would require almost twenty-one thousand 0.21-m^3 steel drums (assuming 0.12 m^3 of water per drum). The total cost of this alternative, including container costs, transportation to a shallow-land burial ground, and disposal charges, is estimated to be about $2.6 million.

11.6.3 Sensitivity of Costs to the Temporary Inability to Dispose of Wastes Offsite

This study uses as a basis the assumption that solid wastes can be disposed of offsite, either at shallow-land burial grounds for certain of the wastes, or at federal repositories for other types of wastes. If these facilities are unavailable or if social, political, regulatory, or other constraints exist that prohibit the shipment of wastes to these facilities, it may be necessary for certain or all of these wastes to be temporarily stored onsite for an extended time period. A requirement for temporary onsite storage of wastes could affect the choice of decommissioning alternative and increase the costs of accident cleanup and of the decommissioning that follows accident cleanup.

The impacts of a temporary inability to dispose of wastes offsite are discussed in detail in Chapter 15. A conclusion of that chapter is that only SAFSTOR or partial DECON appear to be practical at the reference PWR if temporary onsite storage of accident cleanup and decommissioning wastes or of spent fuel were to be necessary. Onsite storage of low-level waste is estimated to have virtually no effect on the total cost (in constant dollars) of accident cleanup and SAFSTOR. However, some costs that normally occur prior to the safe storage period are delayed until deferred decontamination. Onsite storage of process solid wastes and of spent fuel increases the total cost of accident cleanup and decommissioning by about 5 to 10%, depending on which wastes are stored onsite and on the decommissioning alternative chosen.

As indicated in Chapter 15, the costs directly associated with interim storage of spent fuel are considered operational, and hence are not included in this study as decommissioning costs. These costs include the security force made necessary by the presence of the spent fuel, the costs of operating personnel, normal maintenance, energy, equipment, supplies, insurance, and license fees. These costs are estimated in Chapter 15 to be approximately $1 million per year of storage. If there were additional design or maintenance problems associated with the storage of the spent fuel, these would add to the operational cost of storage. These activities and their costs are outside the scope of this study.

11.6.4 Sensitivity of Costs to the Need to Construct Special Buildings

This study uses as a basis the assumption that the filter/demineralizer system used to treat accident water is installed in the spent fuel pool and that other processing equipment and tanks are housed in existing structures. If circumstances (such as the need to use the spent fuel pool exclusively for the storage of fuel) prevent this, it may be necessary to construct a new building to house the filter/demineralizer system and associated tanks and other processing equipment. The cost of this building is estimated to be about $0.5 million and its construction is estimated to add about 6 months to the time requirement for preparations for accident cleanup. Thus, the total impact on accident cleanup costs is estimated to be about $12 to $13 million.

11.6.5 Sensitivity of Costs to the Use of Contractors to Accomplish the Cleanup Activities

A basic assumption of this study, used to estimate labor costs, is that all of the activities associated with accident cleanup (including site support, plant operations, and the actual cleanup operations) are performed by utility staff labor. The only exception in this study is the use of contractor labor to provide engineering support for these activities. It may be that the utility operator may not have sufficient staff available to perform the actual cleanup operations following a reactor accident. The sensitivity of cost estimates to using a contractor for containment building cleanup (exclusive of site support and plant operations functions) is considered in this section.

The effect of contractual arrangements on decommissioning costs is discussed in an addendum[5] to the earlier study of decommissioning the reference PWR following normal shutdown. To estimate the effects of contractual considerations, it is necessary to make several assumptions about how selected costs are paid. It is assumed that the utility pays the radioactive materials transportation and disposal costs directly, as well as the costs of energy and of insurance and license fees. The costs of equipment and supplies (including shipping containers), as well as the costs of contractor staff labor, are assumed to be subject to a fee percentage increment that is anticipated to be in the range of 10 to 15%. Overhead rates applied to direct staff labor are expected to be significantly higher for subcontracting organizations than for operating utilities, because of the larger ratio of supervisory and support personnel to direct labor that usually exists in subcontracting organizations. Having personnel in the field rather than in the home office also increases the overhead costs, because of travel and living expenses for many of these personnel. Thus, in Reference 5, an overhead rate of 110% on direct staff labor is assumed to be applicable to all subcontractor personnel. Finally, there are significant mobilization and demobilization costs associated with having a contractor establish his presence on the site. For decommissioning of the reference PWR following normal shutdown, these mobilization/demobilization costs are estimated to be about $1.25 million.

Based on the contractual considerations discussed in the preceding paragraph, the added costs of using a contractor to perform the accident cleanup in the containment building of the reference PWR (exclusive of site support, plant operations, and certain managerial activities) are estimated to be about $10 million for cleanup following the scenario 1 accident, about $25 million for cleanup following the scenario 2 accident, and about $60 million for cleanup following the scenario 3 accident.

11.6.6 Other Potential Post-Accident Costs

There are other post-accident costs that are outside the scope of this study. These include the costs of replacement power, costs incurred during the stabilization period, and costs of any activities leading to refurbishment of a plant if it is decided to restart the plant.

The cost to stabilize the plant would include the costs of steps taken after the accident to recover key areas of the plant and to isolate and contain contamination resulting from the accident until cleanup facilities are available. This might include temporary construction of certain facilities, portable radwaste systems, addition of emergency ventilation and filtration systems, system isolation provisions, etc. Because the nature of these activities is so dependent on the specific reactor facility and accident situation, they are not studied in detail here. However, the costs of stabilization of the facility, including the labor involved, could be significant.

The costs of refurbishment of a facility for restart are not included in this study. Although it is considered that accident cleanup activities (and therefore costs) are relatively independent of whether the reactor is ultimately decommissioned or refurbished, the period following defueling and reactor coolant system decontamination would result in either decommissioning or additional cleanup leading to refurbishment. The legitimate costs of additional cleanup leading to refurbishment could be significant. However, they are beyond the scope of this study and are not included here.

11.6.7 Effect of Plant Size on Accident Cleanup Costs

A detailed analysis of the effect of nuclear plant size (in terms of operating power level) on the cost of accident cleanup following a serious accident is outside the scope of this study. Many individual cost items contribute to the total cost of accident cleanup. To determine the effect of plant size on accident cleanup costs, it is necessary to evaluate the effect of plant size on each cost item. Complex relations may exist between plant size and these individual items, and factors other than plant size can influence accident cleanup costs. Examples include the following:

1. Site support and plant operations costs, which are a function of the work force required for these activities and of the time required for preparations for cleanup and for the actual cleanup activities. The personnel requirement for site support and plant operations is a function of plant size and of several other factors such as plant design, management philosophy, etc. As indicated previously in this section, the time requirement for accident cleanup is affected by requirements for the design and fabrication of specialized systems and equipment, social and political constraints, regulatory concerns, and financial difficulties as well as by plant size.
2. Contaminated liquid processing costs, which are largely determined by the liquid volumes requiring processing and by requirements for specialized processing systems and equipment. Contaminated liquid volumes may be a function of plant size as well as other factors such as plant design. Specialized equipment needs are largely a function of plant design and contamination levels rather than plant size.

3. Defueling costs, which are a function of core size, the extent of damage to the core, and the difficulty of performing defueling operations in a high-radiation environment. Defueling costs may have little to do with plant size.

On the basis of the above considerations, it is concluded that no simple relation exists between plant size and the total cost of accident cleanup.

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5. R. I. Smith and L. M. Polentz, Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station, NUREG/CR-0130 Addendum, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, August 1979.

CHAPTER 12

ACTIVITIES AND MANPOWER REQUIREMENTS FOR DECOMMISSIONING AT A REFERENCE PWR

The actual decommissioning of an accident-damaged LWR begins following completion of the accident cleanup activities. This chapter contains information concerning activities and manpower requirements for the post-accident decommissioning of the reference PWR via each of the three decommissioning alternatives: DECON, SAFSTOR, and ENTOMB. Selection of the decommissioning alternative to be used at an accident-damaged reactor is independent of the accident cleanup activities that precede the decommissioning. (Accident cleanup activities are discussed in Chapter 10.)

The information presented in this chapter is a summary of the detailed discussion of post-accident decommissioning presented in Appendix G of Volume 2. A comparison of decommissioning following a postulated reactor accident with decommissioning following normal reactor shutdown is provided in Section 12.1. The post-accident decommissioning of the reference PWR via the DECON, SAFSTOR, or ENTOMB alternatives is discussed in Sections 12.2, 12.3, and 12.4, respectively. The costs associated with these decommissioning activities are summarized in Chapter 13.

The post-accident decommissioning analyses in this study use the results of previous analyses of PWR decommissioning following normal shutdown,[1,2] with appropriate modifications as necessary to account for post-accident conditions. In addition, a conceptual analysis of post-accident decommissioning of TMI-2[3] provides useful background information. The decommissioning analyses in this study are based on the assumption that the reactor has experienced a scenario 2 accident; variations in decommissioning activities and requirements that would result from the other two accident scenarios are discussed where applicable. (The three accident scenarios considered in this study are described in Chapter 8.)

A basic assumption of the decommissioning analyses presented here is that all radioactive waste materials resulting from accident cleanup and from decommissioning are shipped offsite for disposal at the time of decommissioning. An analysis of the cost and safety impacts of alternate scenarios for waste disposal is presented in Chapter 8.

12.1 COMPARISON OF DECOMMISSIONING FOLLOWING AN ACCIDENT AND FOLLOWING NORMAL SHUTDOWN

Under normal circumstances, decommissioning of an LWR follows the orderly shutdown of the facility at the end of its planned operating life. However, conditions at an accident-damaged reactor are significantly different from normal, with increased levels of radioactive contamination in the major plant buildings, damage to the reactor core, and possible physical damage to plant equipment and services. Comparisons of decommissioning activities following normal shutdown with those following a reactor accident, for the three decommissioning alternatives, are presented in detail in Tables G.1-1 through G.1-3 of Appendix G. A summary description of the differences between normal-shutdown and post-accident decommissioning is presented in the following paragraphs.

It is assumed in this study that accident cleanup activities are completed prior to the start of the actual decommissioning. The tasks performed during accident cleanup are postulated to be independent of the alternative (DECON, SAFSTOR, or ENTOMB) chosen to complete the decommissioning. In carrying out accident cleanup, certain tasks that are part of normal-shutdown decommissioning are completed (e.g., reactor defueling, comprehensive radiation surveys of the facility, and decontamination of the reactor coolant system). In addition, significant portions of other such tasks are undertaken (e.g., removal and segmentation of reactor vessel internals, surface decontamination in the containment, and disposal of spent fuel storage racks).

Accident cleanup also results in certain new tasks that must be completed during decommissioning. These new tasks are the removal of new equipment for processing accident water and the decommissioning of the onsite storage structures specially constructed for the handling of accident-cleanup wastes.

(In the event that offsite disposal for accident cleanup wastes is not available at the time of decommissioning, the decommissioning of onsite waste storage structures will be deferred. See Chapter 15 for a discussion of alternate waste disposal scenarios.)

A number of decommissioning tasks are common to both post-accident and normal-shutdown decommissioning. However, the physical and radiological condition of the plant resulting from an accident leads to substantial qualitative changes in these tasks. The major change is that, following an accident, radiation doses to decommissioning workers are higher than those following normal shutdown because of the increased levels of contamination on equipment, piping, and structural surfaces. Because of limitations on the radiation doses that can be accumulated by individual workers,[4] the higher radiation levels result in the need for a substantially greater number of decommissioning workers.

Although the accident cleanup activities remove a large portion of the accident-generated contamination in the plant, accident severity does have some impact on decommissioning activities and manpower requirements. The primary impact is on the radiation dose rates to decommissioning workers, which increase with increasing accident severity. It should be noted that the impact of accident severity on decommissioning activities and manpower requirements is much less than the corresponding impact on accident-cleanup activities and manpower requirements.

12.2 DECON ACTIVITIES AND MANPOWER REQUIREMENTS

DECON is the decommissioning alternative that leads to the earliest release of the facility and site for unrestricted use and to the earliest termination of the facility's nuclear license. Compared to the other two decommissioning alternatives, DECON results in a greater occupational radiation dose and a greater cost in the first few years after completion of accident cleanup. Planning and preparation activities, decontamination and dismantlement activities, and the schedule and manpower requirements for post-accident DECON at the reference PWR are described in this section.

12.2.1 Planning and Preparation Activities

Post-accident DECON at the reference PWR is a complex undertaking. Consequently, the success of the project is highly dependent upon good planning and upon execution of necessary preparatory work prior to completion of the accident cleanup campaign that precedes decommissioning. Planning and preparation for DECON is assumed to be accomplished during the final 1-1/2 years of the accident cleanup campaign.

Planning and preparation activities for DECON at the reference PWR following normal reactor shutdown are discussed in Section 9.1 of Reference 1. Planning and preparation activities applicable to post-accident DECON can be summarized as follows:

- Satisfying regulatory requirements - primarily involves: 1) providing the necessary documentation for amending the facility operating license, and 2) obtaining an NRC dismantlement order. In addition, the licensee must submit a radioactive waste handling plan, a quality assurance plan, an environmental report, security and safeguards plans, and possibly updated information concerning the licensee's financial qualifications.
- Gathering and analyzing data - provides input to the documentation and establishes the bases for developing work plans and procedures. The bulk of the required data is assumed to be available as a result of the accident cleanup activities.
- Developing detailed work plans and procedures - provides the decommissioning staff with all the information required to actually carry out the decommissioning tasks. The plans and procedures cover all aspects of the project, and quality assurance, security, and environmental constraints are considered.
- Designing, procuring, and testing special equipment - ensures the availability and proper operation of the required equipment. The testing also serves to train personnel in the use of the equipment and to provide pertinent data on its operation.

- Selecting and training staff - ensures the availability of competent personnel and enables decommissioning to proceed smoothly, safely, and expeditiously.
- Selecting specialty contractors - allows certain specialized decommissioning tasks outside of the expertise or capability of the decommissioning staff to be performed by experts, increasing the overall efficiency and safety of the decommissioning project.

12.2.2 Decontamination and Dismantlement Activities

The decontamination and dismantlement activities during post-accident DECON at the reference PWR are very similar to those during DECON following normal shutdown, which are described in detail in Reference 1. The decontamination and dismantlement activities during post-accident DECON are summarized here, with emphasis on those activities that differ significantly from the ones during normal-shutdown DECON.

12.2.2.1 Containment Building

All of the neutron-activated materials and the majority of the radioactive contamination (both accident-generated and from normal operations) in the reference PWR at the time of decommissioning are located in the containment building. The neutron-activated components (the reactor vessel internals, together with portions of the reactor pressure vessel and the reactor cavity concrete) are segmented and are packaged primarily in steel cask liners for shipment to a shallow-land burial facility. Radioactively contaminated materials (consisting of equipment items, piping components, structural members, liner plates, concrete, etc.) are removed and cut up as required for packaging in steel drums or in plywood boxes. Methods postulated for removal of these materials during DECON following normal reactor shutdown are presented in Table G.1-1 of Reference 1, and these same methods are generally applied during post-accident DECON.

Following the postulated reactor accident and the subsequent accident cleanup campaign, radioactive contamination levels in the containment building exceed those that would be present following normal shutdown to an extent that depends on the severity of the accident and on the particular location in the building. Therefore, to reduce radiation doses to decommissioning workers to practicable levels, the major access routes used by the DECON workers and "hot spots" outside of the access routes that can materially affect worker doses are cleaned up or shielded. This task is undertaken at the start of DECON to obtain the maximum dose-reduction benefits, using the same methods as postulated for similar tasks during accident cleanup or during normal-shutdown DECON. The level of effort required for this task is a function of the amount of contamination present, which increases with increasing accident severity.

The methods used during post-accident DECON for removal and segmentation of the reactor vessel internals are the same as those used during normal-shutdown DECON. However, portions of the vessel internals are removed and segmented during accident cleanup to facilitate defueling of the reactor, reducing the level of effort required during DECON. (The extent of internals removal during cleanup is the same following both the scenario 1 and scenario 2 accidents, but increases following the scenario 3 accident.) On the other hand, additional difficulties may be encountered during DECON because of accident-caused damage to the internals and higher radiation exposure rates in the work area.

Chemical decontamination of the reactor coolant system is completed during the final stages of accident cleanup, and no further decontamination of this system is postulated to be required during DECON.

Decontamination of internal surfaces in the containment building is initiated during accident cleanup to reduce radiation doses to the cleanup workers. However, the bulk of this work is still carried out during DECON, particularly the removal of contaminated structural material. The methods used during post-accident DECON (i.e., concrete spalling, disassembly or cutting of metal components, etc.) are the same as those employed during DECON following normal reactor shutdown. However, accident-generated contamination results in a somewhat greater level of effort and a greater volume of radioactive waste material produced than following normal shutdown, increasing somewhat with increasing accident severity.

12.2.2.2 Fuel Building

The accident scenarios postulated for this study result in relatively limited impacts to the fuel building which are neutralized during the accident cleanup campaign. Therefore, changes in DECON activities in the fuel building following an accident are not the result of accident effects but rather are caused by use of fuel building facilities during accident cleanup. The DECON tasks required in the fuel building and the schedule for performing them do not vary substantially with accident severity, although occupational radiation doses to the workers increase with increasing accident severity because of increased contamination levels in certain areas of the building.

Chemical decontamination of the chemical volume control system is required during decommissioning. Following accident cleanup, however, this decontamination involves a somewhat lower level of effort than following normal shutdown because the system is decontaminated with the reactor coolant system during accident cleanup and only portions of the system are recontaminated during accident water processing.

Although some of the original spent fuel racks in the spent fuel pool are replaced with new, specially fabricated racks that can accommodate the fuel assemblies placed in canisters during defueling, removal and disposal of the spent fuel racks during post-accident DECON is not significantly different than during normal-shutdown decommissioning. However, the radiation doses accumulated by the workers are increased, to an extent dependent on the severity of the postulated accident.

During accident cleanup, the demineralizer system used to process the accident water is installed and operated in the spent fuel pool. Thus, a new requirement for post-accident DECON is the removal, segmentation, and disposal of this system. Furthermore, the use of the spent fuel pool is anticipated to result in greater than normal contamination levels in the pool, increasing radiation doses to workers engaged in activities in the vicinity of the pool.

12.2.2.3 Auxiliary Building

Any accident-caused impacts to the auxiliary building are postulated to be mitigated during accident cleanup. Therefore, post-accident DECON requirements in the auxiliary building are anticipated to be the same as those for DECON following normal reactor shutdown. However, radioactive contamination levels in the building are postulated to be higher than following normal shutdown, increasing with accident severity, resulting in corresponding increases in the radiation doses accumulated by the workers during DECON activities in the building.

12.2.2.4 Ancillaries

The ancillary activities for post-accident DECON at the reference PWR are described in the following paragraphs.

As discussed in Reference 1 for DECON following normal reactor shutdown, some limited decontamination in other site buildings (e.g., the condensate-demineralizer, control, and turbine buildings) is anticipated to be required. Because the postulated accident and the subsequent accident cleanup campaign are assumed to have no impact on these buildings, this decontamination activity for post-accident DECON is the same as for normal-shutdown DECON.

The packaging and shipping of radioactive wastes generated during DECON is handled by standing crews that are available over the entire duration of the tasks that generate the waste (i.e., until activities in the auxiliary building are completed). The amount and contamination levels of the wastes handled by these crews and, consequently, the radiation doses to these workers are anticipated to be greater than during normal-shutdown DECON and to increase with accident severity. Because the duration of the DECON effort varies only slightly with accident severity, the major factor affecting manpower requirements for this task is the limitation on radiation doses to individual workers.(4)

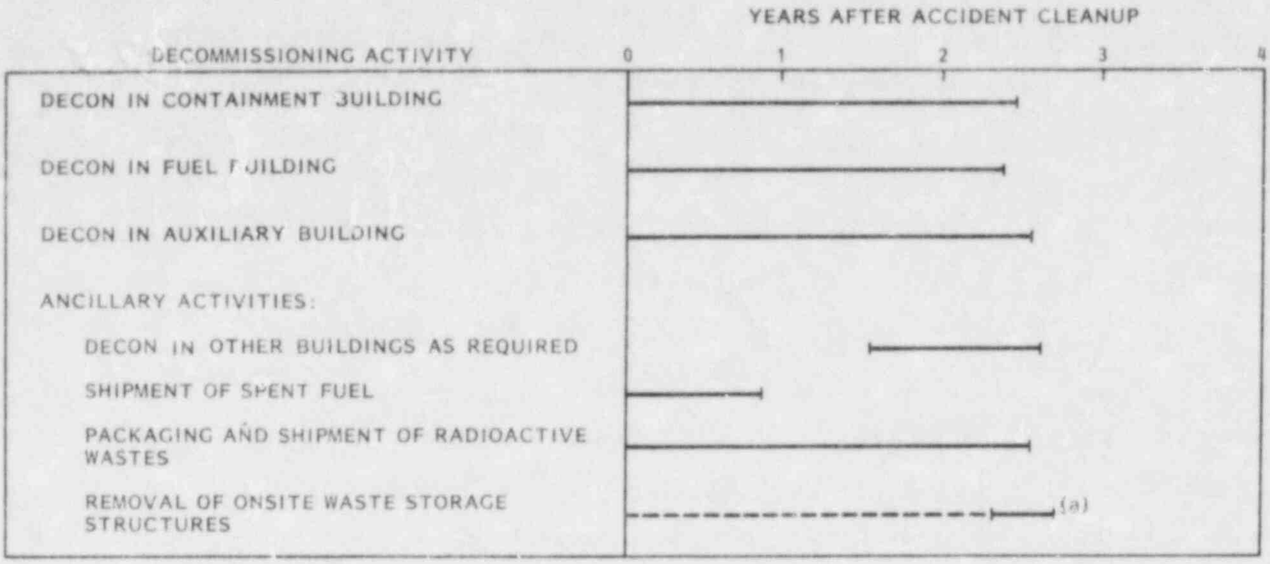
The costs of packaging, shipping, and disposal of the spent fuel are borne by accident cleanup, even though shipment of the fuel is carried out during DECON. This task is carried out on an intermittent basis by the waste handling crews described in the previous paragraph. Shipment of spent fuel from an operating reactor is a relatively routine procedure and, thus, this task poses no special difficulties during DECON. Because the spent fuel removed after a scenario 2 or scenario 3 accident is placed in canisters and, consequently, fewer fuel assemblies can be shipped in each cask, this task takes 40 to 60% longer to complete following either of these accidents, assuming availability of the same number of shipping casks. Radiation doses to the workers involved are higher than during normal-shutdown DECON and increase with increasing accident severity because of increased levels of contamination in the vicinity of the spent fuel pool.

The accident-cleanup wastes stored in the onsite waste storage structures constructed during accident cleanup are removed from the site and shipped to appropriate repositories before the completion of DECON. The waste handling crews described previously perform this task, but the costs for waste transport and disposal are borne by the accident cleanup campaign. After the wastes are removed, the onsite facilities are structurally decontaminated using the same methods employed in the major plant buildings. Because the amounts of accident-cleanup wastes stored onsite increase with increasing accident severity, the size of the structures and, consequently, the level of effort required to decommission them are also functions of accident severity.

12.2.3 DECON Schedule

The overall schedule and sequence of events for DECON at the reference PWR following a scenario 2 accident and the subsequent accident cleanup campaign is shown in Figure 12.2-1. Detailed schedules for DECON are presented in Section G.2 of Appendix G. Planning and preparation activities for DECON begin 1-1/2 years prior to the completion of the accident cleanup campaign that precedes DECON, as discussed previously in Section 12.2.1.

DECON begins in the containment building, which comprises the major effort for the decommissioning staff. The work proceeds through the fuel and auxiliary buildings as staff are available and as the various systems in these buildings complete their required service functions. The ancillary activities are performed on a schedule that depends on the need for the plant areas involved and on the availability of manpower. As shown in Figure 12.2-1, DECON following a scenario 2 accident is completed in 32-1/2 months. Variations in accident severity, within the range of accident scenarios considered in this study, are estimated to change the duration of DECON by no more than ±1 month.



(a) BROKEN LINE INDICATES OFFSITE SHIPMENT OF STORED WASTES, AND SOLID LINE INDICATES DECONTAMINATION OF STRUCTURES.

FIGURE 12.2-1. Overall Schedule and Sequence for DECON at the Reference PWR Following a Scenario 2 Accident

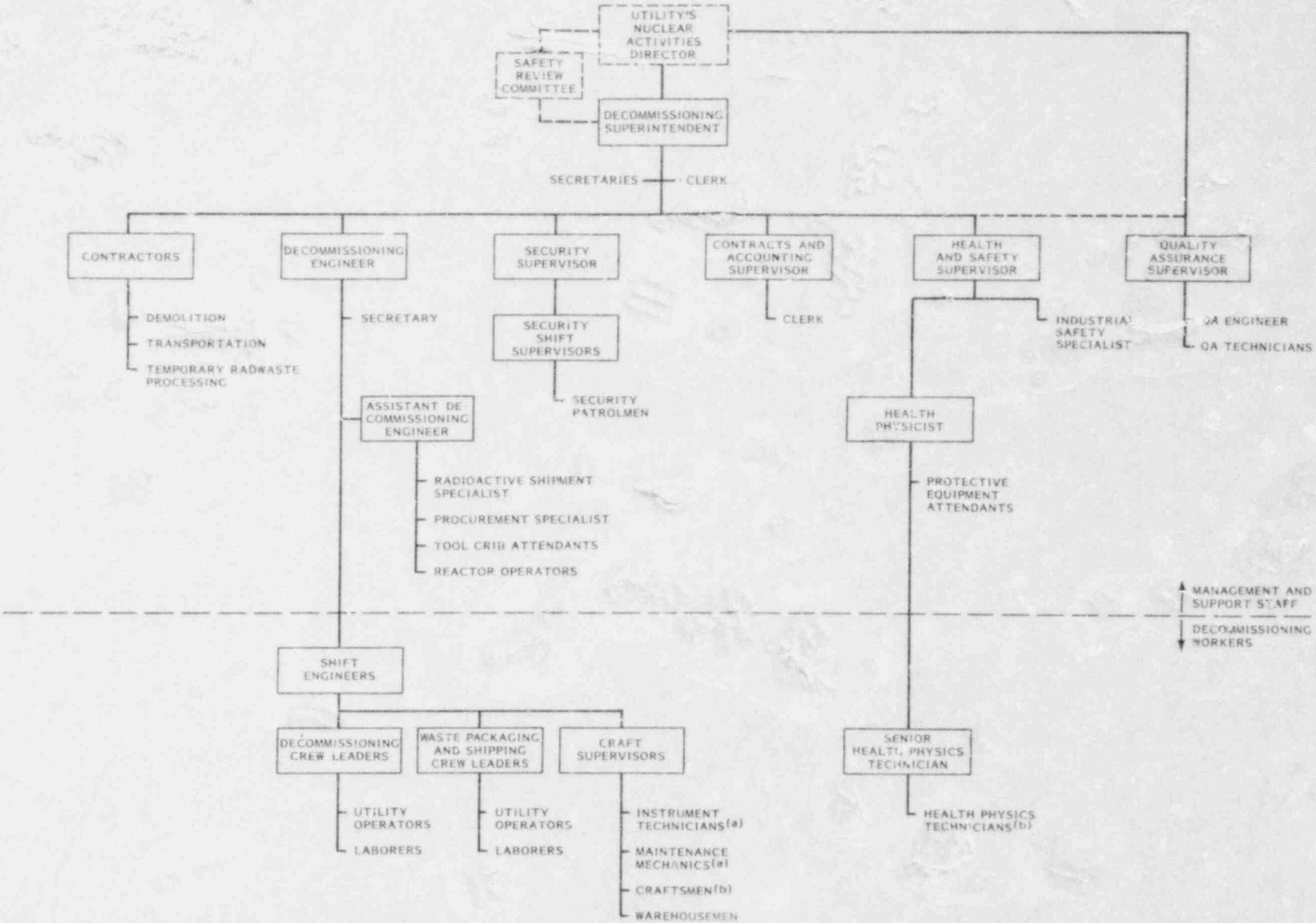
12.2.4 DECON Staff Requirements

In this subsection, the organization of the decommissioning staff and the types and numbers of decommissioning workers needed for post-accident DECON at the reference PWR are discussed.

12.2.4.1 Organization of the Decommissioning Staff

The staff for post-accident decommissioning of the reference PWR is organized as shown in Figure 12.2-2. Five parallel branches report to a decommissioning superintendent. The operational branch plans and performs the actual decommissioning tasks. The safety branch plans and conducts both radiological and industrial safety programs. The three auxiliary branches handle security, financial, and quality assurance matters.

The primary decommissioning activities are performed on a two-shift, 5-day-week basis. However, selected support activities (i.e., CVCS decontamination and radwaste system operation) and security functions are carried out



(a) CARRY OUT MAINTENANCE OF ESSENTIAL SYSTEMS AND SERVICES
(b) ASSIGNED TO WORK CREWS AS REQUIRED

FIGURE 12.2-2. Decommissioning Staff Organization on three shifts, around-the-clock, 7 days per week. In addition, the main control room is manned full time for operation of essential systems and services.

Further discussion pertaining to the staff organization and the functions of key staff members can be found in Chapter 9 of Reference 1.

12.2.4.2 DECON Manpower

Based on the detailed schedules for DECON activities in the various plant buildings, given in Section G.2 of Appendix G, the types and number of decommissioning workers needed to efficiently complete the radiation-zone work in the allotted time are determined. However, because whole-body radiation doses to the decommissioning workers are limited in accordance with 10 CFR 20.101,[4] the decommissioning worker requirements must be adjusted upward so that average individual radiation doses do not exceed 5 man-rem/man-year. For DECON at the reference PWR following a scenario 2 accident, decommissioning worker manpower requirements must be increased by factors of 3 to 5, as shown in Table G.2-3 of Appendix G.

Manpower requirements for management and support staff are primarily a function of the duration of the DECON project. The assumptions used to calculate management and support staff requirements are presented in Section G.2.4 of Appendix G.

Overall staff labor requirements for DECON at the reference PWR following a scenario 2 accident are given in Table 12.2-1. These requirements are given in equivalent man-years for the planning and preparation phase as well as for the actual decontamination and dismantlement, and include the management and support staff as well as the decommissioning workers. A total effort of about 790 man-years is estimated for completion of DECON following a scenario 2 accident.

Because management and support staff requirements are a function of project duration and the duration of the DECON project does not vary substantially with accident severity, requirements for the management and support staff are not estimated to vary with changes in accident scenario.

TABLE 12.2-1. Overall Staff Labor Requirements for DECON at the Reference PWR Following a Scenario 2 Accident

Position	Staff Labor Requirement (man-years) in Decommissioning Phase:[a] Planning and Preparation	DECON	Total Staff Labor Required (man-years)
Management and Support Staff			
Decommissioning Superintendent	1.5	3.0[b]	4.5
Secretary	3.0	8.5[b]	11.5
Clerk	1.0	5.4	6.4
Decommissioning Engineer	1.5	3.0[b]	4.5
Assistant Decommissioning Engineer	1.5	2.7	4.2
Radioactive Shipment Specialist	0	2.7	2.7
Procurement Specialist	0	2.7	2.7
Tool Crib Attendant	0	5.4	5.4
Reactor Operator[c]	0	21.7	21.7
Security Supervisor	0	2.7	2.7
Security Shift Supervisor	0	10.8	10.8
Security Patrolmen	0	28.2	28.2
Contracts and Accounting Supervisor	0	3.0[b]	3.0
Health and Safety Supervisor	0	3.0[b]	3.0
Health Physicist	0	2.7	2.7
Protective Equipment Attendant	0	5.4	5.4
Industrial Safety Specialist	0	2.7	2.7
Quality Assurance Supervisor	0	3.0[b]	3.0
Quality Assurance Engineer	0	2.7	2.7
Quality Assurance Technician	0	10.8	10.8
Consultant (Safety Review)	0	1.4	1.4
Instrument Technician[d]	0	10.8	10.8
Maintenance Mechanic[d]	0	10.8	10.8
Warehouseman	0	5.4	5.4
Subtotals	8.5	158.5	167.0
Decommissioning Workers			
Shift Engineer	0	5.4	5.4
Crew Leader[e]	0	45.5	45.5
Utility Operator[e]	0	114.1	114.1
Laborer[e]	0	194.4	194.4
Craft Supervisor	0	10.8	10.8
Craftsman[e]	0	142.8	142.8
Senior Health Physics Technician	0	16.3	16.3
Health Physics Technician[e]	0	94.4	94.4
Subtotals	0	623.7	623.7
Totals	8.5	782.2	790.7

(a) Rounded to the nearest 0.1 man-year.
(b) Includes an additional 4 months following active decommissioning to complete the documentation and other unspecified license and contract termination requirements.
(c) Based on two operators per shift in the control room, three shifts per day, 7 days per week.
(d) Based on one per shift, three shifts per day, 7 days per week to maintain essential services.
(e) From Table G.2-3.

However, scenario 2 decommissioning worker requirements could be reduced by a factor of about three following a scenario 1 accident and increased by a factor of about two to three following a scenario 3 accident, because of the estimated variations in the occupational radiation doses received by the decommissioning workers and the individual radiation dose limitations. Thus, the total staff labor requirements for DECON following a scenario 1 or a scenario 3 accident are estimated to be about 380 man-years or about 1730 man-years, respectively. For comparison, the total staff labor requirements for DECON at the reference PWR following normal reactor shutdown are about 300 man-years.(1)

12.3 SAFSTOR ACTIVITIES AND MANPOWER REQUIREMENTS

SAFSTOR as a decommissioning alternative satisfies the requirements for protection of the public while reducing, as compared to DECON, the initial commitments of time, money, occupational radiation dose, and waste disposal space. However, these reduced initial commitments are offset somewhat by the need to maintain the nuclear license, by the associated commitment to continuing care of the facility, and by the need for the eventual deferred decontamination of the facility. Furthermore, the decay of radioactive contamination within the stored facility following a postulated reactor accident is considerably slower than following normal reactor shutdown, because the decay of the post-accident radionuclide inventory is dominated by ^{137}Cs (with a half-life of about 30 years) rather than by ^{60}Co (with a half-life of about 5-1/4 years). In addition, deferral of decontamination for long periods of safe storage has the disadvantage that personnel familiar with the facility are no longer available to staff the final decontamination effort.

Planning and preparation activities, facility-preparation activities for safe storage, schedule and manpower estimates, and activities and requirements for continuing care and deferred decontamination are described in this section.

12.3.1 Planning and Preparation Activities

Successful implementation of SAFSTOR at the reference PWR is dependent upon good planning and upon execution of necessary preparatory work prior to completion of the accident cleanup campaign that precedes decommissioning.

Planning and preparation for SAFSTOR is assumed to be accomplished during the final 1-1/2 years of the accident cleanup campaign. The planning and preparation activities for SAFSTOR are essentially the same as those described in Section 12.2.1 for DECON and are not discussed further here.

12.3.2 Activities for Preparations for Safe Storage

The activities during post-accident preparations for safe storage at the reference PWR are generally the same as those during preparations for safe storage following normal reactor shutdown, described in detail in Reference 1. Activities during post-accident preparations for safe storage are summarized here.

In general, the activities during preparations for safe storage come under the following categories:

- decontamination, deactivation, and sealing of systems, equipment items, and plant areas
- fixation of surface contamination
- transfer of contaminated equipment and materials
- decontamination and isolation of contaminated plant areas.

The particular procedure used to decontaminate, deactivate, and seal each system or piece of equipment is identified during the planning phase of decommissioning. Portions of the facility containing significant amounts of radioactivity following appropriate decontamination efforts are isolated by tamper-proof barriers, affected systems are deactivated, and HEPA-filtered vents are installed to allow for temperature and pressure changes in these areas.

After the loose, readily removable contamination is removed from the surfaces of plant structures and equipment, the residual surface contamination is fixed in place to the maximum extent possible by spray painting the surfaces. During continuing care, these painted areas are monitored for deterioration and recoated as necessary.

Some contaminated equipment and other noncombustible radioactive materials are transferred within the plant from areas being decontaminated to other secured areas. Transferred items are spray painted to fix contamination, as are surfaces exposed by removal of the items.

The 13-point procedure postulated for preparing contaminated areas throughout the major plant structures is as follows:[3,5]

1. Evaluate initial radiological conditions.
2. Vacuum interior surfaces.
3. Deactivate nonessential systems and equipment.
4. Clean interior and exposed surfaces of equipment and piping.
5. Clean remaining hot spots.
6. Apply protective paint.
7. Transfer contaminated equipment and materials, where appropriate.
8. Decontaminate and seal vent systems.
9. Install HEPA-filtered vents.
10. Deactivate remaining nonessential systems and equipment.
11. Install security and monitoring systems and provide for servicing and offsite readout.
12. Conduct final radiation survey.
13. Secure the structure.

12.3.3 Schedule for Preparations for Safe Storage

The overall schedule and sequence of events for preparations for safe storage at the reference PWR following a scenario 2 accident and the subsequent accident cleanup campaign is shown in Figure 12.3-1, based on information presented in Section G.3 of Appendix G. Planning and preparation begins 1-1/2 years prior to the completion of the accident cleanup campaign that precedes decommissioning, as discussed previously in Section 12.3.1.

As with DECON, the preparations phase of SAFSTOR begins in the containment building, which represents the major effort for the decommissioning staff. The work proceeds through the other buildings as staff

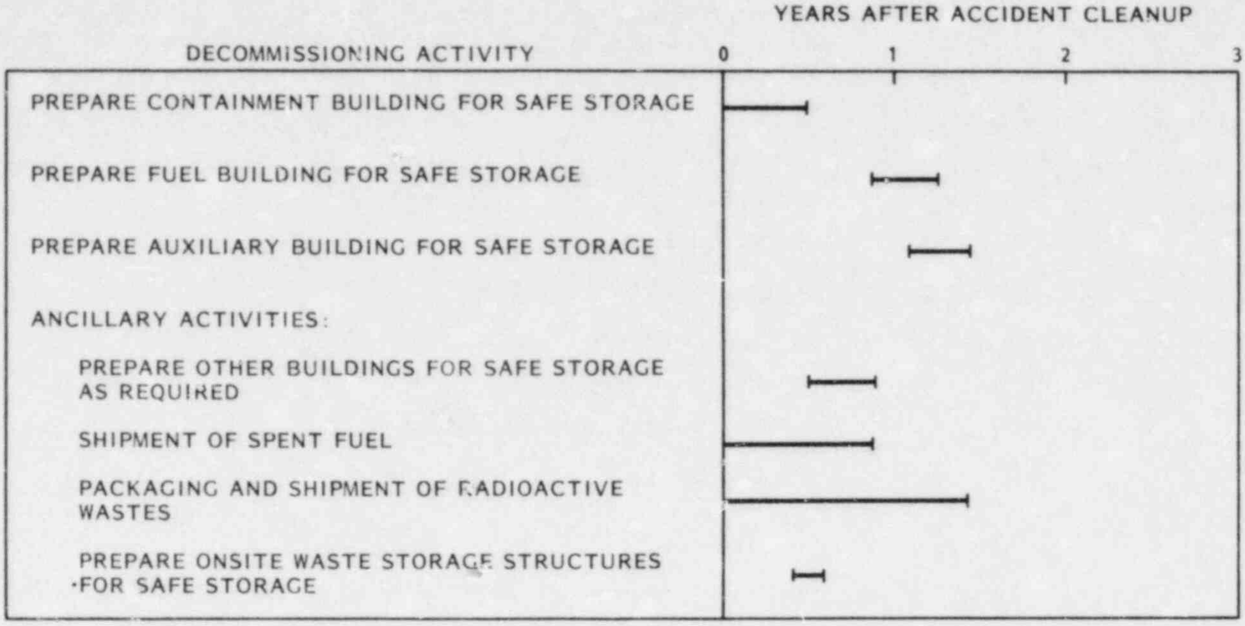


FIGURE 12.3-1. Overall Schedule and Sequence for Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident

are available and as the various systems in these other buildings complete their required service functions. As shown in Figure 12.3-1, preparations for safe storage following a scenario 2 accident are completed in 17 months. The overall duration of the project is not judged to vary significantly with accident severity, within the range of accident scenarios considered in this study, although the level of effort required for some individual tasks within the schedule may vary.

12.3.4 Staff Requirements for Preparations for Safe Storage

As for DECON, individual radiation dose limitations control the number of decommissioning workers needed during preparations for safe storage. The occupational radiation doses to the decommissioning workers during preparations for safe storage and the resulting decommissioning worker requirements are developed and discussed in Section G.3.3 of Appendix G. Manpower requirements for management and support staff are primarily a function of the duration of decommissioning and are developed using the assumptions described in Section 12.2.4.2 for DECON. The organization and the individual functions of the decommissioning staff for preparations for safe storage are the same as those for DECON, presented earlier in Section 12.2.4.1.

Overall staff labor requirements for preparations for safe storage at the reference PWR following a scenario 2 accident are given in Table 12.3-1. These requirements are given in equivalent man-years for the planning and preparation phase as well as for the actual preparations for safe storage, and include both the management and support staff and the decommissioning workers. A total of over 190 man-years of effort is estimated for completion of this first phase of SAFSTOR following a scenario 2 accident.

Management and support staff requirements, primarily dependent on project duration, are postulated not to vary with changes in accident severity. However, scenario 2 decommissioning worker requirements could be reduced by almost half following a scenario 1 accident and increased by more than a factor of two following a scenario 3 accident, because of the estimated variations in the occupational radiation doses received by the decommissioning workers and the individual radiation dose limitations. Therefore, the total staff labor requirements following a scenario 1 or a scenario 3 accident are estimated to be about 155 man-years or about 295 man-years, respectively. For comparison, the staff labor requirements for preparations for safe storage at the reference PWR following normal reactor shutdown total less than 115 man-years.(1)

12.3.5 Activities and Requirements for Continuing Care and Deferred Decontamination

Continuing care (i.e., the safe storage period of SAFSTOR) commences immediately following preparations for safe storage and continues until deferred decontamination of the plant. In this study, two potential safe storage periods are considered, 30 years and 100 years.

The activities carried out during the safe storage period include security, surveillance, and maintenance functions. The level of effort required during continuing care at the reference PWR following an accident is assumed to be approximately the same as that required following normal reactor

TABLE 12.3-1. Overall Staff Labor Requirements for Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident

Position	Staff Labor Requirement (man-years) in Decommissioning Phase:[a] Planning and Preparation	Preparations for Safe Storage	Total Staff Labor Required (man-years)
Management Support Staff			
Decommissioning Superintendent	1.5	1.8[b]	3.3
Secretary	3.0	5.3[b]	8.3
Clerk	1.0	2.8	3.8
Decommissioning Engineer	1.5	1.8[b]	3.3
Assistant Decommissioning Engineer	1.5	1.4	2.9
Radioactive Shipment Specialist	0	1.4	1.4
Procurement Specialist	0	1.4	1.4
Tool Crib Attendant	0	2.8	2.8
Reactor Operator[c]	0	11.3	11.3
Security Supervisor	0	1.4	1.4
Security Shift Supervisor	0	5.7	5.7
Security Patrolmen	0	14.8	14.8
Contracts and Accounting Supervisor	0	1.8[b]	1.8
Health and Safety Supervisor	0	1.8[b]	1.8
Health Physicist	0	1.4	1.4
Protective Equipment Attendant	0	2.8	2.8
Industrial Safety Specialist	0	1.4	1.4
Quality Assurance Supervisor	0	1.8[b]	1.8
Quality Assurance Engineer	0	1.4	1.4
Quality Assurance Technician	0	5.7	5.7
Consultant (Safety Review)	0	0.7	0.7
Instrument Technician[d]	0	5.7	5.7
Maintenance Mechanic[d]	0	5.7	5.7
Warehouseman	0	2.8	2.8
Subtotals	8.5	84.9	93.4
Decommissioning Workers			
Shift Engineer	0	2.8	2.8
Crew Leader[e]	0	6.7	6.7
Utility Operator[e]	0	24.5	24.5
Laborer[e]	0	21.5	21.5
Craft Supervisor	0	5.7	5.7
Craftsmen[e]	0	8.9	8.9
Senior Health Physics Technician	0	8.5	8.5
Health Physics Technician[e]	0	19.9	19.9
Subtotals	0	98.5	98.5
Totals	8.5	183.4	191.9

(a) Rounded to the nearest 0.1 man-year.
(b) Includes an additional 4 months following active decommissioning to complete the documentation and other unspecified license and contract termination requirements.
(c) Based on two operators per shift in the control room, three shifts per day, 7 days per week.
(d) Based on one per shift, three shifts per day, 7 days per week to maintain essential services.
(e) From Table G.3-2.

shutdown. From Table H.4-4 of Reference 1, the annual labor requirement is estimated to be less than 2 man-year/year and, thus, the total cumulative labor requirement for the 30-year or the 100-year safe storage period is conservatively estimated to be 60 man-years or 200 man-years, respectively.

The basic activities performed during deferred decontamination are the same as those performed during DECON. Therefore, the level of effort required to efficiently perform the work during deferred decontamination is assumed to be the same as that for DECON, as shown in Table H.5-2 of Reference 1. However, the decommissioning worker requirements presented in Section 12.2.4 are controlled by the limit on radiation dose to individual workers. Thus, based on the decay of ^{137}Cs (the controlling radionuclide in the post-accident radionuclide inventory, with a 30-year half-life), the decommissioning worker requirements for DECON are estimated to be reduced by about 50% following 30-year safe storage and by about 75% following 100-year safe storage. (The radioactivity of ^{137}Cs would be reduced by about 90% after 100 years of safe storage; however, the decommissioning worker requirements would not be reduced below those required for efficient performance of the work.) Overall staff labor requirements for deferred decontamination following a scenario 2 accident are estimated to total about 480 man-years after 30-year safe storage and about 320 man-years after 100-year safe storage. Following a scenario 1 accident, overall staff labor requirements are estimated to total about 270 man-years after 30-year safe storage and about 220 man-years after 100-year safe storage. Following a scenario 3 accident, overall staff labor requirements for deferred decontamination are estimated to total about 950 man-years after 30-year safe storage and about 560 man-years after 100-year safe storage.

12.4 ENTOMB ACTIVITIES AND MANPOWER REQUIREMENTS

ENTOMB as a decommissioning alternative requires continuation of the facility's nuclear license, unless the entombment structure is eventually reopened and the materials stored inside are surveyed and either released for unrestricted use or shipped to a disposal site. In the first few years after completion of accident cleanup, ENTOMB results in manpower requirements, occupational radiation doses, and costs that are significantly greater than those for preparations for safe storage but somewhat less than those for DECON. ENTOMB appears to be less acceptable following a reactor accident than following normal shutdown of the reactor because: 1) residual radioactivity levels in the facility following an accident, even after substantial accident cleanup efforts, are significantly higher than following normal shutdown, and 2) the post-accident radionuclide inventory decays more slowly than the normal-shutdown inventory because of the presence of large quantities of longer-lived radionuclides (i.e., ^{90}Sr and ^{137}Cs) released by the accident (see Chapter 8).

Planning and preparation activities, entombment activities, schedules and manpower estimates, and activities and requirements for continuing care and possible deferred decontamination are described in this section.

12.4.1 Planning and Preparation Activities

Decommissioning a power reactor via ENTOMB is a complex undertaking that is dependent for successful completion on good planning and on execution of necessary preparatory work prior to the end of the accident cleanup campaign that precedes the decommissioning. Planning and preparation for ENTOMB is assumed to take place during the last 1-1/2 years of the accident cleanup campaign. The planning and preparation activities for ENTOMB are essentially the same as those described in Section 12.2.1 for DECON and are not discussed further.

12.4.2 Entombment Activities

The entombment activities at the reference PWR following a postulated reactor accident are very similar to those following normal shutdown of the reactor. The post-accident entombment activities are summarized here. Details of the entombment activities following normal reactor shutdown are presented in Chapter 4 of Reference 2 and provide the basis for the following discussion.

In this study, as in the previous analysis of ENTOMB following normal shutdown, it is assumed that construction of the entombment structure should make use of existing plant features to the maximum extent possible, to avoid the costs and time requirements of extensive modifications. Thus, entombment is assumed to take place in the lower portion of the containment building, inside the shielded central structures that house the steam generators, the pressurizer, and the reactor vessel and below the operating floor.[2] All penetrations through the structure exterior are sealed and, after emplacement of the waste to be entombed, the top is also sealed to complete the structure. The portions of the plant outside of the entombment structure, including the fuel and auxiliary buildings and the upper portion of the containment building, are decontaminated in the same manner as for DECON. The upper portion of the containment dome is then equipped with security and surveillance monitoring equipment, after which the building is sealed to provide a secondary barrier and weather shield for the entombment structure. One door into the building is fitted with an intrusion alarm and locked, rather than sealed completely, to allow access during continuing care.

With the exception of the reactor vessel internals, which are segmented and packaged for offsite disposal, the radioactive materials remaining within the entombment structure following accident cleanup are entombed onsite, together with as much as possible of the radioactive equipment and structural material from the rest of the plant.[2] It is estimated that up to 7250 m^3 of the radioactive waste material from outside the entombment structure can be entombed with the material originating inside, thus reducing the need for offsite disposal of radioactive wastes by that amount.

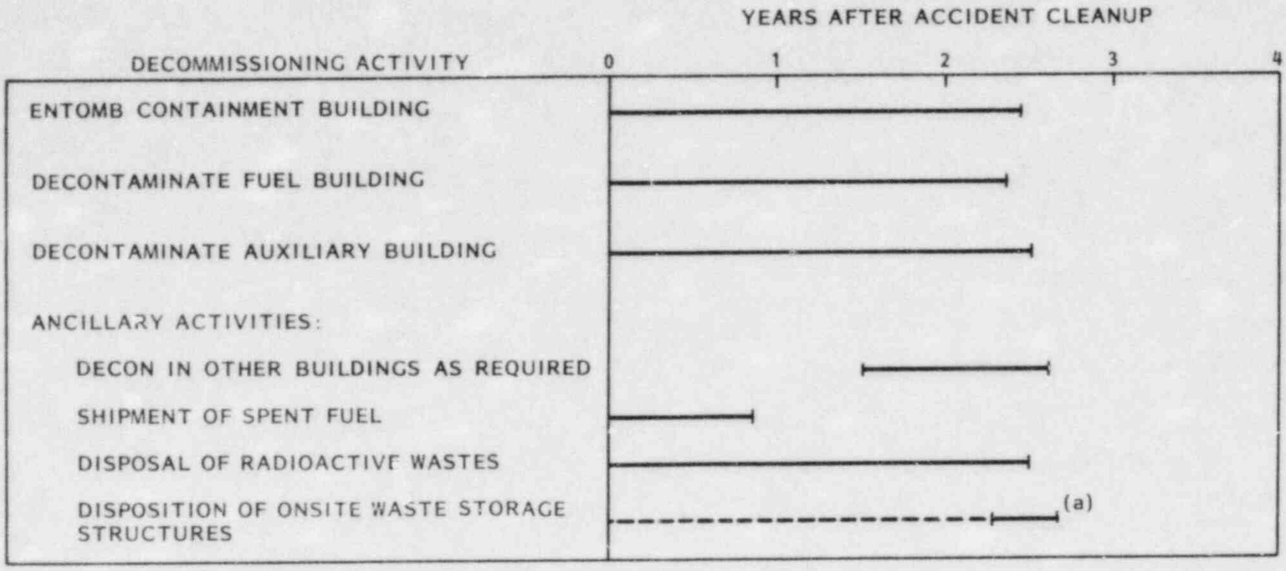
To prepare the selected area to serve as the entombment structure and to receive the wastes to be entombed, several activities are required. Piping that penetrates the postulated entombment structure is cut off and the resulting openings are sealed with welded steel plates, following which the sealed piping sections embedded in the concrete walls are filled with cast-in-place reinforced concrete to provide a continuous concrete skin for the entombment structure. Selected sections of piping and conduit within the structure are removed to improve access and to facilitate movement into the structure of the materials to be entombed. The steam generators and the pressurizer must be relocated so they do not extend above the top of the shielded concrete structures that house them; this involves severing the piping connections, removing the equipment mountings, lowering the equipment to the base slab, and securing the equipment in place. Finally, additional hatchways are cut as needed through the operating floor to aid in moving materials into the structure.

Of the materials selected to be entombed, the larger items are moved into place and stacked, while smaller items and contaminated concrete rubble are simply dumped on top of the larger items to fill the spaces between. Some plant materials are not amenable to entombment because of their large size, extremely high radioactivity levels, or other factors; these materials are packaged and shipped for offsite disposal. The materials to be placed in the entombment structure are selected on the basis of their time of removal and the corresponding progress of the filling of the structure (e.g., larger items removed late in the project may not be entombed simply because the remaining space in the structure can be more easily filled with smaller items).

As mentioned previously, ENTOMB activities outside of the entombment structure area are generally the same as the corresponding DECON activities in those areas. There is, however, one exception. The shielded waste storage facilities (i.e., canyon and caissons) constructed onsite to house accident-cleanup wastes are postulated to be entombed rather than shipping the wastes offsite and decontaminating the structures. Entombing these structures involves the sealing of the cover blocks in place and the decontamination of the upper parts of the structures, and is estimated to require about the same length of time and the same manpower as the DECON activities in these structures. The (unshielded) waste storage warehouse is decontaminated as during DECON.

12.4.3 ENTOMB Schedule

The overall schedule and sequence of events for ENTOMB at the reference PWR following a scenario 2 accident and the subsequent accident cleanup campaign is shown in Figure 12.4-1, based on information presented in Section G.4 of Appendix G. Planning and preparation begins 1-1/2 years prior to the completion of the accident cleanup campaign, as discussed previously in Section 12.4.1.



(a) BROKEN LINE INDICATES INTERMITTENT OFFSITE SHIPMENT OF STORED WASTES IN WAREHOUSE FACILITY, AND SOLID LINE INDICATES DECONTAMINATION OF WAREHOUSE FACILITY AND ENTOMBMENT OF OTHER STRUCTURES.

FIGURE 12.4-1. Overall Schedule and Sequence for ENTOMB at the Reference PWR Following a Scenario 2 Accident

As with the other decommissioning alternatives, ENTOMB begins in the containment building and proceeds through the other buildings as staff are available and as the various systems in these other buildings complete their required service functions. As shown in Figure 12.4-1, ENTOMB following a scenario 2 accident is completed in an estimated 32-1/2 months. Variations in accident severity, within the range of accident scenarios considered in this study, are estimated to result in changes in ENTOMB duration of no more than ±1 month.

12.4.4 ENTOMB Staff Requirements

As during DECON, individual radiation dose limitations control the requirements for decommissioning workers during ENTOMB. The occupational radiation doses to the decommissioning workers during ENTOMB and the resulting decommissioning worker requirements are developed and discussed in Section G.4.3 of Appendix G. Manpower requirements for management and support staff are primarily a function of the duration of decommissioning and are developed using the assumptions described in Section 12.2.4.2 for DECON. The organization and the individual functions of the decommissioning staff for ENTOMB are the same as those for DECON, presented earlier in Section 12.2.4.1.

Overall staff labor requirements for ENTOMB at the reference PWR following a scenario 2 accident are given in Table 12.4-1. Included are both the management and support staff and the decommissioning workers, with equivalent man-years shown for the planning and preparation phase as well as for the actual ENTOMB. A total of almost 680 man-years of effort is estimated for completion of ENTOMB following a postulated scenario 2 accident.

Management and support staff requirements, primarily dependent upon project duration, are not estimated to vary significantly with changes in accident severity. However, the scenario 2 decommissioning worker requirements could be reduced by a factor of about 2.5 following a scenario 1 accident and increased by a factor of between 2 and 2.5 following a scenario 3 accident, because of the estimated variations in the occupational radiation doses received by the decommissioning workers and the limits on radiation doses to the individual workers. Therefore, the total staff labor requirements following a scenario 1 or a scenario 3 accident are estimated to be about 375 man-years or about 1345 man-years, respectively. Although no explicit estimate of manpower requirements for ENTOMB at the reference PWR following normal reactor shutdown is given in Reference 2, the manpower costs shown in Reference 2 for ENTOMB are very nearly the same as the corresponding costs shown in Reference 1 for DECON. Therefore, it is assumed that the staff labor requirements for ENTOMB at the reference PWR following normal reactor shutdown are very nearly 300 man-years.

12.4.5 Activities and Requirements for Continuing Care and Possible Deferred Decontamination

The initial decommissioning activities for ENTOMB are followed by a period of continuing care that includes security, surveillance, and maintenance functions (see Chapter 4). These activities are judged to require a lower level of effort during ENTOMB than the comparable activities during

TABLE 12.4-1. Overall Staff Labor Requirements for ENTOMB at the Reference PWR Following a Scenario 2 Accident

Position	Staff Labor Requirement (man-years) in Decommissioning Phase:[a] Planning and Preparation	ENTOMB	Total Staff Labor Required (man-years)
Management and Support Staff			
Decommissioning Superintendent	1.5	3.0[b]	4.5
Secretary	3.0	8.5[b]	11.5
Clerk	1.0	5.4	6.4
Decommissioning Engineer	1.5	3.0[b]	4.5
Assistant Decommissioning Engineer	1.5	2.7	4.2
Radioactive Shipment Specialist	0	2.7	2.7
Procurement Specialist	0	2.7	2.7
Tool Crib Attendant	0	5.4	5.4
Reactor Operator[c]	0	21.7	21.7
Security Supervisor	0	2.7	2.7
Security Shift Supervisor	0	10.8	10.8
Security Patrolmen	0	28.2	28.2
Contracts and Accounting Supervisor	0	3.0[b]	3.0
Health and Safety Supervisor	0	3.0[b]	3.0
Health Physicist	0	2.7	2.7
Protective Equipment Attendant	0	5.4	5.4
Industrial Safety Specialist	0	2.7	2.7
Quality Assurance Supervisor	0	3.0[b]	3.0
Quality Assurance Engineer	0	2.7	2.7
Quality Assurance Technician	0	10.8	10.8
Consultant (Safety Review)	0	1.4	1.4
Instrument Technician[d]	0	10.8	10.8
Maintenance Mechanic[d]	0	10.8	10.8
Warehouseman	0	5.4	5.4
Subtotals	8.5	158.5	167.0
Decommissioning Workers			
Shift Engineer	0	5.4	5.4
Crew Leader[e]	0	37.0	37.0
Utility Operator[e]	0	92.2	92.2
Laborer[e]	0	156.7	156.7
Craft Supervisor	0	10.8	10.8
Craftsman[e]	0	113.4	113.4
Senior Health Physics Technician	0	16.3	16.3
Health Physics Technician[e]	0	80.0	80.0
Subtotals	0	511.8	511.8
Totals	8.5	670.3	678.8

(a) Rounded to the nearest 0.1 man-year.
(b) Includes an additional 4 months following active decommissioning to complete the documentation and other unspecified license and contract termination requirements.
(c) Based on two operators per shift in the control room, three shifts per day, 7 days per week.
(d) Based on one per shift, three shifts per day, 7 days per week to maintain essential services.
(e) From Table G.4-2.

SAFSTOR, because of the more rigorous preparation of the facility and the resulting reduced risk from the facility. No specific estimate is made of the manpower requirements for the annual continuing care activities following ENTOMB. However, it has been previously estimated that the costs of continuing care following ENTOMB are about half those for SAFSTOR.[3,5] Therefore, assuming that labor accounts for about the same percentage of the total costs in either case and that the make-up of the labor force is approximately the same, the annual labor requirement for continuing care following ENTOMB is estimated to be less than 1 man-year/year. Thus, as an example, the total cumulative labor requirement for 100 years of continuing care is conservatively estimated to be 100 man-years.

Deferred decontamination following ENTOMB, though not analyzed in detail in this study, is anticipated to be an extensive project. Although there is less radioactive material to remove from the plant (because of some offsite disposal during the initial phase of ENTOMB) and this remaining radioactive material is consolidated in a relatively small portion of the facility, the operation is complicated by having to break into the entombment structure (which is designed to retain its integrity under any but the most severe conditions) and to remove the more-or-less randomly placed radioactive materials stored inside.[3] Therefore, the level of effort required for deferred decontamination following ENTOMB is anticipated to be similar to that for deferred decontamination following SAFSTOR, which is discussed previously in Section 12.3.5. The methods used for deferred decontamination following ENTOMB are similar to those for DECON, which are described in some detail in this study, and are not discussed further here.

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3. Final Programmatic Environmental Impact Statement Related to the Decontamination and Disposal of Radioactive Wastes Resulting from March 28, 1979, Accident: Three Mile Island Nuclear Station, Unit 2, NUREG-0683, U.S. Nuclear Regulatory Commission, Washington, D.C., March 1981.

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5. H. D. Oak, et al., Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station, NUREG/CR-0672, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, June 1980.

CHAPTER 13

COSTS OF DECOMMISSIONING AT A REFERENCE PWR

The costs of accomplishing the decommissioning of the reference PWR following a postulated reactor accident and subsequent accident cleanup campaign are developed in detail in Appendix H of Volume 2. These costs are summarized in the following sections. Costs of accident cleanup are summarized previously in Chapter 11.

The principal assumptions made in developing the cost estimates for the post-accident decommissioning of the reference PWR are as follows:

- The costs of decommissioning are considered separate from the costs of accident cleanup. The costs of decommissioning systems and facilities installed during accident cleanup are included with the decommissioning costs.
- Detailed cost estimates are prepared for decommissioning following a scenario 2 accident, with estimates for decommissioning following the other two postulated accidents arrived at by appropriate adjustment of the post-scenario 2 estimates, taking into account the specific conditions that can affect costs.
- Cost estimates are made for decommissioning via each of three decommissioning alternatives: DECON, SAFSTOR, and ENTOMB.
- To the maximum extent possible, the decommissioning staff is drawn from the technical and operational staffs of the plant and from the accident cleanup staff. The support services and the part-time assistance of the accident cleanup staff are assumed to be available during planning and preparations for decommissioning with only nominal costs to the decommissioning portion of the overall cleanup and decommissioning project.
- The amended facility license allowing possession but not operation of the plant is in place at the end of the accident cleanup campaign, permitting decommissioning activities to commence without unnecessary delays.

- Chemical decontamination of the selected systems and equipment permits the decommissioning staff to work in direct contact with these systems.
- Pool liners, most piping and equipment, and significant portions of the concrete surfaces in the containment, auxiliary, fuel, and control buildings are contaminated and require proper management as radioactive wastes.
- The cost estimates presented here are based on the ability to dispose of the radioactive waste materials at offsite facilities at the time of decommissioning. Cost variations resulting from alternate waste disposal scenarios are developed separately in Chapter 15.
- Costs are based on early-1981 prices and wage rates.

13.1 DECON COSTS

The estimated cost of DECON at the reference PWR following a scenario 2 accident, including a 25% contingency, is $67.8 million, as summarized in Table 13.1-1. Corresponding costs following a scenario 1 or scenario 3 accident are $49.4 million or $106.2 million, respectively. Details of the development of these costs are discussed in Section H.1 of Appendix H (Volume 2). Information pertaining to the individual cost categories is summarized in the following subsections.

13.1.1 Costs of Staff Labor

The costs of staff labor during DECON following a postulated scenario 2 accident are shown in detail in Table H.1-2 of Appendix H. A major portion of the total DECON cost is associated with staff labor. A total staff labor cost of about $26.8 million is estimated for DECON following a scenario 2 accident. Staff labor costs following a scenario 1 or scenario 3 accident are estimated to be $13.1 million and $57.9 million, respectively, with the differences between the labor costs for the various accident scenarios attributable almost entirely to the number of decommissioning workers needed to comply with individual radiation dose limitations to those workers (see Sections G.2.3 and G.2.4 of Appendix G, Volume 2). Specialty contractor labor is not included in the labor costs given here, but rather in the costs of specialty contractors presented in Section 13.1.6.

TABLE 13.1-1. Summary of Estimated Costs of Post-Accident DECON at the Reference PWR

Cost Category	Following Scenario 1 Accident[a] Estimated Costs ($ millions)[b]	Following Scenario 1 Accident[a] Percent of Total	Following Scenario 2 Accident Estimated Costs ($ millions)[b]	Following Scenario 2 Accident Percent of Total	Following Scenario 3 Accident[a] Estimated Costs ($ millions)[b]	Following Scenario 3 Accident[a] Percent of Total
Staff Labor						
Management and Support Staff	6.1		6.1		6.1	
Decommissioning Workers	7.0		20.7		51.8	
Total Staff Labor Costs	13.1	33	26.8	49	57.9	68
Waste Management						
Neutron-Activated Materials	4.5		4.5		3.2	
Contaminated Materials	9.5		10.0		10.5	
Radioactive Wastes	1.5		1.7		1.8	
Total Waste Management Costs	15.5	39	16.2	30	15.5	18
Energy	6.2	16	6.5	12	6.9	8
Special Tools and Equipment	1.0	2	1.0	2	1.0	1
Miscellaneous Supplies	1.9	5	1.9	4	1.9	2
Specialty Contractors	0.7	2	0.7	1	0.7	1
Nuclear Insurance and License Fees	1.1	3	1.1	2	1.1	1
Subtotals	39.5	100	54.2	100	85.0	99[c]
Contingency (25%)	9.9		13.6		21.2	
Totals, DECON Costs	49.4		67.8		106.2	

(a) No detailed analysis performed for DECON following scenario 1 or scenario 3 accidents; estimates shown derived by difference from those for scenario 2.
(b) Individually rounded to the nearest $0.1 million; costs adjusted to early 1981.
(c) Total does not equal 100 because individual percentages are rounded to the nearest whole percent.

13.1.2 Costs of Radioactive Waste Management

The three types of radioactive materials in the reference PWR that require proper waste management are: 1) neutron-activated materials, 2) contaminated materials, and 3) radioactive wastes. The total waste management cost for these materials during DECON following a scenario 2 accident is $16.2 million and represents approximately 30% of the total DECON cost. Following either a scenario 1 or scenario 3 accident, the total estimated cost is $15.5 million. The waste management costs include the container, transportation, and burial site costs but not the direct labor costs for removing and packaging the materials.

The neutron-activated materials are contained in the reactor pressure vessel, the vessel internals, and the biological shield, all located in the central shielded portion of the containment building. Details of the disposal of these materials are given in Section H.1.2.1 of Appendix H. The total radioactivity estimated to be present in the neutron-activated materials is approximately 4.8 million curies. The packaged materials require an estimated 203 overweight truck shipments and occupy 1124 m^3 of space in a shallow-land burial facility. The total estimated cost for management of neutron-activated materials during DECON following a scenario 2 accident is $4.5 million. More of the reactor internals are removed during accident cleanup following a scenario 3 accident and, accordingly, the neutron-activated waste management cost during DECON following that accident is estimated to be about $3.2 million.

Contaminated materials in the reference PWR are assumed to include nearly all of the piping and equipment in the containment, auxiliary, fuel, and control buildings. In addition, many concrete surfaces in these buildings and in the onsite waste storage structures erected during the accident cleanup campaign are assumed to be contaminated, thus requiring removal of a surface layer. Breakdowns of the disposal costs for contaminated materials are presented in Section H.1.2.2 of Appendix H. The contaminated materials removed during DECON following a postulated scenario 2 accident require an estimated 1001 truckload shipments to and an estimated 16,150 m^3 of space (including the disposable containers, as required) at a shallow-land burial site. The total waste management cost for these materials following a scenario 2 accident is about \$10.0 million. Corresponding costs following a scenario 1 or scenario 3 accident are estimated at \$9.5 million or \$10.5 million, respectively.

Radioactive wastes generated during DECON at the reference PWR are categorized as radioactive trash,[a] spent ion exchange resins and filter cartridges, and solidified evaporator bottoms. Details of the waste management of these materials are given in Section H.1.2.3 of Appendix H. Following a postulated scenario 2 accident, these wastes are estimated to fill 158 truckload shipments and to occupy 1516 m^3 in a shallow-land burial facility, at a total cost of almost \$1.7 million. Management of these wastes is estimated to cost about \$1.5 million following a scenario 1 accident and about \$1.8 million following a scenario 3 accident.

13.1.3 Costs of Energy

The costs of electrical energy are based on estimated usage during DECON and are discussed in Section H.1.3 of Appendix H. A total of 261,000 MWh of electricity, costing about \$6.5 million, is estimated for DECON at the reference PWR following a scenario 2 accident, representing about 12% of the total DECON cost. For DECON following a scenario 1 or scenario 3 accident, the estimated total energy cost is \$6.2 million or \$6.9 million, respectively.

13.1.4 Costs of Special Tools and Equipment

The estimated costs of the special tools and equipment that are required for post-accident DECON at the reference PWR are presented in Table H.1-4 of Appendix H. The estimated total cost of special tools and equipment is approximately \$1.0 million, which is about 2% of the total DECON cost following a scenario 2 accident. Because the needs for special tools and equipment during DECON are essentially independent of accident severity, the special tools and equipment costs following the other two accident scenarios are considered to be the same as those following a scenario 2 accident.

(a) Includes a compactible and combustible fraction, a compactible but noncombustible fraction, and a noncompactible (and noncombustible) fraction.

13.1.5 Costs of Miscellaneous Supplies

A variety of expendable supplies are used during DECON. These include decontamination chemicals, ion exchange resins, glass-fiber and HEPA filters, cartridge-type fluid filters, disposable protective clothing, assorted cleaning supplies, and expendable tools and materials. The estimated costs of these items are given in Table H.1-5 of Appendix H. The total estimated cost of miscellaneous supplies during DECON at the reference PWR after a scenario 2 accident is about $1.9 million, which represents about 4% of the total DECON cost. Costs of miscellaneous supplies are judged not to vary significantly with changes in accident severity, within the range of accident scenarios considered in this study.

13.1.6 Costs of Specialty Contractors

The estimated requirements for and costs of specialty contractors during DECON are discussed in Section H.1.6 of Appendix H. These specialty contractors perform explosives work, temporary radwaste handling, and environmental surveillance services. Hauling contractor costs are not included here but are shown as "transportation costs" in Section H.1.2 of Appendix H for waste management.

The total cost of specialty contractors during DECON at the reference PWR, excluding the hauling contractor, is less than $0.7 million, regardless of the severity of the postulated accident. Specialty contractors account for about 1% of the total DECON costs following a scenario 2 accident.

13.1.7 Costs of Nuclear Insurance and License Fees

The costs of nuclear liability insurance during DECON are estimated for an assumed policy limit of $160 million carried through the DECON period. The total estimated cost of nuclear insurance is $1.1 million. The fees charged for licensing services performed by the NRC are detailed in Table H.1-6 of Appendix H. These fees total an estimated $38,000 during DECON. Together, the costs of nuclear insurance and license fees account for about 2% of the total costs for DECON at the reference PWR following a scenario 2 accident. These costs are unaffected by the severity of the postulated accident.

13.2 SAFSTOR COSTS

The estimated costs of SAFSTOR at the reference PWR, after the reactor has experienced a postulated accident and the accident cleanup campaign has been completed, are summarized in Table 13.2-1. Costs are included for the three phases of SAFSTOR:

- preparations for safe storage
- continuing care (i.e., the safe storage period)
- deferred decontamination.

The costs given for each phase include a 25% contingency. Details of the development of these costs are given in Section H.2 of Appendix H.

Total SAFSTOR costs at the reference PWR following a scenario 2 accident are estimated to be almost $78 million with a 30-year safe storage period and over $72 million with 100 years of safe storage. Deferred decontamination accounts for the majority of the cost, while preparations for safe storage make up between one-fourth and one-fifth of the total. All costs are given in constant 1981 dollars, with no escalation for inflationary effects included.

Approximate costs of SAFSTOR following a scenario 1 accident total about $60 million with 30-year safe storage and just over $58 million with 100-year storage. Corresponding totals following a scenario 3 accident are about $115 million and $102 million for SAFSTOR with 30 and 100 years, respectively, of safe storage. The cost differences between the three accident scenarios result primarily from the deferred decontamination, with a smaller impact from the preparations for safe storage.

13.2.1 Costs of Preparations for Safe Storage

The estimated total cost of preparations for safe storage at the reference PWR following a scenario 2 accident, as shown in Table 13.2-1, is $16.6 million, including a 25% contingency. Corresponding costs following a scenario 1 or scenario 3 accident are $14.7 million or $21.4 million, respectively. Information pertaining to the individual cost categories that make up this total is summarized in the following subsections.

TABLE 13.2-1. Summary of Estimated Costs of Post-Accident SAFSTOR at the Reference PWR

Cost Category	Following Scenario 1 Accident[a] Estimated Costs ($ millions)[b]	Following Scenario 1 Accident[a] Percent of Total	Following Scenario 2 Accident Estimated Costs ($ millions)[b]	Following Scenario 2 Accident Percent of Total	Following Scenario 3 Accident[a] Estimated Costs ($ millions)[b]	Following Scenario 3 Accident[a] Percent of Total
Preparations for Safe Storage						
Staff Labor						
Management and Support Staff	3.5		3.5		3.5	
Decommissioning Workers	2.1		3.4		6.9	
Total Staff Labor Costs	5.6	47	6.9	52	10.4	61
Waste Management (Radioactive Wastes)	0.8	7	0.9	6	1.0	6
Energy	3.2	27	3.4	26	3.6	21
Special Tools and Equipment	0.3	3	0.3	3	0.3	2
Miscellaneous Supplies	0.9	8	0.9	7	0.9	5
Specialty Contractors	0.3	3	0.3	2	0.3	2
Nuclear Insurance and License Fees	0.6	5	0.6	4	0.6	3
Subtotal	11.7	100	13.3	100	17.1	100
Contingency (25%)	3.0		3.3		4.3	
Totals, Preparations for Safe Storage Costs	14.7		16.6		21.4	
Annual Continuing Care Costs	0.11		0.11		0.11	
Deferred Decontamination Costs						
After 30-Year Safe Storage	42.0		57.6		90.3	
After 100-Year Safe Storage	32.6		44.7		70.1	
Total SAFSTOR Costs						
With 30-Year Safe Storage	60.0		77.5		115.0	
With 100-Year Safe Storage	58.3		72.3		102.3	

(a) No detailed analysis performed for SAFSTOR following scenario 1 or scenario 3 accidents; estimates shown derived by difference from those for scenario 2.
(b) Individually rounded to the nearest $0.1 million, except annual continuing care costs rounded to nearest $10 thousand; costs adjusted to early 1981.

13.2.1.1 Costs of Staff Labor

The costs of staff labor during preparations for safe storage following a scenario 2 accident are shown in detail in Table H.2-2 of Appendix H. Over 50% of the total cost of preparations for safe storage is attributable to staff labor. A total staff labor cost of about $6.9 million is estimated for preparations for safe storage following a scenario 2 accident. Staff labor costs following a scenario 1 or scenario 3 accident are estimated to be $5.6 million and $10.4 million, respectively, with the differences between the totals for the various accident scenarios due almost entirely to changes in the number of decommissioning workers needed (see Sections G.3.3 and G.3.4 of Appendix G). Specialty contractor labor is not included in the labor costs given here, but rather in the specialty contractor costs presented in Section 13.2.1.6.

13.2.1.2 Costs of Radioactive Waste Management

Only one type of radioactive waste material, termed radioactive wastes in the previous discussion of waste management during DECON (Section 13.1.2), requires packaging, shipping, and offsite disposal during preparations for safe storage. Included in this waste type are radioactive trash, spent ion exchange resins and filter cartridges, and solidified evaporator bottoms. The total waste management cost during preparations for safe storage at the reference PWR following a scenario 2 accident is about $0.9 million, which represents about 6% of the overall total cost of preparations for safe storage. Following a scenario 1 or a scenario 3 accident, the waste management cost is estimated to be about $0.8 million or $1.0 million, respectively, corresponding to variations in the amount of radioactive trash generated during the decommissioning activities. The waste management costs include the container, transportation, and burial site costs but not the direct labor costs for removing and packaging the materials. A breakdown of the costs and other pertinent parameters associated with waste management during preparations for safe storage is given in Table H.2-3 of Appendix H.

13.2.1.3 Costs of Energy

The costs of electrical energy during preparations for safe storage at the reference PWR are discussed in Section H.2.1.3 of Appendix H. The total energy usage and resulting costs during preparations for safe storage following a scenario 2 accident are about 137,000 MWh and $3.4 million, respectively. Energy costs represent over 25% of the total cost of preparations for safe storage following a scenario 2 accident. Energy costs following a scenario 1 or a scenario 3 accident are estimated at $3.2 million and $3.6 million, respectively.

13.2.1.4 Costs of Special Tools and Equipment

The estimated costs of the special tools and equipment that are needed in preparing the reference PWR for safe storage are presented in Table H.2-4 of Appendix H. The estimated total cost of special tools and equipment is approximately $0.3 million, regardless of the accident scenario considered. Special tools and equipment account for 2 to 3% of the overall cost of preparations for safe storage.

13.2.1.5 Costs of Miscellaneous Supplies

A variety of expendable supplies are used during preparations for safe storage, the estimated costs of which are shown in Table H.2-5 of Appendix H. The total estimated cost of miscellaneous supplies during preparations for safe storage at the reference PWR after a scenario 2 accident is about $0.9 million, which represents 7% of the overall costs for preparations for safe storage. Costs of miscellaneous supplies are not judged to vary significantly with changes in accident severity, within the range of accident scenarios considered in this study.

13.2.1.6 Costs of Specialty Contractors

The estimated requirements for and costs of specialty contractors during preparations for safe storage are discussed in Section H.2.1.6 of Appendix H. These specialty contractors provide decontamination services, temporary radwaste handling, and environmental surveillance services. Hauling contractor costs are not included here but, rather, are included in the waste management costs presented previously. The total cost of specialty contractors during preparations for safe storage at the reference PWR, excluding the hauling contractor, is just about $0.3 million, regardless of the accident scenario considered. Specialty contractors account for 2 to 3% of the total cost of preparations for safe storage.

13.2.1.7 Costs of Nuclear Insurance and License Fees

The costs of nuclear liability insurance during preparations for safe storage are estimated for an assumed policy limit of $160 million carried through the preparations period and are estimated to total $600,000. The fees charged for licensing services performed by the NRC, detailed in Table H.2-6 of Appendix H, total an estimated $25,000 during preparations for safe storage. Together, the costs of nuclear insurance and license fees during preparations for safe storage account for about 4% of the total cost following a scenario 2 accident. These costs are unaffected by the severity of the postulated accident.

13.2.2 Costs of Continuing Care During Safe Storage

The estimated annual costs of continuing care at the reference PWR during safe storage are presented in Table H.2-7 of Appendix H. The total annual cost is estimated to be approximately $110,000. Staff labor accounts for 70% of the total, with allowances for repairs and utilities and services contributing 16%. Nuclear insurance (11%), equipment and supplies (1%), and license fees (<1%) constitute the balance of the annual cost. These costs are essentially independent of the accident scenario considered.

13.2.3 Costs of Deferred Decontamination to Terminate SAFSTOR

The costs of deferred decontamination at the reference PWR are estimated based on the assumption that the ratio of the deferred decontamination cost (after a specified period of safe storage) to the DECON cost (in effect, the immediate decontamination cost) is not substantially altered by the occurrence of a reactor accident if the accident cleanup campaign preceding SAFSTOR achieves the objectives presented in Appendix E (Volume 2). A comparison of the costs of DECON and of deferred decontamination, both following normal shutdown and following a postulated scenario 2 accident, is presented in Table H.2-8 of Appendix H. The estimated cost of deferred decontamination following a scenario 2 accident is about $58 million after 30 years of safe storage or about $45 million after 100 years of storage. Similarly, the deferred decontamination cost following a scenario 1 accident is approximately $42 million or $33 million after 30 or 100 years, respectively, of safe storage. Following a scenario 3 accident, the corresponding cost is about $90 million or $70 million following safe storage periods of 30 or 100 years, respectively.

13.3 ENTOMB COSTS

The estimated costs of ENTOMB at the reference PWR following a postulated accident, developed in detail in Section H.3 of Appendix H, are summarized in Table 13.3-1. The total estimated ENTOMB cost following a scenario 2 accident, including a 25% contingency, is $52.5 million. Corresponding costs following a scenario 1 or scenario 3 accident are $38.5 million or $79.6 million, respectively. Continuing care of the entombed plant is estimated to cost $55,000 annually, independent of the accident that is postulated to have occurred. Information pertaining to the individual cost categories in Table 13.3-1 is summarized in the following subsections.

13.3.1 Costs of Staff Labor

The costs of staff labor during ENTOMB following a postulated scenario 2 accident are shown in detail in Table H.3-2 of Appendix H. A major portion of the total ENTOMB cost is associated with staff labor. A total staff labor cost of $23.2 million is estimated for ENTOMB following a scenario 2 accident. Staff labor costs following a scenario 1 or scenario 3 accident are estimated to be $12.9 million or $45.4 million, respectively, with the differences between the labor costs for the three accident scenarios attributable almost entirely to the number of decommissioning workers necessary to comply with individual radiation dose limitations to the workers (see Sections G.4.3 and G.4.4 of Appendix G). Specialty contractor labor is included in the specialty contractor costs given below and is not included in the labor costs given here.

13.3.2 Costs of Radioactive Waste Management

Costs of radioactive waste management include those associated with the management of neutron-activated materials, contaminated materials, and radioactive wastes that require packaging, transportaton, and disposal at an offsite shallow-land burial facility. For ENTOMB following a postulated

TABLE 13.3-1. Summary of Estimated Costs of Post-Accident ENTOMB at the Reference PWR

Cost Category	Following Scenario 1 Accident[a] Estimated Costs ($ millions)[b]	Percent of Total	Following Scenario 2 Accident Estimated Costs ($ millions)[b]	Percent of Total	Following Scenario 3 Accident[a] Estimated Costs ($ millions)[b]	Percent of Total
Staff Labor						
Management and Support Staff	6.1		6.1		6.1	
Decommissioning Workers	6.8		17.1		39.3	
Total Staff Labor Costs	12.9	42	23.2	55	45.4	71
Waste Management						
Neutron-Activated Materials	2.2		2.2		0.9	
Contaminated Materials	3.5		3.9		4.3	
Radioactive Wastes	1.5		1.7		1.8	
Total Waste Management Costs	7.2	23	7.8	19	7.0	11
Energy	6.2	20	6.5	16	6.8	11
Special Tools and Equipment	1.0	3	1.0	2	1.0	2
Miscellaneous Supplies	1.9	6	1.9	4	1.9	3
Specialty Contractors	0.5	2	0.5	1	0.5	1
Nuclear Insurance and License Fees	1.1	4	1.1	3	1.1	2
Subtotals	30.8	100	42.0	100	63.7	101[c]
Contingency (25%)	7.7		10.5		15.9	
Totals, ENTOMB Costs	38.5		52.5		79.6	
Annual Continuing Care Costs	0.055		0.055		0.055	

(a) No detailed analysis performed for ENTOMB following scenario 1 or scenario 3 accidents; estimates shown derived by difference from those for scenario 2.
(b) Individually rounded to the nearest $0.1 million, except annual continuing care costs rounded to nearest $5000; costs adjusted to early 1981.
(c) Total does not equal 100 because individual percentages are rounded to the nearest whole percent.

scenario 2 accident, the total waste management cost is estimated to be $7.8 million and to represent almost 19% of the total ENTOMB cost. The waste management costs are detailed in Section H.3.2 of Appendix H. Waste management costs following a scenario 1 or scenario 3 accident are estimated at $7.2 million or $7.0 million, respectively. Although waste management requirements and costs generally increase somewhat with increasing accident severity, the waste management cost for ENTOMB following a scenario 3 accident is lower than that following either of the other scenarios because more of the activated reactor vessel internals are removed and disposed of during the accident cleanup campaign following that accident, thus reducing activated-material disposal requirements and costs during ENTOMB.

13.3.3 Costs of Energy

Electrical energy usage and costs during ENTOMB are discussed in Section H.3.3 of Appendix H, and are assumed to be the same as those shown previously in Section 13.1.3 for DECON. A total of 261,000 MWh, costing about $6.5 million, is estimated following a scenario 2 accident, accounting for nearly 16% of the total ENTOMB cost. Following a scenario 1 or scenario 3 accident, the estimated total energy cost is $6.2 million or $6.8 million, respectively.

13.3.4 Costs of Special Tools and Equipment

The estimated costs of the special tools and equipment that are required for post-accident ENTOMB at the reference PWR are presented in Table H.3-4 of Appendix H. The estimated total cost of special tools and equipment is about $1.0 million, which is about 2 to 3% of the overall ENTOMB cost and is assumed to be independent of the accident scenario considered.

13.3.5 Costs of Miscellaneous Supplies

The estimated costs of the expendable supplies used during ENTOMB are given in Table H.3-5 of Appendix H. The total estimated cost of miscellaneous supplies during post-accident ENTOMB at the reference PWR is about $1.9 million, which represents about 4% of the total ENTOMB cost following a scenario 2 accident. Costs of miscellaneous supplies are not judged to vary significantly with changes in accident severity, within the range of accident scenarios considered in this study.

13.3.6 Costs of Specialty Contractors

The estimated requirements for and costs of specialty contractors that provide temporary radwaste handling and environment surveillance services during ENTOMB are discussed in Section H.3.6 of Appendix H. The cost of a hauling contractor is not included here but is included in the waste management costs. The total cost of specialty contractors during ENTOMB at the reference PWR, excluding the hauling contractor, is estimated to be about \$0.5 million, regardless of the accident scenario considered. Specialty contractors account for just over 1% of the total cost of ENTOMB following a scenario 2 accident.

13.3.7 Costs of Nuclear Insurance and License Fees

The costs of nuclear liability insurance during ENTOMB, for an assumed policy limit of \$160 million carried through the ENTOMB period, is estimated to total \$1.1 million. The fees charged for licensing services performed by the NRC, detailed in Table H.3-6 of Appendix H, add an estimated \$26,000 to the total. Together, the costs of nuclear insurance and license fees during ENTOMB account for almost 3% of the total cost following a scenario 2 accident. These costs are unaffected by the severity of the postulated accident.

13.3.8 Costs of Continuing Care and Possible Deferred Decontamination

The costs of continuing care and possible deferred decontamination following ENTOMB of the reference PWR are discussed in Section H.3.8 of Appendix H. The annual continuing care costs are estimated to total approximately \$55,000. Thus, a continuing care period of 100 years would add about \$5.5 million to the cost of decommissioning the reference PWR by the ENTOMB alternative. If required, deferred decontamination at the end of the continuing care period for ENTOMB is estimated to be more costly than deferred decontamination at the end of the corresponding continuing care period for SAFSTOR because dismantlement of an entombed structure is more difficult than dismantlement of an unentombed facility.

CHAPTER 14

SAFETY IMPACTS OF POST-ACCIDENT CLEANUP AND DECOMMISSIONING AT A REFERENCE PWR

Occupational, public, and transportation safety impacts from post-accident decommissioning, including post-accident cleanup, at the reference PWR are summarized in this chapter. Safety impacts from accident cleanup and decommissioning include: 1) radiation doses to the public from atmospheric releases of radioactivity resulting from routine tasks and industrial accidents during accident cleanup and decommissioning, 2) radiation doses to and industrial accidents involving workers performing the cleanup and decommissioning tasks, and 3) radiation doses to and accidents involving the transportation workers and the public during the shipment of radioactive materials from the site. A conservative approach, using parameters that tend to realistically maximize the consequences, is used to evaluate the safety impacts of accident cleanup and decommissioning. The evaluation uses current analysis data and methodology.

This chapter is divided into three sections: technical approach, accident cleanup safety, and decommissioning safety. Each of the latter two sections is divided into subsections that encompass public safety, occupational safety, and transportation safety. The information presented in this chapter is summarized from Appendix J of Volume 2. Activities and manpower requirements for PWR accident cleanup and for decommissioning are discussed in Chapters 10 and 12, respectively.

A basic assumption of the safety analyses in this chapter is that the radioactive waste materials from accident cleanup and decommissioning are disposed of offsite at the time of decommissioning. The safety impacts of alternate scenarios for waste disposal are addressed separately in Chapter 15.

14.1 TECHNICAL APPROACH

The safety evaluation is divided into two areas of interest: radiological safety and nonradiological safety. Radiological safety impacts are evaluated using a three-part approach. First, a description of the reference facility is developed (see Chapter 7). Second, the radionuclide inventories and external dose rates within the facility resulting from normal reactor operations and from the postulated reactor accidents are characterized and quantified (see Chapters 7 and 8). Finally, reference tasks are used to calculate radiation exposures to workers and to the public from accident cleanup and from decommissioning (by any of the three decommissioning alternatives). (The tasks for accident cleanup are described in Chapter 10; the tasks for decommissioning are described in Chapter 12.)

The nonradiological safety evaluation is based on the potential for industrial and transportation accidents that result in injuries or fatalities. The technical approach is divided into two parts. First, the total labor requirements for accident cleanup and decommissioning are analyzed and divided into categories of effort, and transportation requirements are quantified; second, injuries and fatalities are calculated based on statistical information from the literature concerning accident frequencies for the selected categories of effort.

To provide a basis for the estimation of the safety impacts of accident cleanup and decommissioning and to ensure consistency between the various parts of the safety analysis, the following basic assumptions are made:

1. Appropriate radiation protection and contamination control techniques are applied to conform to the principle of keeping occupational radiation doses and radioactivity levels in effluents as low as reasonably achievable (ALARA).

2. The analysis of the public safety impacts resulting from the release of radioactive materials during accident cleanup is based primarily on information developed in Reference 1 for the cleanup of TMI-2, with appropriate adjustments to account for differences in fuel burnup and in accident severities postulated for the reference PWR.

3. The assessments of the safety impacts of post-accident decommissioning are based primarily on information pertaining to the decommissioning of the reference PWR following normal reactor shutdown,

developed in Appendix J of Reference 2. Appropriate adjustments are made to account for differences between post-accident and normal-shutdown radionuclide inventories and decommissioning requirements.

4. All offsite shipments of radioactive wastes are in accordance with applicable Department of Transportation regulations. Spent reactor fuel is shipped by rail 1600 km to either an ISFSI or a federal repository, and radioactive wastes are shipped 1600 km by truck to either a shallow-land burial site or a federal repository.

5. The largest potential radiological consequences of a given cleanup or decommissioning task are associated with performing that task in the area of the plant with the largest inventory of radionuclides.

6. The maximum radioactive release from a specific task is assigned to that task whenever it is performed in the facility. In performing the dose calculations for releases from routine tasks, the estimated total releases for the entire cleanup or decommissioning period are assumed to be released at a uniform rate over a 1-year period.

7. In calculating atmospheric releases of radioactivity during decommissioning, no credit is taken for the radioactive decay of the radionuclide inventories present at the completion of the accident cleanup campaign.

8. Public radiation doses are calculated using the environmental data and assumptions discussed in Appendix E of Reference 2, consistent with the methods outlined in Regulatory Guide 1.109.[3]

Other specific assumptions used in calculating the occupational doses are found in Appendices E and G of Volume 2. A discussion of the assumptions used for the public and transportation radiation dose calculations is included in Appendix J.

14.2 ACCIDENT CLEANUP SAFETY

Accident cleanup activities at the reference PWR precede the actual refurbishment or decommissioning of the plant and are essentially independent of whether the facility is to be refurbished or decommissioned and, in the latter case, of the alternative chosen for completing the decommissioning. As a practical matter, accident cleanup efforts contribute to the refurbishment or decommissioning effort. However, in this analysis, accident cleanup is addressed separately from decommissioning.

This section summarizes the detailed analysis of the safety impacts resulting from the accident cleanup activities at the reference PWR, presented in Section J.2 of Appendix J. Radiological safety impacts to the public are described in Section 14.2.1, occupational safety impacts of accident cleanup are addressed in Section 14.2.2, and transportation safety impacts both to transportation workers and to the public are described in Section 14.2.3.

14.2.1 Public Safety Impacts of Accident Cleanup

The public radiological safety impacts of onsite activities during accident cleanup are discussed in this section. Safety impacts are evaluated for routine tasks carried out during accident cleanup as well as for postulated industrial accidents that may occur. Nonradiological safety impacts to the public from onsite activities are judged to be negligible and are not considered further. Public safety impacts from offsite transportation activities are included later in Section 14.2.3.

The consequences of atmospheric releases of radioactivity from routine tasks during accident cleanup are determined by calculating radiation doses to the maximum-exposed individual and to the population residing within 80 km of the site. Radiation exposure pathways considered for these releases are direct external exposure, inhalation, and ingestion of food products. The consequences of postulated industrial accidents that could occur during accident cleanup and that could result in airborne releases of radioactivity are determined by calculating inhalation radiation doses to the maximum-exposed individual.

The estimated atmospheric releases during accident cleanup at the reference PWR, used to estimate public radiation doses from accident cleanup, are based on estimated values for releases from equivalent situations during cleanup at TMI-2, reported in Reference 1, with appropriate adjustments made in this study to account for differences in the fuel burnup and in the release fractions of radionuclides escaping the reactor core at the time of the accident. Adjustments are based on the post-accident radionuclide inventories existing at TMI-2 and on those postulated to exist at the reference PWR following each of the reactor accident scenarios described in Chapter 8 of this study.

14.2.1.1 Public Radiation Doses from Routine Tasks During Accident Cleanup

Loss of confinement of radioactive materials resulting in public radiation exposure is a primary safety concern during accident cleanup at the reference PWR. The radiation doses from these releases are calculated using the dose models discussed in Appendix E of Reference 2 in conjunction with the appropriate radionuclide inventories as presented in Appendix C of this study. (The dose conversion factors used in this study are different than those used in Reference 2 because of the differences in the radionuclide inventories involved.)

Tables 14.2-1 and 14.2-2 contain summaries of the calculated radiation doses to the maximum-exposed individual and to the population residing within 80 km of the site as a result of releases from routine tasks during accident cleanup. The radiation doses listed in the tables are the first-year radiation dose and fifty-year committed radiation dose equivalent to total body, bone, and lung. The doses from cleanup activities following accident scenario 1 are about an order of magnitude lower than those following accident scenario 2 which, in turn, are about a factor of 2 lower than those following accident scenario 3. Doses to the maximum-exposed individual in any given year are estimated to be below the appropriate dose design objectives set forth in 10 CFR 50, Appendix I.[5]

The radiation doses shown in the tables include only activities in the reference PWR containment building. However, it is postulated that some accident cleanup activities may be required in the fuel and auxiliary buildings following a scenario 2 or 3 accident (see Chapter 10). The releases and resulting public doses from accident cleanup activities in these buildings

TABLE 14.2-1. Summary of Calculated Doses to the Maximum-Exposed Individual from Releases of Radioactivity from Routine Tasks During Accident Cleanup[a,b]

Accident Cleanup Activity	First-Year Dose (rem)			Fifty-Year Committed Dose Equivalent (rem)		
	Total-Body	Bone	Lung	Total-Body	Bone	Lung
Accident Scenario 1						
Preparations for Accident Cleanup	5.9×10^{-6}	5.9×10^{-6}	5.9×10^{-6}	5.9×10^{-6}	5.9×10^{-6}	5.9×10^{-6}
Initial Decontamination of Containment	1.4×10^{-3}	6.2×10^{-9}	1.4×10^{-3}	1.5×10^{-3}	9.0×10^{-9}	1.5×10^{-3}
Defueling the Reactor	1.3×10^{-3}	2.3×10^{-8}	1.3×10^{-3}	1.4×10^{-3}	2.3×10^{-8}	1.4×10^{-3}
Cleanup of Primary Coolant System	8.7×10^{-4}	6.9×10^{-11}	8.7×10^{-4}	9.3×10^{-4}	1.1×10^{-10}	9.3×10^{-4}
Waste Treatment and Packaging	5.6×10^{-4}	3.4×10^{-4}	1.8×10^{-4}	6.1×10^{-4}	4.0×10^{-4}	1.9×10^{-4}
Accident Scenario 2						
Preparations for Accident Cleanup	1.2×10^{-4}	1.2×10^{-4}	1.2×10^{-4}	1.2×10^{-4}	1.2×10^{-4}	1.2×10^{-4}
Initial Decontamination of Containment	7.3×10^{-3}	8.9×10^{-8}	7.3×10^{-3}	7.8×10^{-3}	4.7×10^{-7}	7.8×10^{-3}
Defueling the Reactor	6.3×10^{-3}	4.4×10^{-7}	6.3×10^{-3}	6.8×10^{-3}	4.4×10^{-7}	6.8×10^{-3}
Cleanup of Primary Coolant System	4.3×10^{-3}	9.8×10^{-10}	4.3×10^{-3}	4.7×10^{-3}	5.2×10^{-9}	4.7×10^{-3}
Waste Treatment and Packaging	6.2×10^{-3}	4.9×10^{-3}	1.6×10^{-3}	9.1×10^{-3}	1.4×10^{-2}	1.7×10^{-3}
Accident Scenario 3						
Preparations for Accident Cleanup	8.9×10^{-4}	8.9×10^{-4}	8.9×10^{-4}	8.9×10^{-4}	8.9×10^{-4}	8.9×10^{-4}
Initial Decontamination of Containment	1.4×10^{-2}	1.3×10^{-7}	1.4×10^{-2}	1.5×10^{-2}	8.8×10^{-7}	1.5×10^{-2}
Defueling the Reactor	1.3×10^{-2}	3.4×10^{-6}	1.3×10^{-2}	1.4×10^{-2}	3.4×10^{-6}	1.4×10^{-2}
Cleanup of Primary Coolant System	8.7×10^{-3}	1.5×10^{-9}	8.7×10^{-3}	9.3×10^{-3}	1.0×10^{-8}	9.3×10^{-3}
Waste Treatment and Packaging	8.5×10^{-3}	7.5×10^{-3}	1.7×10^{-3}	1.5×10^{-2}	2.7×10^{-2}	1.8×10^{-3}

(a) Summarized from Tables J.2-1 through J.2-3 of Appendix J.
(b) Doses are not totaled because they occur during different years. Doses to the maximum-exposed individual in any given year are estimated to be below the appropriate dose design objectives as set forth in 10 CFR 50, Appendix I.

TABLE 14.2-2. Summary of Calculated Doses to the Population from Releases of Radioactivity from Routine Tasks During Accident Cleanup[a,b]

Accident Cleanup Activity	First-Year Dose (man-rem)			Fifty-Year Committed Dose Equivalent (man-rem)		
	Total-Body	Bone	Lung	Total-Body	Bone	Lung
Accident Scenario 1						
Preparations for Accident Cleanup	7×10^{-3}	7×10^{-3}	7×10^{-3}	7×10^{-3}	7×10^{-3}	7×10^{-3}
Initial Decontamination of Containment	9×10^{-1}	4×10^{-6}	9×10^{-1}	1×10^{0}	7×10^{-6}	9×10^{-1}
Defueling the Reactor	8×10^{-1}	3×10^{-5}	8×10^{-1}	9×10^{-1}	3×10^{-5}	8×10^{-1}
Cleanup of Primary Coolant System	6×10^{-1}	5×10^{-9}	6×10^{-1}	6×10^{-1}	8×10^{-9}	6×10^{-1}
Waste Treatment and Packaging	9×10^{0}	5×10^{0}	3×10^{0}	9×10^{0}	6×10^{0}	3×10^{0}
Accident Scenario 2						
Preparations for Accident Cleanup	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-1}
Initial Decontamination of Containment	5×10^{0}	7×10^{-5}	5×10^{0}	5×10^{0}	4×10^{-4}	4×10^{0}
Defueling the Reactor	4×10^{0}	5×10^{-4}	4×10^{0}	4×10^{0}	5×10^{-4}	4×10^{0}
Cleanup of Primary Coolant System	3×10^{0}	7×10^{-6}	3×10^{0}	3×10^{0}	4×10^{-6}	3×10^{0}
Waste Treatment and Packaging	9×10^{1}	7×10^{1}	2×10^{1}	1×10^{2}	2×10^{2}	2×10^{1}
Accident Scenario 3						
Preparations for Accident Cleanup	1×10^{0}	1×10^{0}	1×10^{0}	1×10^{0}	1×10^{0}	1×10^{0}
Initial Decontmaination of Containment	9×10^{0}	1×10^{-4}	9×10^{0}	9×10^{0}	7×10^{-4}	9×10^{0}
Defueling the Reactor	8×10^{0}	4×10^{-3}	8×10^{0}	9×10^{0}	4×10^{-3}	8×10^{0}
Cleanup of Primary Coolant System	6×10^{0}	1×10^{-6}	6×10^{0}	6×10^{0}	8×10^{-6}	6×10^{0}
Waste Treatment and Packaging	1×10^{2}	1×10^{2}	3×10^{1}	2×10^{2}	4×10^{2}	3×10^{1}

(a) All doses rounded to 1 significant figure, summarized from Tables J.2-4 through J.2-6 of Appendix J.
(b) Doses are not totaled because they occur during different years.

are judged to be negligible in comparison with those from activities in the containment building and are not considered further.

14.2.1.2 Public Radiation Doses from Releases Due to Postulated Industrial Accidents During Accident Cleanup

The consequences of postulated industrial accidents that result in releases of radioactivity from the plant are determined by calculating the dose to the maximum-exposed individual. The industrial accident situations considered in this study are the same as those analyzed in Reference 1 for cleanup at TMI-2.

A summary of estimated doses to the maximum-exposed individual from postulated releases due to industrial accidents during accident cleanup is given in Table 14.2-3. First-year radiation doses and fifty-year committed radiation dose equivalents are listed for the lung of the maximum-exposed individual. The industrial accidents are listed in order of decreasing magnitude of release. The industrial accident that is postulated to result in the largest release is a liquid release to the river adjacent to the reference facility, and all other accidents considered involve atmospheric releases. The postulated industrial accident that results in the largest calculated doses to the maximum-exposed individual is a waste handling accident involving a spent ion-exchange liner from the accident-water cleanup demineralizer system. This results in a first-year dose of about 4 rem and a fifty-year dose of about 8 rem to the lung of the maximum-exposed individual for cleanup following a scenario 3 reactor accident. In general, the calculated doses from postulated industrial accidents for accident cleanup following a scenario 1 accident are 1 to 2 orders of magnitude below those for accident cleanup following a scenario 2 accident which, in turn, are about an order of magnitude less than those following a scenario 3 reactor accident. The transportation accident shown in the table for comparison purposes is discussed in Section 14.2.3.

14.2.2 Occupational Safety Impacts of Accident Cleanup

Occupational safety during accident cleanup at the reference PWR is evaluated both for radiation exposure to cleanup workers and for nonradiological industrial accidents.

TABLE 14.2-3. Summary of Maximum-Exposed-Individual Radiation Doses from Releases Due to Industrial Accidents During Post-Accident Cleanup[a]

Accident	Reference Radionuclide Inventory[b]	Accident Scenario 1: Total Release (Ci/hr)[c]	Accident Scenario 1: Radiation Dose to Lung (rem), First-Year	Accident Scenario 1: Radiation Dose to Lung (rem), Fifty-Year	Accident Scenario 2: Total Release (Ci/hr)[c]	Accident Scenario 2: Radiation Dose to Lung (rem), First-Year	Accident Scenario 2: Radiation Dose to Lung (rem), Fifty-Year	Accident Scenario 3: Total Release (Ci/hr)[c]	Accident Scenario 3: Radiation Dose to Lung (rem), First-Year	Accident Scenario 3: Radiation Dose to Lung (rem), Fifty-Year
Liquid Release to River	^{3}H	9.0×10^{2}	8.9×10^{-5}	8.9×10^{-5}	3.1×10^{3}	3.1×10^{-4}	3.1×10^{-4}	9.0×10^{3}	8.9×10^{-4}	8.9×10^{-4}
	AGFPC[d]	3.1×10^{0}	6.2×10^{-5}	6.8×10^{-5}	8.6×10^{0}	1.5×10^{-4}	2.9×10^{-4}	3.8×10^{1}	6.9×10^{-4}	7.6×10^{-4}
Release of Trapped Fission Products	^{85}Kr	6.2×10^{0}	2.0×10^{-6}	2.0×10^{-6}	1.2×10^{2}	3.8×10^{-5}	3.8×10^{-5}	9.3×10^{2}	3.0×10^{-4}	3.0×10^{-4}
Accident-Water Cleanup Demin. System Waste Handling	AGFPC	9.6×10^{-1}	2.3×10^{-3}	2.5×10^{-3}	1.2×10^{1}	4.1×10^{-1}	8.2×10^{-1}	8.2×10^{1}	3.9×10^{0}	8.0×10^{0}
Transportation Accident	AGFPC	9.6×10^{-2}	2.3×10^{-4}	2.5×10^{-4}	1.2×10^{0}	4.1×10^{-2}	8.2×10^{-2}	8.2×10^{0}	3.9×10^{-1}	8.0×10^{-1}
HEPA Failure During Liquid Waste Treatment	AGFPC	5.3×10^{-3}	1.3×10^{-5}	1.4×10^{-5}	6.6×10^{-2}	2.2×10^{-3}	4.5×10^{-3}	4.5×10^{-1}	2.2×10^{-2}	4.4×10^{-2}
Storage Area Fire	AGFPC	3.2×10^{-3}	7.7×10^{-6}	8.3×10^{-6}	4.0×10^{-2}	1.3×10^{-3}	2.7×10^{-3}	2.8×10^{-1}	1.3×10^{-2}	2.7×10^{-2}
Other Waste Handling	AGFPC	2.7×10^{-3}	6.5×10^{-6}	7.0×10^{-6}	3.4×10^{-2}	1.2×10^{-3}	2.3×10^{-3}	2.3×10^{-1}	1.1×10^{-2}	2.2×10^{-2}
Spill of Decontamination Liquids from RCS	AGFPC	1.6×10^{-4}	3.1×10^{-7}	4.2×10^{-7}	2.0×10^{-3}	6.8×10^{-5}	1.4×10^{-4}	1.4×10^{-2}	6.7×10^{-4}	1.4×10^{-3}
HEPA Filter Failure	AGFPC	7.6×10^{-5}	1.8×10^{-7}	2.0×10^{-7}	9.4×10^{-4}	3.2×10^{-5}	6.4×10^{-5}	6.4×10^{-3}	3.1×10^{-4}	6.2×10^{-4}
Solid Waste Handling	AGFPC	2.2×10^{-5}	5.3×10^{-8}	5.7×10^{-8}	2.7×10^{-4}	9.2×10^{-6}	1.8×10^{-5}	1.8×10^{-3}	8.6×10^{-5}	1.7×10^{-4}
Chemical Decontamination Waste Handling	AGFPC	8.0×10^{-8}	1.9×10^{-10}	2.1×10^{-10}	1.0×10^{-6}	3.4×10^{-8}	6.8×10^{-8}	7.0×10^{-6}	3.4×10^{-7}	6.8×10^{-7}

(a) Summarized from Tables J.2-7 through J.2-9 of Appendix J.
(b) Reference radionuclide inventories discussed in Appendix C.
(c) All releases assumed to occur during a 1-hr period, for comparison purposes.
(d) Accident-Generated Fission-Product Contamination, see Appendix C.

Estimates of occupational radiation doses are based on the postulated radiation dose rates in various areas of the facility and on the estimated staff labor required to complete the accident cleanup activities. A summary of the detailed information given in Appendices E and J is given in this section. This section also contains estimates of worker injuries and fatalities resulting from industrial accidents during the accident cleanup effort. These casualty estimates are based on nuclear industry experience.

14.2.2.1 Occupational Radiation Doses from Accident Cleanup Activities

A summary of the estimated occupational radiation doses for accident cleanup following a scenario 1, scenario 2, or scenario 3 reactor accident is given in Table 14.2-4. The table contains a listing of the major accident cleanup activities and the associated estimated total external radiation doses to the workers.

TABLE 14.2-4. Summary of Estimated Occupational Radiation Doses from Accident Cleanup of the Reference PWR[a]

Cleanup Activity	Estimated Total Occupational Doses[b] (man-rem) Following:		
	Accident Scenario 1	Accident Scenario 2	Accident Scenario 3
Planning and Preparations	4	45	360
Cleanup of Auxiliary and Fuel Buildings	--[c]	1 612	1 612
Processing of Contaminated Liquids from Containment Building	43	144	288
Initial Decontamination of Containment Building	206	750	1 925
Defueling of the Reactor	197	1 010	3 913
Cleanup of the Primary Coolant System	32	140	540
Support Operations	189	877	3 464
Totals	670	4 579	12 103

(a) Summarized from Tables J.2-10 through J.2-12 of Appendix J, individually rounded to the nearest whole man-rem.
(b) Doses shown are external doses from gamma radiation; workers are assumed to use respiration equipment as appropriate to protect against inhalation of radioactive materials.
(c) Not postulated to be required following a scenario 1 accident.

The radiation doses to accident cleanup workers are calculated as the product of the estimated radiation zone manpower requirements and the external radiation dose rates postulated for each specific area. The occupational dose estimates are based on the following basic assumptions: 1) personnel exposure to radiation is minimized by using temporary shielding, remote handling techniques, and respiration equipment as appropriate, and by keeping workers not actively engaged in a task out of the radiation fields, 2) decontamination efforts are reasonably successful in reducing radiation dose rates, 3) careful, prompt accounting of radiation doses is maintained to rapidly identify jobs that are causing excessive dose accumulations so that corrective action can be taken, and 4) ^{137}Cs is the dominant radioactive species contributing to occupational dose. Although the radioactive materials that are the source of the doses decay throughout the accident cleanup period, no credit is taken for this decay because the calculated effect is minimal due to the 30-year half-life of the dominant radionuclide (^{137}Cs).

The total estimated occupational radiation doses during accident cleanup at the reference PWR are 670 man-rem following a postulated scenario 1 accident, about 4580 man-rem following a scenario 2 accident, and over 12,100 man-rem following a scenario 3 accident. Accident cleanup worker requirements are adjusted in this study as necessary to ensure compliance with limitations on individual radiation dose accumulations of 5 man-rem/man-year.(6) The results presented here do not include the radiation doses to transportation workers, which are given in Section 14.2.3.

14.2.2.2 Industrial Safety Impacts of Accident Cleanup

The industrial safety impacts of accident cleanup include potential injuries and fatalities resulting from industrial accidents to cleanup workers. Estimated casualties are calculated by finding the products of a) the frequencies of injuries and fatalities during various categories of work and b) the estimated worker time applied to each work category. The estimated worker injuries and fatalities during accident cleanup at the reference PWR following each of the three postulated reactor accidents considered in this study are summarized in Table 14.2-5. As shown in the table, less than 1 injury is estimated for accident cleanup following a scenario 1 accident, about 1 injury is estimated following a scenario 2 accident, and about 2 injuries are estimated following a scenario 3 accident. Fatalities resulting from industrial accidents appear unlikely during accident cleanup.

TABLE 14.2-5. Summary of Estimated Occupational Lost-Time Injuries and Fatalities During Accident Cleanup[a]

Accident-Potential Category	Accident Scenario 1 Lost-Time Injuries	Accident Scenario 1 Fatalities	Accident Scenario 2 Lost-Time Injuries	Accident Scenario 2 Fatalities	Accident Scenario 3 Lost-Time Injuries	Accident Scenario 3 Fatalities
Light Construction	0.19	1.1×10^{-3}	0.86	4.8×10^{-3}	1.5	8.7×10^{-3}
Operational Support	0.11	1.2×10^{-3}	0.42	4.6×10^{-3}	0.57	6.2×10^{-3}
Totals	0.30	2.3×10^{-3}	1.3	9.4×10^{-3}	2.1	1.5×10^{-2}

(a) Summarized from Table J.2-13 of Appendix J.

14.2.3 Transportation Safety Impacts of Accident Cleanup

Spent reactor fuel from the reactor defueling and radioactive wastes collected during accident cleanup are assumed to be shipped offsite as part of planned accident cleanup tasks. Spent fuel and fuel core debris are assumed to be shipped by rail in an IF-300 cask to an ISFSI or a federal respository located 1600 km from the reference PWR site. Radioactive waste materials are assumed to be shipped by exclusive-use truck to either a shallow-land burial ground or a federal respository, both of which are assumed to be 1600 km from the site. The safety impacts of transportation activities considered in this study include radiation doses to transport workers and to the public along the transport routes, radiation doses to the maximum-exposed individual from atmospheric releases during transportation accidents, and injuries and fatalities resulting from potential transportation accidents. Radiation doses received by workers unloading the radioactive materials at a repository or disposal site are not estimated in this study, since they are assumed to occur at a separate licensed facility.

14.2.3.1 Radiation Doses from Routine Transportation Activities During Accident Cleanup

To calculate radiation doses to transport workers and to the public from transportation activities, a number of assumptions are made concerning radiation exposure rates from the shipments and exposure times for the various individuals involved. These assumptions are detailed in Section J.2.3.1 of

Appendix J. The primary assumption is that each shipment contains enough radioactive material to result in the maximum exposure rates allowed by Department of Transportation regulations.

The estimated radiation doses from transportation activities during accident cleanup are listed in Table 14.2-6. The values in the table include both truck and rail shipments, and the numbers of each type of shipment required following each accident scenario are given. The total estimated doses from transportation activities following a scenario 3 reactor accident are 99 man-rem to transport workers and 9.6 man-rem to members of the public along the transportation routes. The corresponding doses following a scenario 1 or a scenario 2 accident are about 1/6 and 1/2, respectively, of those following a scenario 3 accident.

TABLE 14.2-6. Summary of Estimated Radiation Doses from Transportation Activities During Accident Cleanup[(a)]

Activity/Group	Accident Scenario 1 Number of Shipments	Accident Scenario 1 Radiation Dose (man-rem)	Accident Scenario 2 Number of Shipments	Accident Scenario 2 Radiation Dose (man-rem)	Accident Scenario 3 Number of Shipments	Accident Scenario 3 Radiation Dose (man-rem)
Truck Shipments						
Truck Drivers	90	13	289	40	652	91
Garagemen	90	0.60	289	1.9	652	4.4
Total Trucking Worker Dose		14		42		95
Onlookers	90	0.90	289	2.9	652	6.5
General Public	90	0.33	289	1.1	652	2.4
Total Public Dose from Truck Shipments		1.2		4.0		8.9
Rail Shipments						
Train Brakemen	30	2.5	50	4.2	52	4.3
Onlookers	30	0.30	50	0.50	52	0.52
General Public	30	0.11	50	0.19	52	0.19
Total Public Dose from Rail Shipments		0.41		0.69		0.71
Total						
Total Transport Worker Dose		17		46		99
Total Public Dose from Transportation		1.6		4.7		9.6

(a) Summarized from Tables J.2-14 and J.2-15 of Appendix J; all doses rounded to two significant figures, based on one-way trips of 1600 km.

14.2.3.2 Radiation Doses from Postulated Transportation Accidents During Accident Cleanup

Transportation accidents during the offsite shipment of radioactive materials from accident cleanup can potentially result in inadvertent releases of radioactivity and corresponding radiation doses to individuals near the accident location. For this study, a realistic "worst-case" accident during truck transport is analyzed based on information in Section 10.4 of Reference 1, with appropriate adjustments to account for the differences between the post-accident radionuclide inventories at TMI-2 and at the reference PWR. It is assumed that a Type B container is broken open after which there is a fire, releasing 10^{-5} of the contained radioactivity. The resulting releases and doses from such an accident following each of the three reactor accident scenarios considered in this study are shown previously in Table 14.2-3. The fifty-year committed dose equivalent to the lung of the maximum-exposed individual for this transportation accident during scenario 3 accident cleanup is 800 mrem, with corresponding doses during scenario 1 or scenario 2 accident cleanup estimated to be lower by about 3 orders of magnitude or 1 order of magnitude, respectively. Less severe impacts would result from a transportation accident involving a Type A package; these impacts would be similar to those for such an accident during decommissioning, as discussed in Section 14.3.3.2.

Since the transportation of spent fuel is not unique to accident cleanup and decommissioning, and since the probabilities of accidents that could lead to atmospheric releases of radionuclides from spent fuel casks are very low, no analysis of spent fuel transport accidents is performed in this study. A discussion of the impact of spent fuel transport accidents on public safety is given in Reference 7.

14.2.3.3 Nonradiological Safety Impacts of Transportation Activities During Accident Cleanup

As with any transportation task, a certain potential for accidental injury or death exists from transportation accidents during accident cleanup activities at the reference PWR.[8] A summary of the casualties estimated to result during transportation activities for accident cleanup is shown in Table 14.2-7. As shown in the table, about 1.1 injuries and 0.066 fatalities are estimated for accident cleanup transportation activities following a scenario 3 accident; the corresponding values following a scenario 1 or scenario 2 accident are estimated to be lower by factors of about 6 or 2, respectively. Casualties from truck transport are estimated to be much greater than those from rail transport because of the greater number of truck shipments required and because of the greater frequency of casualties during truck transport.

TABLE 14.2-7. Summary of Estimated Casualties from Transportation Accidents During Accident Cleanup[a]

Transportation Category	Accident Scenario 1			Accident Scenario 2			Accident Scenario 3		
	Number of Shipments	Injuries	Fatalities	Number of Shipments	Injuries	Fatalities	Number of Shipments	Injuries	Fatalities
Rail Transport	30	0.023	0.0017	50	0.038	0.0028	52	0.040	0.0030
Truck Transport	90	0.15	0.0087	289	0.47	0.028	652	1.1	0.063
Totals	120	0.17	0.010	339	0.51	0.30	704	1.1	0.066

(a) Summarized from Table J.2-16 of Appendix J, estimates rounded to two significant figures, based on 3200-km round trip per shipment.

14.3 DECOMMISSIONING SAFETY

Decommissioning activities at the reference PWR follow completion of the accident cleanup activities. The detailed analysis of the safety impacts resulting from post-accident decommissioning activities at the reference PWR, presented in Section J.3 of Appendix J, is summarized in this section. Radiological safety impacts to the public are described in Section 14.3.1, occupational safety impacts are described in Section 14.3.2, and transportation safety impacts are addressed in Section 14.3.3.

14.3.1 Public Safety Impacts of Post-Accident Decommissioning

The public radiological safety impacts of onsite activities during post-accident decommissioning are discussed in this section. Nonradiological safety impacts to the public from onsite activities are judged to be negligible and are not considered further. Public safety impacts from offsite transportation activities during decommissioning are included in Section 14.3.3.

The consequences of atmospheric releases of radioactivity from routine tasks during post-accident decommissioning are determined by calculating radiation doses to the maximum-exposed individual and to the population residing within 80 km of the site. Radiation exposure pathways considered for these releases are direct external exposure, inhalation, and ingestion of food products. The consequences of postulated industrial accidents that release radioactivity from the plant are determined by calculating inhalation radiation doses to the maximum-exposed individual.

As discussed previously in Section 14.1, the releases and resulting doses for post-accident decommissioning at the reference PWR are based largely on the releases and doses calculated for normal-shutdown decommissioning of the facility, as reported in Reference 2, with appropriate adjustments made to account for differences between post-accident and normal-shutdown radionuclide inventories and decommissioning requirements in the plant.

14.3.1.1 Public Radiation Doses from Routine Tasks During Post-Accident Decommissioning

As it is during accident cleanup, loss of confinement of radioactive materials resulting in public radiation exposure is a primary safety concern during decommissioning. The radiation doses from these releases are calculated using the dose models discussed in Appendix E of Reference 2 in conjunction with the appropriate radionuclide inventories as shown in Reference 2 and in Appendix C of this study.

Tables 14.3-1 and 14.3-2 contain summaries of the calculated radiation doses to the maximum-exposed individual and to the population residing within 80 km of the site as a result of releases from routine tasks during post-accident decommissioning following a scenario 2 accident and the subsequent accident cleanup campaign. Data are presented in the tables for releases during DECON and during preparations for safe storage (the first phase of SAFSTOR). The releases and resulting doses from ENTOMB are not shown, but are judged to be almost the same as and only slightly less than those from DECON (see Section J.3.1.1 of Appendix J). Listed in the tables are the first-year radiation dose and fifty-year committed radiation dose equivalent to total

TABLE 14.3-1. Summary of Calculated Doses to the Maximum-Exposed Individual from Releases of Radioactivity During Routine Decommissioning Tasks Following a Scenario 2 Accident[(a)]

Decommissioning Activity	First-Year Dose (rem) Total-Body	Bone	Lung	Fifty-Year Committed Dose Equivalent (rem) Total-Body	Bone	Lung
DECON						
Segmentation of Nonactivated Stainless	1.7×10^{-8}	1.6×10^{-8}	4.3×10^{-8}	1.7×10^{-8}	1.6×10^{-8}	5.8×10^{-8}
Segmentation of Activated Components	1.2×10^{-10}	1.2×10^{-10}	2.8×10^{-10}	1.2×10^{-10}	1.6×10^{-10}	3.9×10^{-10}
Waste Handling Bioshield Concrete	4.7×10^{-11}	5.1×10^{-10}	8.0×10^{-11}	1.1×10^{-9}	1.1×10^{-9}	1.6×10^{-9}
Surface Cleaning Operations	2.6×10^{-8}	2.4×10^{-8}	1.3×10^{-8}	5.1×10^{-8}	1.3×10^{-7}	2.8×10^{-8}
In-Situ Chemical Decontamination	3.3×10^{-9}	3.1×10^{-9}	1.7×10^{-9}	6.7×10^{-9}	1.6×10^{-8}	3.5×10^{-9}
Removal of Bioshield	1.4×10^{-13}	1.5×10^{-12}	2.4×10^{-13}	3.4×10^{-12}	3.2×10^{-12}	4.7×10^{-12}
Radiation Survey	1.3×10^{-9}	1.2×10^{-9}	6.5×10^{-10}	2.6×10^{-9}	6.4×10^{-9}	1.3×10^{-9}
Removal of Concrete Areas	5.6×10^{-14}	5.4×10^{-14}	2.4×10^{-14}	1.1×10^{-12}	2.9×10^{-13}	6.0×10^{-14}
Totals	4.8×10^{-8}	4.5×10^{-8}	5.9×10^{-8}	7.8×10^{-8}	1.7×10^{-7}	9.2×10^{-8}
Preparations for Safe Storage						
Surface Cleaning Operations	1.1×10^{-8}	1.1×10^{-8}	6.7×10^{-9}	2.3×10^{-8}	5.8×10^{-8}	1.2×10^{-8}
In-Situ Chemical Decontamination	3.3×10^{-9}	3.1×10^{-9}	1.7×10^{-9}	6.7×10^{-9}	1.6×10^{-8}	3.5×10^{-9}
Radiation Survey	1.3×10^{-9}	1.2×10^{-9}	6.5×10^{-10}	2.6×10^{-9}	6.2×10^{-9}	1.3×10^{-9}
Onsite Retrievable Waste Storage	7.1×10^{-8}	6.7×10^{-8}	3.4×10^{-8}	1.4×10^{-7}	3.6×10^{-7}	7.5×10^{-8}
Totals	8.7×10^{-8}	8.2×10^{-8}	4.3×10^{-8}	1.7×10^{-7}	4.4×10^{-7}	9.2×10^{-8}

(a) Summarized from Tables J.3-1 and J.3-2 of Appendix J.

TABLE 14.3-2. Summary of Calculated Doses to the Population from Releases of Radioactivity During Routine Decommissioning Tasks Following a Scenario 2 Accident[a]

Decommissioning Activity	First-Year Dose (man-rem)			Fifty-Year Committed Dose Equivalent (man-rem)		
	Total-Body	Bone	Lung	Total-Body	Bone	Lung
DECON						
Segmentation of Nonactivated Stainless	1×10^{-5}	9×10^{-6}	4×10^{-5}	1×10^{-5}	9×10^{-6}	6×10^{-5}
Segmentation of Activated Components	7×10^{-8}	7×10^{-8}	2×10^{-7}	7×10^{-8}	9×10^{-8}	3×10^{-7}
Waste Handling Bioshield Concrete	2×10^{-7}	6×10^{-7}	3×10^{-7}	1×10^{-6}	1×10^{-6}	2×10^{-6}
Surface Cleaning Operations	1×10^{-5}	1×10^{-5}	1×10^{-5}	4×10^{-5}	1×10^{-5}	3×10^{-5}
In-Situ Chemical Decontamination	2×10^{-6}	2×10^{-6}	2×10^{-6}	5×10^{-6}	1×10^{-5}	3×10^{-6}
Removal of Bioshield	7×10^{-10}	2×10^{-9}	8×10^{-10}	3×10^{-9}	3×10^{-9}	6×10^{-9}
Radiation Survey	1×10^{-6}	1×10^{-6}	5×10^{-7}	2×10^{-6}	5×10^{-6}	1×10^{-6}
Removal of Concrete Areas	4×10^{-11}	4×10^{-11}	2×10^{-11}	8×10^{-11}	2×10^{-10}	6×10^{-11}
Totals	2×10^{-5}	2×10^{-5}	5×10^{-5}	6×10^{-5}	4×10^{-5}	1×10^{-4}
Preparations for Safe Storage						
Surface Cleaning Operations	5×10^{-6}	5×10^{-6}	3×10^{-6}	9×10^{-6}	3×10^{-5}	7×10^{-6}
In-Situ Chemical Decontamination	2×10^{-6}	2×10^{-6}	1×10^{-6}	5×10^{-6}	1×10^{-5}	3×10^{-6}
Radiation Survey	1×10^{-6}	1×10^{-6}	5×10^{-7}	2×10^{-6}	5×10^{-6}	1×10^{-6}
Onsite Retrievable Waste Storage	5×10^{-5}	5×10^{-5}	3×10^{-5}	1×10^{-4}	3×10^{-4}	7×10^{-5}
Totals	6×10^{-5}	6×10^{-5}	3×10^{-5}	1×10^{-4}	3×10^{-4}	8×10^{-5}

(a) All doses rounded to 1 significant figure, summarized from Tables J.3-3 and J.3-4 of Appendix J.

body, bone, and lung. The corresponding doses following a scenario 1 reactor accident are a factor of about 2 to 5 less than those shown in the tables and those following a scenario 3 accident are greater than those in the tables by a similar factor, because of the decrease or increase in the accident generated fission-product contamination in the plant following these accidents.

The public radiation doses from routine tasks during post-accident decommissioning are estimated to be small by comparison to the range of annual radiation doses to an individual from natural background in the United States (from 80 to 170 mrem per year).[4] Furthermore, these estimated radiation doses are several orders of magnitude below the permissible levels of radiation in unrestricted areas as set forth in 10 CFR Part 50, Appendix I.[5]

The release of radionuclides during continuing care is expected to be negligible compared to the release during preparations for safe storage.[2] This is because of the rugged construction of the facility, the erection of barriers to radionuclide migration, and the limited human contact during surveillance and maintenance operations. Thus, no public radiation doses are calculated for continuing care. Similarly, since the calculated radiation doses for DECON are small, and since the radioactivity levels are reduced by radioactive decay during continuing care, public radiation doses for deferred decontamination are expected to be insignificant and are not calculated.

14.3.1.2 Public Radiation Doses from Releases Due to Postulated Industrial Accidents During Post-Accident Decommissioning

The consequences of postulated industrial accidents that result in releases of radioactivity from the facility during decommissioning are determined by calculating the dose to the maximum-exposed individual. The industrial accident situations considered in this study are the same as those analyzed in Reference 2 for decommissioning of the reference PWR following normal reactor shutdown. While it is beyond the scope of this study to evaluate every potential industrial accident situation during decommissioning, the postulated situations presented here are judged to represent the range of credible events and to reflect realistic maximum impacts from such situations to the public.

A summary of the estimated radiation doses to the maximum-exposed individual from releases due to postulated industrial accidents during post-accident decommissioning is given in Table 14.3-3. The accidents are listed in order of decreasing magnitude of release. First-year radiation doses and fifty-year committed radiation dose equivalents are listed for the lung of the maximum-exposed individual, for either DECON or preparations for safe storage following a scenario 2 reactor accident. The releases and corresponding doses for ENTOMB are assumed to be the same as those shown for DECON, with the deletion of those situations that arise from activities not undertaken during ENTOMB (e.g., blasting, segmenting of the reactor pressure vessel). The postulated accident that results in the largest calculated release and the largest doses to the maximum-exposed individual is the explosion of liquid petroleum gas (LPG) leaked from a front-end loader. It is calculated that 1.8 x 10^{-2} curies of accident-generated fission-product contamination could be released, resulting in a first-year dose of about 0.6 mrem and a fifty-year committed dose equivalent of about 1.2 mrem to the lung of the maximum-exposed individual. The calculated doses from releases due to postulated industrial accidents during post-accident decommissioning are more than 3 orders of magnitude below those during the accident cleanup campaign that precedes the decommissioning. This is due primarily to the marked reduction in radioactive contamination levels in the facility accomplished during the accident cleanup campaign.

14.3.2 Occupational Safety Impacts of Post-Accident Decommissioning

Occupational safety during post-accident decommissioning at the reference PWR is evaluated both for radiation exposure and for nonradiological industrial accidents.

Estimates of occupational radiation doses are based on the postulated radiation dose rates in various areas of the facility following the completion of accident cleanup and on the estimated staff labor required to complete the

TABLE 14.3-3. Summary of Maximum-Exposed Individual Radiation Doses from Releases Due to Postulated Industrial Accidents During Post-Accident Decommissioning[(a)]

Accident	Reference Radionuclide Inventory[(b)]	Total Release (μCi/hr)[(c)]	Radiation Dose to Lung (rem) During: DECON First-Year	DECON Fifty-Year	Preparations for Safe Storage First-Year	Preparations for Safe Storage Fifty-Year
Explosion of LPG Leaked from Loader	AGFPC[(d)]	1.8×10^{4}	6.1×10^{-4}	1.2×10^{-3}	--[(e)]	--
Explosion of Oxyacetylene During Vessel Segmentation	2	3.6×10^{2}	6.1×10^{-6}	6.9×10^{-6}	--	--
Explosion/Fire of Ion Exchange Resin	AGFPC	1.9×10^{2}	6.5×10^{-6}	1.3×10^{-5}	--	--
Gross Leak During Decontamination - Spray Leak	AGFPC	1.1×10^{2}	3.8×10^{-6}	7.5×10^{-6}	3.8×10^{-6}	7.5×10^{-6}
- Liquid Leak	AGFPC	3.5×10^{-1}	1.2×10^{-8}	2.4×10^{-8}	1.2×10^{-8}	2.4×10^{-8}
Segmenting Undecontaminated RCS Piping	4	1.1×10^{1}	7.3×10^{-7}	7.9×10^{-7}	--	--
Vacuum Bag Rupture	AGFPC	5.0×10^{0}	--	--	1.7×10^{-7}	3.4×10^{-7}
Loss of Contamination Control During Vessel Segmentation	2	2.3×10^{0}	3.9×10^{-10}	4.4×10^{-8}	--	--
Accidental Spraying of Concentrated Contamination with High-Pressure Spray	AGFPC	6.0×10^{-1}	--	--	2.0×10^{-8}	4.1×10^{-8}
Filter Loss During Blasting of Concrete Bioshield	3	3.0×10^{-1}	2.0×10^{-9}	2.2×10^{-9}	--	--
Loss of Portable Filtered Ventilation Enclosure	AGFPC	1.5×10^{-1}	5.1×10^{-9}	1.0×10^{-8}	--	--
Accidental Break of Contaminated Piping	4	1.1×10^{-1}	--	--	7.3×10^{-9}	7.9×10^{-9}
Fire Involving Combustible Radioactive Wastes	AGFPC	3.0×10^{-2}	1.0×10^{-9}	2.0×10^{-9}	1.0×10^{-9}	2.0×10^{-9}

(a) Summarized from Table J.3-5 and J.3-6 of Appendix J, events resulting in doses $<10^{-9}$ rem not shown.
(b) Reference radionuclide inventories discussed in Reference 2.
(c) All releases assumed to occur during a 1-hr period, for comparison purposes.
(d) Accident-Generated Fission-Product Contamination, see Appendix C.
(e) A dash indicates the particular accident situation is not considered for that decommissioning alternative because either the accident situation does not apply to that alternative or a similar accident of greater consequences is analyzed.

decommissioning. A summary of the detailed occupational dose information developed in Appendices G and J is given in this section. Estimates of worker injuries and fatalities resulting from industrial accidents during the decommissioning effort are also presented in this section. These casualty estimates are based on nuclear industry experience. The information presented in this section is developed on a consistent basis with the corresponding information for accident cleanup that is presented previously in Section 14.2.2.

14.3.2.1 Occupational Radiation Doses from Post-Accident Decommissioning Activities

A summary of the estimated occupational radiation doses from external exposure to gamma radiation during DECON, preparations for safe storage, and ENTOMB at the reference PWR, following a scenario 2 accident and the subsequent accident cleanup, is given in Table 14.3-4. The table contains a listing of the major areas being decommissioned and the associated estimated total external radiation doses to the decommissioning workers. The radiation

TABLE 14.3-4. Summary of Estimated Occupational Radiation Doses from Decommissioning of the Reference PWR Following a Scenario 2 Accident[(a)]

Area Being Decommissioned	Estimated Total Occupational Doses[(b)] (man-rem) During: DECON	Preparations for Safe Storage	ENTOMB
Containment Building	2580	172	2049
Fuel Building	26	10	26
Auxiliary Building	96	20	96
Ancillaries	361	227	347
Totals	3063	429	2518

(a) Summarized from Tables J.3-7 through J.3-9 of Appendix J; individually rounded to nearest whole man-rem.
(b) Doses shown are external doses from gamma radiation; workers are assumed to use respiration equipment as appropriate to protect against inhalation of radioactive materials.

doses to decommissioning workers are calculated in the same manner as the doses to accident cleanup workers, as discussed previously in Section 14.2.2.1.

The total estimated occupational radiation doses during DECON are over 3060 man-rem, almost 430 man-rem during preparations for safe storage, and nearly 2520 man-rem during ENTOMB. In general, the bulk of the occupational radiation dose from decommissioning results from activities in the containment building. The doses shown in the table are estimated to be increased by a factor of 2 to 3 following a scenario 3 accident and to be reduced by a similar factor following a scenario 1 accident. The results presented here do not include the radiation doses to transport workers, which are discussed in Section 14.3.3.

Except for DECON, which on completion results in the unrestricted release of the facility, the initial phase of decommissioning is followed by a period of continuing care. During this continuing care period, the surveillance and maintenance staff is exposed to the residual radiation levels present in the facility, which continually decline by radioactive decay. The dominant isotope during continuing care is again assumed to be ^{137}Cs. Radioactive decay also reduces the radiation doses received by workers during deferred decontamination that follows continuing care.

Table 14.3-5 contains a summary of the estimated total occupational radiation doses during all phases of SAFSTOR, for assumed continuing care periods of 30 and 100 years. Total occupational doses from SAFSTOR are about 2050 man-rem with 30 years of continuing care and about 950 man-rem with 100 years of continuing care.

No detailed estimate is developed in this study for the occupational doses during continuing care and deferred decontamination following ENTOMB. However, because the level of effort required during continuing care following ENTOMB is anticipated to be about half that during continuing care for SAFSTOR, the occupational doses accumulated during continuing care for ENTOMB are assumed to be about half of those accumulated during continuing care for SAFSTOR. Thus, for example, the total occupational radiation doses during 100 years of continuing care following ENTOMB are assumed to be about 110 man-rem,

TABLE 14.3-5. Summary of Estimated Occupational Radiation Doses During All Phases of SAFSTOR

SAFSTOR Phase	Estimated Occupational Doses (man-rem) with Safe Storage Period of: 30 Years	100 Years
Preparations for Safe Storage[a]	429	429
Continuing Care[b]	120	225
Deferred Decontamination[b]	1500	300
Totals	2049	954

(a) From Table 14.3-4.
(b) From Section G.3.5 of Appendix G.

assuming the facility has experienced a scenario 2 reactor accident. It is further assumed that deferred decontamination following ENTOMB is similar in level of effort and in occupational radiation dose to deferred decontamination for SAFSTOR. Therefore, following 100 years of continuing care, deferred decontamination would result in occupational radiation doses of about 300 man-rem. Based on these assumptions, the total occupational radiation doses resulting from ENTOMB at the reference PWR following a scenario 2 accident are expected to be about 2930 man-rem, or nearly the same as the total doses for DECON, based on a retention period of 100 years for the entombment structure.

The estimates of occupational radiation dose presented here are sensitive to management philosophy and to the decommissioning methods used. Different basic assumptions, changes in decommissioning procedures, or staffing variations may result in significant changes in the expected occupational radiation doses.

14.3.2.2 Industrial Safety Impacts of Post-Accident Decommissioning

The industrial safety impacts of post-accident decommissioning include potential injuries and fatalities from industrial accidents to the decommissioning workers. Estimated casualties during decommissioning are calculated in the same manner as for accident cleanup. Table 14.3-6 contains a summary of the estimated worker injuries and fatalities from industrial accidents

TABLE 14.3-6. Summary of Estimated Occupational Lost-Time Injuries and Fatalities During Decommissioning Following a Scenario 2 Accident[(a)]

Accident-Potential Category	DECON Lost-Time Injuries	DECON Fatalities	Preparations for Safe Storage Lost-Time Injuries	Preparations for Safe Storage Fatalities	ENTOMB Lost-Time Injuries	ENTOMB Fatalities
Heavy Construction	0.31	1.3×10^{-3}	--[(b)]	--	0.31	1.3×10^{-3}
Light Construction	0.36	2.0×10^{-3}	0.10	5.7×10^{-4}	0.36	2.0×10^{-3}
Operational Support	0.12	1.3×10^{-3}	0.034	3.7×10^{-4}	0.11	1.2×10^{-3}
Totals	0.79	4.6×10^{-3}	0.13	9.4×10^{-4}	0.78	4.5×10^{-3}

(a) Summarized from Table J.3-11 of Appendix J.
(b) No activities in this category performed during preparations for safe storage.

during DECON, preparations for safe storage, and ENTOMB at the reference PWR following a scenario 2 reactor accident. As shown in the table, nearly 1 lost-time injury is estimated during either DECON or ENTOMB, and less than 1 injury is estimated during preparations for safe storage. Fatalities from industrial accidents appear unlikely during post-accident decommissioning.

Estimates of the number of injuries and fatalities from industrial accidents that could occur during continuing care at the reference PWR are not made because these impacts during continuing care are expected to be considerably smaller than the already minor impacts calculated for the initial decommissioning phases. Casualty estimates for deferred decontamination are expected to be similar to those for DECON, because of the similarity in the requirements for deferred decontamination and for DECON, and no further estimates for deferred decontamination are made.

14.3.3 Transportation Safety Impacts of Post-Accident Decommissioning

Radioactive wastes generated during post-accident decommissioning are assumed to be shipped offsite to a shallow-land burial site 1600 km from the facility. Exclusive-use trucks are used for these waste shipments. The safety impacts of transportation activities include radiation doses to transport workers and to the public along the transport routes, radiation doses to the maximum-exposed individual from atmospheric releases during transportation accidents, and injuries and fatalities resulting from potential transportation accidents. Radiation doses received by workers unloading the radioactive materials at the disposal site are not estimated, since they are assumed to occur at a separate licensed facility.

14.3.3.1 Radiation Doses from Routine Transportation Activities During Decommissioning

The assumptions made to estimate radiation doses to transport workers and to the public from exposure to radioactive shipments during decommissioning are the same as those for accident cleanup, as described previously in Section 14.2.3.1. The primary assumption is that each shipment contains enough radioactive material to result in the maximum exposure rates allowed by Department of Transportation regulations.

The estimated radiation doses from transportation activities during DECON, preparations for safe storage, and ENTOMB at the reference PWR following a scenario 2 reactor accident are listed in Table 14.3-7. Data presented in the table include the number of truck shipments required for each decommissioning alternative and the resulting doses to the exposed groups. The estimated total doses from transportation activities during DECON are about 200 man-rem to transport workers and about 19 man-rem to members of the public along the transportation route. The corresponding doses during preparations for safe storage and during ENTOMB are about 7% and 45%, respectively, of those during DECON. The largest calculated doses occur during DECON because this alternative requires more waste shipments than either of the other two decommissioning alternatives.

TABLE 14.3-7. Summary of Estimated Radiation Doses from Truck Transportation Activities During Post-Accident Decommissioning[a]

Group	DECON		Preparations for Safe Storage		ENTOMB	
	Number of Shipments	Radiation Dose (man-rem)	Number of Shipments	Radiation Dose (man-rem)	Number of Shipments	Radiation Dose (man-rem)
Truck Drivers	1352	190	86	12	613	86
Garagemen	1352	9.1	86	0.58	613	4.1
Total Transport Worker Dose		200		13		90
Onlookers	1352	14	86	0.86	613	6.1
General Public	1352	5.0	86	0.32	613	2.3
Total Public Dose from Transportation		19		1.2		8.4

(a) Summarized from Table J.3-12 of Appendix J; all doses rounded to two significant figures, based on one-way trips of 1600 km.

No specific estimate is made of the radiation doses that result from transportation activities during deferred decontamination. However, based on the decay of ^{137}Cs, the dominant radionuclide in the post-accident inventory, these doses following 30 years of continuing care are anticipated to be about one-half of those estimated for DECON; following 100 years of continuing care, these doses from deferred decontamination transportation activities are anticipated to be about one-tenth of those estimated for DECON.

14.3.3.2 Radiation Doses from Postulated Transportation Accidents During Decommissioning

Transportation accidents during the offsite shipment of radioactive materials from decommissioning can potentially result in inadvertent releases of radioactivity and corresponding radiation doses to individuals near the accident location. The methods used to estimate these potential doses are discussed in Section J.3.3.2 of Appendix J. Because the radioactive materials that are transported in Type B packages during decommissioning are in solid, noncombustible forms that are unlikely to become airborne in an accident, no accident anal ype B packages is considered. Instead, two more realistic accider combustible radioactive wastes in Type A packages are defined, bot an expected low frequency of occurrence. Waste packages of 1 curie each are assumed to rupture and burn. The minor accident is assumed to involve 1 such package and the severe accident 40 such packages. A release fraction of 5×10^{-4} is assumed.

The estimated radiation doses to the total body, bone, and lung of the maximum-exposed individual as a result of these postulated transportation accidents are shown in Table 14.3-8. Both first-year radiation doses and fifty-year committed radiation dose equivalents are included. The severe accident results in an estimated fifty-year committed dose equivalent of 0.19 rem to the bone of the maximum-exposed individual and 0.064 rem to the lung. Doses from the minor accident are a factor of 40 less than those from the severe accident.

TABLE 14.3-8. Summary of Estimated Radiation Doses to the Maximum-Exposed Individual From Postulated Transportation Accidents During Decommissioning[a]

Accident Severity	Total Release (Ci/hr)[b]	First-Year Dose (rem)			Fifty-Year Committed Dose Equivalent (rem)		
		Total-Body	Bone	Lung	Total-Body	Bone	Lung
Minor	5×10^{-4}	2.5×10^{-4}	6.0×10^{-4}	8.0×10^{-4}	5.5×10^{-4}	4.8×10^{-3}	1.6×10^{-3}
Severe	2×10^{-2}	1.0×10^{-2}	2.4×10^{-2}	3.2×10^{-2}	2.2×10^{-2}	1.9×10^{-1}	6.4×10^{-2}

(a) Summarized from Table J.3-13 of Appendix J.
(b) Releases assumed to occur in a 1-hr period for comparison purposes.

14.3.3.3 Nonradiological Safety Impacts of Decommissioning Transportation Activities

A certain potential for accidental injury or death exists from transportation accidents during the decommissioning of the reference PWR. A summary of the casualties estimated to result during transportation activities for post-accident decommissioning is presented in Table 14.3-9. As shown in the table, about 2.2 injuries and 0.13 fatalities are calculated for transportation activities during DECON following a scenario 2 reactor accident. The corresponding casualty estimates for preparations for safe storage and for ENTOMB are about 6% and 45%, respectively, of the values estimated for DECON.

TABLE 14.3-9. Summary of Estimated Casualties from Transportation Accidents During Decommissioning[a]

Decommissioning Alternative	Number of Shipments	Injuries	Fatalities
DECON	1352	2.2	0.13
Preparations for Safe Storage	86	0.14	0.0084
ENTOMB	613	1.0	0.060

(a) Summarized from Table J.3-14 of Appendix J; estimates rounded to two significant figures, based on 3200-km round trip per shipment.

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CHAPTER 15

IMPACTS OF TEMPORARY INABILITY TO DISPOSE OF WASTES OFFSITE

A basic assumption of the analysis presented in the preceding chapters is that all radioactive waste materials from accident cleanup and from decommissioning that follows accident cleanup are disposed of offsite. If offsite waste disposal capability were to be temporarily unavailable during this period due to technical, regulatory, or political constraints, the selection of the preferred alternative for completing the decommissioning could be affected. The purpose of this chapter is to examine the impacts on accident cleanup and decommissioning of the reference PWR for three cases predicated on the temporary inability to dispose of radioactive materials offsite. The three cases are:

Case 1: offsite disposal for process solids and spent fuel available; low-level waste (LLW) stored onsite

Case 2: offsite disposal for LLW available; process solids and spent fuel stored onsite

Case 3: no offsite waste disposal available; process solids, spent fuel, and LLW all stored onsite.

The analysis of this chapter is limited to identifying some onsite disposal options and estimating cost and safety impacts for these options. This chapter does not examine in detail the technical or regulatory requirements for converting the facility to one in which waste storage can take place.

A basic assumption of this chapter is that if offsite disposal capacity is available (or unavailable) for a particular waste form, this condition exists for both accident cleanup and decommissioning wastes.

In this analysis, for wastes requiring onsite storage, a maximum safe storage period of 100 years is assumed. Use of a 100-year safe storage period is made to provide a basis for estimates of cost and safety impacts. Use of a

100-year safe storage period is not meant to imply that the radioactive wastes from accident cleanup and decommissioning would require onsite storage for this period, or that 100 years is a preferred time period. Impacts of storage times less than 100 years can be inferred from this analysis.

Process solids are the solid wastes that arise from the treatment of accident water and decontamination liquids during accident cleanup. These wastes include filter cartridge assemblies, ion exchange media (inorganic zeolites and organic resins), and evaporator bottoms. The amounts and properties of process solids generated during accident cleanup at the reference PWR are listed in Table E.4-2 of Appendix E. The base case assumption made in this study is that these solids may be temporarily stored onsite, but that they are transported to a federal repository for storage or disposal during or shortly after the completion of accident cleanup.

Spent fuel includes fuel assemblies stored in the fuel pool from prior refuelings during normal operation of the reactor and the undamaged and damaged fuel discharged during defueling following the accident. As described in Section E.4.1.3 of Appendix E, damaged fuel is assumed to be packaged in canisters prior to temporary storage in the spent fuel pool. The base case assumption made in this study is that undamaged fuel assemblies are transported to an independent spent fuel storage installation (ISFSI) and that damaged assemblies are shipped to a federal repository for examination and storage or disposal.

LLW includes dry radioactive wastes (activated structural components, contaminated equipment and material, and trash) generated from decontamination and defueling operations during accident cleanup and from post-cleanup decommissioning, and chemical decontamination solutions that are immobilized by incorporation in cement or in vinyl ester styrene. Waste quantities and packaging requirements for LLW from PWR accident cleanup are given in Table E.4-2 of Appendix E. Waste quantities and packaging requirements for LLW from post-accident decommissioning are given in Tables H.1-3 (DECON wastes), H.2-3 (preparations for safe storage wastes), and H.3-3 (ENTOMB wastes) of Appendix H. The base case assumption for this study is that LLW is transported to a shallow-land burial ground for disposal in accordance with applicable regulations.

15.1 DECOMMISSIONING ALTERNATIVES

Three alternative approaches to completing the decommissioning, following a reactor accident and post-accident cleanup, are analyzed in this study. These alternatives, discussed in detail in Section 4.1.3 of Chapter 4, are:

- DECON - The immediate removal from the facility of all material with residual radioactivity levels greater than those permitted for unrestricted use of the property. DECON meets the requirements for termination of the facility license and renders the facility and site available for unrestricted use within a finite time period.
- SAFSTOR - Activities designed to place (preparations for safe storage) and maintain (safe storage) a radioactive facility in such a condition that the risk to public safety is within acceptable bounds. At the conclusion of the safe storage period, the facility must be decontaminated to levels that permit its release for unrestricted use (deferred decontamination).
- ENTOMB - Cleanup and decontamination, to a lesser extent than for DECON, is coupled with the confinement of the remaining contaminated components in a strong and structurally long-lived material to assure retention until the radioactivity decays to levels that permit unrestricted release of the property.

The applicability of each of these alternatives for completing the decommissioning is affected by the temporary inability to dispose of accident cleanup and decommissioning wastes at offsite locations. In this chapter, analyses are performed only for those decommissioning alternatives that appear practical under each of the three onsite waste storage cases. Practical alternatives with onsite waste storage, summarized in Table 15.1-1, are discussed briefly in the following paragraphs.

TABLE 15.1-1. Practical Decommissioning Alternatives with Onsite Waste Storage

Decommissioning Alternative	Waste Storage Case: 1. LLW Stored Onsite	2. Process Solids and Spent Fuel Stored Onsite	3. Process Solids, Spent Fuel, and LLW All Stored Onsite
DECON	Partial DECON may be practical[(a)]	Partial DECON may be practical[(b)]	Partial DECON may be practical[(c)]
SAFSTOR	Practical	Practical	Practical
ENTOMB	Not practical	Not practical	Not practical

(a) Not analyzed in this chapter; see text.
(b) This involves the conversion of the spent fuel pool to a facility for temporary storage of the fuel and the decontamination of the containment building and of other onsite structures not needed for fuel storage to levels that permit unrestricted use.
(c) Not analyzed in this chapter. This would be a combination of case 1 and case 2.

DECON implies the prompt removal of all radioactive wastes from the site in order to allow unrestricted release of the property. Onsite storage of accident cleanup and decommissioning wastes would prevent release of the site until the wastes are subsequently shipped to an offsite disposal facility. Therefore, DECON appears to be generally inconsistent with onsite storage of radioactive wastes.

A form of partial DECON may be practical if only LLW must be stored onsite (case 1). In partial DECON, portions of the facility not required for waste storage would be decontaminated to levels that permit unrestricted use; but, unlike DECON as defined on the previous page, a nuclear license would be retained for the portion of the facility not decontaminated to unrestricted use levels. Wastes from accident cleanup and decommissioning activities would be packaged and stored inside one of the buildings that is not released for unrestricted use. This may be desirable if the availability of offsite LLW disposal is likely to be restored in a short time (i.e., less than about 5 years) because the waste could then be shipped offsite and the site released for unrestricted use in the shortest possible time. However, if early release of the site is not of paramount importance, SAFSTOR is the more logical choice because of reduced initial costs. Partial DECON with short-term onsite storage of LLW would result in very nearly the same costs and safety impacts as DECON with prompt offsite waste disposal and therefore this alternative is not analyzed further.

Another form of partial DECON that might be feasible involves the conversion of the spent fuel pool in the fuel building to an onsite facility for the temporary storage of spent fuel. In this case, the containment building and other onsite structures not needed for temporary fuel storage activities would be decontaminated to levels permitting unrestricted use. Decontamination and modification of facilities in the auxiliary and fuel buildings would be made to permit the continued onsite storage of spent fuel (both the fuel from defueling following the accident and any fuel remaining in the pool from previous refuelings). Appropriate security and surveillance measures would be established. The nuclear license would be amended in accordance with appropriate regulations to permit continued operation of the spent fuel pool as a fuel storage facility.

For the onsite storage of spent fuel, the spent fuel pool and its associated systems would require operation and maintenance throughout the entire storage period. Furthermore, the spent fuel pool is assumed to have adequate capacity to store all of the fuel (both the bare and canistered assemblies) that must remain onsite. This might require extensive reracking of the existing pool. Use of the pool exclusively for spent fuel storage would require that some other provision be made to house the filter/demineralizer system used during accident cleanup to process contaminated water. (In Section 10.4.1.1, the assumption is made that the demineralizer is installed in the spent fuel pool.) Alternatively, the filter/demineralizer system would require removal from the pool following completion of water processing operations and prior to the start of defueling operations. The base case analysis of this study assumes removal of the demineralizer during decommissioning operations. This is also discussed in Section 11.6 of Chapter 11.

SAFSTOR appears to be a practical decommissioning alternative for all three onsite waste storage cases. Initial decommissioning activities are minimized, resulting in relatively little waste being generated during preparations for safe storage. The radioactive contamination in the plant is reduced by radioactive decay during safe storage, reducing the amount of waste requiring eventual removal. (The ^{137}Cs radioactivity would decay by about a factor of 10 during the 100-year storage period postulated for this analysis.)

ENTOMB does not appear to be a practical alternative in any of the three onsite waste storage cases considered here. ENTOMB is not amenable to onsite storage of spent fuel at the reference PWR without significant changes in the preparations for entombment, changes that require considerable extra effort and expense to achieve. The presence of spent fuel onsite necessitates an onsite security force plus operations staff for the spent fuel pool, thus negating the principal reasons for selecting ENTOMB, that of minimizing costs during the storage period. In addition, the postulated entombment structure will not hold all of the LLW resulting from accident cleanup and decommissioning. Therefore, a commitment to ENTOMB in the event of unavailability of offsite LLW disposal appears unlikely because there can be no adequate incentive for entombing only a portion of the wastes to remain onsite.

To determine the effects of each of the three waste storage cases on accident cleanup and decommissioning of the reference PWR, the following analyses are performed:

1) The postulated treatment and storage conditions for the wastes to be retained onsite are outlined.

2) Major changes in accident cleanup activities and in decommissioning activities and requirements from the base case are identified and discussed, to provide a basis for quantification of the impacts.

3) Cost and safety impacts are estimated for each of the onsite storage cases and are compared with those of the base case.

The base case is that in which all accident cleanup and decommissioning wastes are shipped offsite. Activities, costs, and safety information for the base case are summarized in Chapters 10 through 14. Information in these chapters and in their associated appendices provides the bases for the cost and safety analyses of this chapter.

15.2 DECOMMISSIONING ANALYSIS

The analysis of the effects of the temporary unavailability of offsite waste disposal during accident cleanup and decommissioning of the reference PWR is presented in this section. SAFSTOR and partial DECON are the decommissioning alternatives considered. Cost and safety impacts of onsite waste storage are analyzed for accident cleanup and decommissioning following a scenario 2 accident. Wastes stored onsite are assumed to remain onsite for 100 years, and offsite disposal capacity is assumed to be established prior to the time of deferred decontamination. Impacts of storage for periods of less than 100 years can be inferred from the results of this analysis. The four onsite waste storage cases considered are:

Case 1: SAFSTOR with LLW stored onsite (offsite disposal for process solids and spent fuel available)

Case 2: SAFSTOR with process solids and spent fuel stored onsite (offsite LLW disposal available)

Case 3: SAFSTOR with all accident cleanup and decommissioning wastes stored onsite (no offsite waste disposal available)

Case 4: Partial DECON with spent fuel stored onsite (offsite disposal for process solids and LLW available).

Changes from the base case that result from each of the onsite waste storage cases are described in Section 15.2.1. The resulting cost impacts are described in Section 15.2.2. Safety impacts are described in Section 15.2.3.

15.2.1 Technical Requirements for Accident Cleanup and Decommissioning with Onsite Waste Storage

The potential need for onsite storage of accident cleanup and decommissioning wastes is anticipated to result in some changes in the requirements for carrying out accident cleanup and decommissioning activities at the reference PWR. Changes that would have significant effects on the costs and/or safety of post-accident cleanup and decommissioning are discussed, for each onsite waste storage case, in the following subsections.

15.2.1.1 Case 1: SAFSTOR with LLW Stored Onsite

Total treated volumes of LLW generated during accident cleanup in the reference PWR are estimated to be approximately 930 m^3 following the scenario 1 accident, 4000 m^3 following the scenario 2 accident, and 6600 m^3 following the scenario 3 accident (see Tables F.2-3, F.3-3, F.3-4, and F.3-5 of Appendix F). Treated volumes assume incineration of combustible waste with solidification of the resulting ash and compaction of compactible, noncombustible wastes. During preparations for safe storage at the reference PWR following the scenario 2 accident, approximately 480 m^3 of radioactive wastes are shipped offsite (see Section H.2.1.2 of Appendix H). The requirements for waste management during preparations for safe storage following accident cleanup are judged not to be significantly altered by changes in accident scenario (no more than $\pm$10% from scenario 2 to scenario 1 or scenario 3).

For the base case (offsite shipment of radioactive wastes during cleanup and decommissioning operations), a metal-sided building on a concrete foundation is postulated to be constructed during preparations for accident cleanup to serve as an interim storage facility for wastes awaiting shipment to a shallow-land burial ground (see Section E.2.1.4). For onsite retrievable storage of LLW, the capacity of this storage warehouse could be increased to accommodate the total volume of LLW from accident cleanup and decommissioning. Use of various locations in the reference PWR for onsite retrievable storage of radioactive wastes is also possible and is discussed in Section H.3.3 of Reference 1. Retrievable storage of LLW from both accident cleanup and preparations for safe storage could be implemented at selected locations in the containment building or in the auxiliary buildings as indicated in Reference 1. However, in this chapter, the former storage alternative (i.e., interim storage in a warehouse constructed onsite) is analyzed since it appears to represent the higher-cost alternative.

Changes in the requirements for accident cleanup and preparations for safe storage to accommodate onsite storage of LLW are anticipated to be minimal. The warehouse facility for temporary storage of some of the LLW from accident cleanup would be increased in size to accommodate all of the LLW from accident cleanup and preparations for safe storage. Manpower requirements for treating the waste and placing it in storage onsite are judged to be about the same as for packaging and loading the waste on trucks for shipment to an offsite disposal facility.

Changes in the requirements during the safe storage period would be the additional surveillance and maintenance required to ensure the integrity of the warehouse structure and the waste packages for the 100-year storage period. A detail ' analysis of the safety aspects of onsite storage of this waste is outside the scope of this study.

The only changes in the requirements during deferred decontamination following safe storage are that the packaged LLW would be removed and shipped offsite for disposal and the warehouse structure would be decommissioned. These activities are anticipated to have only a minor impact on the deferred decontamination of the facility

15.2.1.2 Case 2: SAFSTOR with Process Solids and Spent Fuel Stored Onsite

Total volumes of process solids (filter cartridges, zeolite and organic-resin ion exchange liners, and evaporator bottoms) accumulated during accident cleanup in the reference PWR are estimated to be approximately 50 m^3 following the scenario 1 accident, 105 m^3 following the scenario 2 accident, and 230 m^3 following the scenario 3 accident (see Tables F.2-3, F.3-3, F.3-4, and F.3-5 of Appendix F). A below-ground storage facility consisting of concrete cells with concrete cover blocks for temporary onsite storage of process solid wastes is postulated to be constructed during preparations for accident cleanup (see Section E.2.1.4 of Appendix E). Continued storage of these wastes for the reference 100-year storage period is assumed to be possible if provision is made for the removal of gases that could be generated as a result of radiolytic processes. Heat generation from the stored wastes is estimated not to be a problem. The liners having the largest heat generation rates (zeolite liners containing 50,000 Ci of ^{137}Cs) will produce heat at a rate of about 350 watts/liner. Most of the liners will generate heat at rates of 1 to 3 orders of magnitude below this value.

It is assumed for this analysis that ample space is available in the spent fuel pool for all of the spent fuel that must remain onsite. Therefore, no significant expenditures or efforts are assumed to be needed during preparations for accident cleanup to provide additional facilities for the spent fuel. The spent fuel storage pool and its associated systems and services are kept in operation throughout the safe storage period rather than being deactivated. This results in some changes in the activities that take place during preparations for safe storage since the fuel pool and related systems and services are not deactivated. Any maintenance required to ensure the continued operation of the fuel pool facilities during the safe storage period is performed during preparations for safe storage. A license amendment is required to permit retention of the spent fuel onsite during the safe storage period.

As explained in Section 10.4.1.1, a filter/demineralizer system for processing contaminated water is postulated to be installed in the spent fuel pool during preparations for accident cleanup. Use of the fuel pool exclusively for fuel storage would require that some other provision be made to accommodate the demineralizer. It is postulated that a steel building containing a small pool and equipment for handling the highly radioactive ion exchange liners is constructed adjacent to the auxiliary building during preparations for accident cleanup and that the demineralizer system is installed in this building. The cost of constructing and equipping this building is estimated to be about \$0.5 million. Construction is estimated to add about 6 months to the time of preparations for accident cleanup, resulting in an increase of about \$12 million in preparations-for-cleanup costs.

During continuing care of the reactor facility (the safe storage period), additional staffing and funding is required to operate and maintain the spent fuel storage pool and associated systems and services. A detailed discussion of the technology of onsite storage of spent fuel is beyond the scope of this study. However, as an example, provision may need to be made during this period for the overpacking of fuel assemblies that develop leaks due to corrosion. In addition, according to current regulations set forth in 10 CFR 73,[2] a security force consisting of a minimum of five armed personnel per shift must be maintained as long as the spent fuel remains onsite. Assuming a four-shift operation, this results in security manpower requirements of 20 man-years annually during the safe storage period, 4 man-years of which are supervisory personnel. Since retention of the fuel onsite following the termination of reactor operations is equivalent to conversion of the site to an ISFSI, maintenance and other factors related to spent fuel storage are not considered part of decommissioning. The costs of onsite storage of the fuel during the safe storage period are considered to be operational costs rather than decommissioning costs. An estimate of the potential magnitude of these costs is made in Section 15.2.2.2 for informational purposes only.

During the deferred decontamination of the reference PWR, the security personnel remain onsite until the spent fuel is shipped to an offsite repository. The spent fuel shipment is carried out in the same manner as described in Section F.1.8 of Appendix F. The spent fuel storage pool and associated systems and services are then decommissioned in the same manner as during DECON. Process solid wastes are shipped to an offsite repository and the shielded storage facility is decontaminated. The building constructed to house the demineralizer system is also decontaminated. The remainder of the deferred decontamination activities are the same as those described in Section G.3.5 of Appendix G. Costs for deferred decontamination are expected to be higher for this case than for the base case, with most of the cost differential related to the shipment of spent fuel.

As an alternative to wet storage of the spent fuel to be retained onsite, the fuel, after an appropriate cooling period, could be placed in dry storage casks situated onsite. This would provide additional storage capacity if the spent fuel pool is of inadequate size to store all the spent fuel that must remain onsite. Alternately, all of the fuel to remain onsite could be placed in such casks and the spent fuel pool and related systems then deactivated and decontaminated as for normal SAFSTOR. Spent fuel storage casks that could be used for this purpose are described in a forthcoming decommissioning study in this series.[a] Such casks are estimated to cost upwards of $0.5 million each, and to hold nine PWR fuel assemblies. The cost calculations performed in this study are for the wet storage of the spent fuel pool and do not include costs for dry cask storage of the fuel.

15.2.1.3 Case 3: SAFSTOR with All Wastes Stored Onsite

This case represents a combination of the two cases described in the previous subsections. The LLW from accident cleanup and post-cleanup decommissioning is placed in onsite storage. Process solid wastes remain in storage in the shielded storage facility instead of being shipped offsite. The spent fuel storage pool and associated systems and services remain in operation during the preparations for safe storage and subsequent safe storage period. Requirements for maintenance and for security during the safe storage period are significantly increased from the base case requirements. During deferred decontamination, the LLW, process solid wastes, and the spent fuel are shipped offsite. The spent fuel pool and its associated systems and the onsite storage facilities are decommissioned. Deferred decontamination then proceeds in the same manner as if the wastes (including the spent fuel) had been shipped offsite during accident cleanup or preparations for safe storage.

15.2.1.4 Case 4: Partial DECON with Spent Fuel Stored Onsite

For this case, the LLW and process solids are assumed to be shipped offsite. The containment building and all other onsite structures except the fuel and auxiliary buildings are decontaminated to levels permitting

(a) Technology, Safety and Costs of Decommissioning Reference Independent Spent Fuel Storage Installations (NUREG/CR-2210); to be published in FY-83.

unrestricted use. Portions of the auxiliary building that do not house equipment required for safe operation of the spent fuel pool are also decontaminated to unrestricted release levels. Appropriate controls are established for the movement of personnel and equipment from unrestricted to restricted use areas.

The spent fuel storage pool and its associated systems and services are maintained in operation during the fuel storage period. Any maintenance required to ensure the proper operation of these facilities is performed during the decommissioning of the other onsite structures. The nuclear license is amended in accordance with the provisions of applicable regulations to permit onsite retention of the fuel. The considerations relative to spent fuel storage requirements and costs discussed in Section 15.2.1.2 also apply to this case.

At the conclusion of the onsite fuel storage period, the fuel is shipped offsite and the spent fuel pool, fuel building, and portions of the auxiliary building that house systems and services required during onsite fuel storage are decontaminated to unrestricted use levels.

15.2.2 Cost Impacts of Onsite Waste Storage During Accident Cleanup and Decommissioning at the Reference PWR

Comparisons of the costs of accident cleanup and decommissioning at the reference PWR for each of the onsite waste storage options considered in this chapter with costs for the base case (no onsite waste storage) are presented in Tables 15.2-1 and 15.2-2. Cost comparisons for those cases that involve SAFSTOR (cases 1, 2, and 3) are presented in Table 15.2-1. Cost comparisons for case 4 that involves partial DECON are presented in Table 15.2-2. Costs are shown for accident cleanup and decommissioning following the scenario 2 accident. All costs are in early-1981 dollars and include a 25% contingency. Costs of accident cleanup for the base case are taken from Tables F.1-1, F.2-1, and F.3-1 of Appendix F. Costs of DECON and SAFSTOR for the base case are taken from Tables H.1-1 and H.2-1 of Appendix H.

It should be noted that the waste management costs shown in Tables 15.2-1 and 15.2-2 include cask rental and transportation charges for the shipment of

TABLE 15.2-1. Estimated Costs of Accident Cleanup and SAFSTOR at the Reference PWR as a Function of Onsite Waste Storage

	Estimated Costs (Millions of 1981 Dollars)[a]			
Cost Category	Base Case: No Onsite Waste Storage[b]	Case 1: LLW Stored Onsite	Case 2: Process Solids and Spent Fuel Stored Onsite	Case 3: All Wastes Stored Onsite
Accident Cleanup				
Staff Labor	101.481	101.481	107.577	107.577
Waste Management[c]	12.048	8.863	6.195	2.817
Other Costs	84.153	84.653	90.653	90.653
Totals, Accident Cleanup	197.7	195.0	204.4	201.0
Preparation for Safe Storage				
Staff Labor	8.623	8.623	8.623	8.623
Waste Management	1.066	0.198	1.066	0.198
Other Costs	6.985	6.985	6.889	6.889
Totals, Preparations for Safe Storage	16.7	15.8	16.6	15.7
Safe Storage				
Annual Continuing Care Costs	0.111	0.121	0.111	0.121
Totals, 100 years of Safe Storage	11.1	12.1	11.1	12.1
Annual Spent Fuel Storage Costs[d]			1.0	1.0
Deferred Decontamination				
Totals, Deferred Decontamination	44.7	49.0	50.8	55.0
TOTAL COSTS (100-year Storage)[e]	270.2	271.9	282.9	283.8

(a) Costs are for accident cleanup and SAFSTOR following a scenario 2 accident. All costs include a 25% contingency.
(b) Accident cleanup costs are from Tables F.1-1, F.2-1, and F.3-1 of Appendix F.
(c) No disposal charge for spent fuel included; see text.
(d) Costs of interim storage of spent fuel are considered operational rather than decommissioning costs and are shown for informational purposes only. Not included in total costs.
(e) Includes costs of offsite shipment of spent fuel from post-accident reactor defueling only.

TABLE 15.2-2. Estimated Costs of Accident Cleanup and DECON at the Reference PWR as a Function of Onsite Waste Storage

Cost Category	Estimated Costs (Millions of 1981 Dollars)(a): Base Case: No Onsite Waste Storage(b)	Case 4: Partial DECON with Onsite Storage of Spent Fuel
Accident Cleanup		
Staff Labor	101.481	107.577
Waste Management(c)	12.048	7.157
Other Costs	84.153	90.653
Totals, Accident Cleanup	197.7	205.4
DECON(d)		
Staff Labor	33.533	30.995
Waste Management	20.281	20.181
Other Costs	14.023	13.823
Totals, DECON	67.8	65.0
Annual Spent Fuel Storage Costs(e)		1.0
Deferred Decontamination		
Totals, Deferred Decontamination	———	15.0
TOTAL COSTS (100-year Storage)(f)	265.5	285.4

(a) Costs are for accident cleanup and DECON following a scenario 2 accident. All costs include a 25% contingency.
(b) Accident cleanup costs are from Tables F.1-1, F.2-1, and F.3-1 of Appendix F. DECON costs are from Table H.1-1 of Appendix H.
(c) No disposal charge for spent fuel included; see text.
(d) For case 3, costs are for partial DECON as described in Section 15.2.1.
(e) Costs of interim storage of spent fuel are considered operational rather than decommissioning costs and are shown for informational purposes only. Not included in total costs.
(f) Includes costs of offsite shipment of spent fuel from post-accident reactor defueling only.

the spent fuel to an offsite repository, but do not include a disposal charge for the fuel. This is because the ultimate disposition of spent fuel is as yet undefined and disposal costs are speculative. Since the cost of disposal of the fuel (in 1981 dollars) is the same for each case considered, omission of this cost from the tables does not affect comparisons between cases.

Details of cost estimates for the three onsite waste storage cases are given in the following subsections.

15.2.2.1 Case 1: SAFSTOR with LLW Stored Onsite

For accident cleanup and preparations for safe storage with LLW stored onsite (case 1), no significant changes in the costs of staff labor are anticipated, because offsite shipment of the LLW is replaced by placement of the waste in onsite storage locations. Waste management costs are reduced by the costs of transportation and burial of the LLW. Costs in the "other cost" category are increased by the amount required for construction of the warehouse facility for onsite waste storage. The total costs of accident cleanup and preparations for safe storage are estimated to be about \$195 million, or about \$3 million less than the corresponding costs with offsite shipment of the LLW.

Annual costs during the safe storage period are estimated to be increased by about \$10,000 to provide for maintenance of the warehouse facility used for storage of the LLW.

During deferred decontamination for case 1, additional decommissioning staff are needed to ship the LLW offsite and to decommission the storage facility. The cost of this additional staff is estimated as shown in Table 15.2-3. It is assumed that the additional crews are needed for 6 months (two shifts per day, five days per week) to ship the waste and decommission the warehouse. The total additional staff labor cost is approximately \$250,000 in 1981 dollars, including the 25% contingency. There are also additional waste management costs associated with the transportation and disposal of the LLW. Additional costs during deferred decontamination other than for staff labor and waste management are judged to be negligible. Thus, the total additional cost during deferred decontamination of the reference PWR with

TABLE 15.2-3. Additional Staff Labor Requirements and Costs During Case 1 Deferred Decontamination at the Reference PWR

Labor Category	Number per Shift	Labor Requirements(a) (man-years)	Unit Cost/ Man-Year(b) ($ thousands)	Total Cost ($ thousands)
Crew Leader	1	1.0	44.4	44.4
Utility Operator	2	2.0	32.1	64.2
Laborer	2	2.0	30.9	61.8
Health Physics Technician	1	1.0	30.0	30.0
Subtotal		6.0		200.4
25% Contingency				50.1
Total				250

(a) Based on two shifts/day, 5 days/week for 6 months.
(b) From Table I.1-1 of Appendix I.

onsite storage of the LLW is estimated to be about $4.3 million, bringing the total cost for the deferred decontamination to about $49.0 million in 1981 dollars.

The total estimated cost (1981 dollars) for accident cleanup and SAFSTOR at the reference PWR following a scenario 2 accident is about $272 million with onsite LLW storage, as compared to a total of about $270 million with no onsite storage. Thus, there is no major difference between the total costs for the two cases. However, it should be noted that the costs of accident cleanup and preparations for safe storage are reduced by about $3 million while the costs of deferred decontamination are increased by slightly more than $4 million. Thus, financing considerations for decommissioning are somewhat affected.

15.2.2.2 Case 2: SAFSTOR with Process Solids and Spent Fuel Stored Onsite

As stated previously in Section 15.2.1.2, it is assumed for this analysis that continuing storage of process solid wastes in the shielded storage facility described in Section E.2.1.4 of Appendix E is possible for the entire

100-year storage period. The spent fuel pool and its associated systems and services are kept in operation throughout the safe storage period to provide for onsite storage of the spent reactor fuel. During preparations for safe storage, decommissioning activities in the fuel and auxiliary buildings are changed because the fuel pool and related systems and services are not deactivated. Activities in these buildings during this period are directed toward conversion of the facility to its next use.

Use of the fuel pool exclusively for fuel storage requires the construction of a special building to house the demineralizer system used to process contaminated water during accident cleanup operations. Construction of this facility takes place during preparations for accident cleanup. The cost of this building is estimated to be about $0.5 million and its construction is estimated to add about 6 months to the time requirement for preparations for accident cleanup. As shown in Tables F.1-1 and F.1-2 of Appendix F, the annual cost of utility staff labor during preparations for accident cleanup following a scenario 2 accident is about $12.2 million, including a 25% contingency. The annual cost of energy is about $6.0 million. Total annual costs of specialty contractors are about $5.1 million, and annual costs of nuclear insurance and license fees are about $1.9 million, including a 25% contingency. Thus, a 6-month increase in the time for preparations for accident cleanup is estimated to result in a $6.1 million increase in staff labor costs and a $6.5 million increase in other costs for preparations for accident cleanup.

Although the spent fuel pool and related systems and services are not deactivated during preparations for safe storage, these activities are assumed to be replaced by other activities needed to prepare for the long-term onsite storage of spent fuel. Therefore, these changes in fuel building activities are judged not to have significant effects on manpower requirements and costs during preparations for safe storage.

Waste management costs during case 2 accident cleanup are reduced by the cask rental and transport costs for spent fuel. There is no significant change in waste management costs for preparations for safe storage from those shown in Table H.2-3 of Appendix H.

For preparations for safe storage without onsite waste or spent fuel storage, it was assumed that a specialty contractor was hired for spent fuel pool decontamination and covering at a cost of $66,000 in 1978 dollars, without contingency (see Section 10.2.2 of Reference 1). This contractor would not be required for case 2 preparations for safe storage. Applying a cost updating factor of 1.2 and adding the 25% contingency, this would result in a savings in other costs during case 2 preparations for safe storage of about $96,000.

The total costs of case 2 accident cleanup and preparations for safe storage are estimated to be about $221 million, or about $7 million more than the corresponding costs with offsite shipment of the LLW.

Annual continuing care costs during case 2 safe storage are estimated to be approximately the same as they are for the base case. These costs do not include the costs of onsite storage of spent fuel, which are assumed to be operational costs rather than decommissioning costs. Routine surveillance and maintenance operations related to spent fuel storage are estimated to cost about $1 million annually. If there were to be problems, such as badly leaking fuel elements or structural damage to the pool liner, these costs could increase significantly. The rationale for these estimated spent fuel storage costs is given in the following paragraphs. The costs of onsite storage of spent fuel are shown as a line item in Table 15.2-1, but are not included in the total case 2 costs.

Current regulations regarding the security force that must be maintained during onsite spent fuel storage are described in Reference 2. The total number of guards and armed, trained personnel that must be available onsite at all times may not be reduced to less than five (10 CRR 73.55(h)(2)). Of this number at least one full-time member of the security force with supervisory authority must be onsite at all times. Thus, during onsite spent fuel storage, assuming a four-shift operation, the security personnel required total 4 man-years/year for security supervisors and an additional 16 man-years/year for security patrolmen. Based on staff labor costs presented in Table I.1-1 of Appendix I, the total additional cost for onsite security during continuing care due to the onsite storage of spent fuel is $720,000 per year in 1981 dollars, including a 25% contingency.

To ensure the continued operational capability of the spent fuel storage system, it is judged that utility operators and maintenance personnel will be required on a full-time, single-shift basis. In addition, allowance must be made for the costs of energy, equipment and suppliers, insurance, and license fees. It is judged that an annual allowance of about $250,000 is sufficient to cover these additional expenses of onsite spent fuel storage. Based on the above considerations, the total cost of routine security and maintenance operations for onsite spent fuel storage is estimated to be about $1 million annually.

For case 2 deferred decontamination of the reference PWR, the costs are increased from those incurred during deferred decontamination without onsite waste storage by: 1) the shipment and disposal costs for removing the process solid wastes and the spent fuel, and 2) the additional costs associated with decommissioning the spent fuel pool and the facility for shielded storage of the process solid wastes. The cost of shipment and disposal of the process solid wastes is $971,000 in 1981 dollars, including a 25% contingency, based on information in Tables F.2-3 and F.3-4 of Appendix F. The cost of shipment of the fuel and fuel core debris removed from the reactor following the accident is $4,891,000. As stated previously, the cost of disposal of this fuel is omitted from this analysis. Costs are estimated only for the removal and offsite shipment of the fuel removed from the reactor following the accident, because removal and shipment of fuel discharged to the spent fuel pool prior to the accident is more properly charged to reactor operations. The added cost of decommissioning the spent fuel pool is estimated to be about $200,000.

The added costs of waste shipment and disposal and of spent fuel pool decontamination during case 2 deferred decontamination increase the total costs of deferred decontamination by almost 15%, to approximately $50.8 million. The total cost of accident cleanup and all phases of SAFSTOR at the reference PWR, with onsite storage of process solids and spent fuel, is estimated to be about $283 million, or 5% greater than the corresponding costs with no onsite waste storage. The increased cost is primarily due to increased expenditures during preparations for accident cleanup to provide a separate facility for the filter/demineralizer system.

15.2.2.3 Case 3: SAFSTOR with All Wastes Stored Onsite

For accident cleanup and SAFSTOR at the reference PWR with all of the radioactive wastes stored onsite, the net cost impacts, as compared to SAFSTOR with no onsite storage, are anticipated to be the sum of the cost impacts for case 1 (LLW stored onsite) and case 2 (process solids and spent fuel stored onsite), as derived in the previous two subsections.

During case 3 accident cleanup, staff labor costs are estimated to be about $107.6 million. Waste management costs are estimated to be reduced to about $2.8 million, the cost of disposable containers for the radioactive wastes and canistered fuel stored onsite. Other costs are estimated to total about $90.6 million. The total cost for accident cleanup with all of the wastes stored onsite is estimated to be about $201 million, or 2% more than the cost with no onsite waste storage.

During case 3 preparations for safe storage, staff labor costs are estimated to be about $8.6 million. Waste management costs are estimated to be reduced to about $198,000, the cost of containers for the LLW generated during preparations for safe storage and stored onsite. Other costs are estimated to total almost $6.9 million, the same as for case 2 preparations for safe storage. The total cost of preparations for safe storage with all wastes stored onsite is estimated to be about $15.7 million, about 95% of the cost with no onsite waste storage.

Annual continuing care costs during case 3 are estimated to be the same as those during case 1, or $0.12 million.

The total cost of deferred decontamination if all of the radioactive wastes are stored onsite during the safe storage period is estimated to be about $55.0 million, about 23% higher than for deferred decontamination with no onsite waste storage. The additional cost is due primarily to the costs of shipment and disposal of these radioactive wastes and the spent fuel.

The total cost for accident cleanup and all phases of SAFSTOR at the reference PWR with onsite storage of the radioactive wastes and spent fuel is estimated to be about $284 million in 1981 dollars. This is about 5% greater than the cost of accident cleanup and SAFSTOR with no onsite waste storage. The increased cost is due primarily to increased expenditures during preparations for accident cleanup to provide a separate facility for the filter/demineralizer system rather than installing it in the spent fuel pool. Accident cleanup, preparations for safe storage, continuing care, and deferred decontamination account for about 71%, 6%, 4%, and 19%, respectively, of the total estimated cost of case 3 accident cleanup and SAFSTOR at the reference PWR.

15.2.2.4 Case 4: Partial DECON with Spent Fuel Stored Onsite

As stated previously, it is assumed for this case that the containment building and other onsite structures except the fuel and auxiliary buildings are decontaminated to levels that permit unrestricted use. The spent fuel pool in the fuel building is converted to a facility for onsite storage of spent fuel. Portions of the auxiliary building that house equipment needed for operation of the fuel pool are also retained in operational status until the end of the fuel storage period. At the conclusion of the fuel storage period, the fuel and auxiliary buildings are decontaminated to unrestricted use levels.

As for case 2, use of the fuel pool exclusively for fuel storage requires the construction of a separate facility to house the filter/demineralizer system used to process contaminated water from accident cleanup operations.

Accident cleanup costs for case 4 are the same as they are for case 2, except that the waste management costs are greater because only the spent fuel is stored onsite (i.e., for case 4, the waste management costs include shipment and disposal costs for both the LLW and the process solid wastes).

DECON costs for case 4 are reduced by the amount of the expenditures for decontamination of the fuel and auxiliary buildings. This estimated cost reduction is small because most of the costs of DECON are associated with the decontamination of the containment building or are site support costs. As discussed in Section 12.2.3 of Chapter 12, decontamination of the fuel and auxiliary buildings for the base case is assumed to take place concurrently with decontamination of the containment building as staff are available and as the systems in these buildings complete their required service functions. Omission of fuel and auxiliary building decontamination would not significantly reduce the overall level of effort or the total time requirement for DECON of the reference PWR.

There are no continuing care costs for this case. The costs during the period of onsite spent fuel storage are considered to be operational costs rather than decommissioning costs and are shown in Table 15.2-2 for informational purposes only.

When the fuel is ultimately removed from the pool, decontamination of the fuel and auxiliary buildings is estimated to cost about $15 million in 1981 dollars. This cost is about equally distributed between labor, waste management, and other costs, and includes the cost of shipment of the spent fuel.

The total cost of accident cleanup and partial DECON with onsite storage of spent fuel followed by deferred decontamination of the fuel and auxiliary buildings is estimated to be about $285 million, or about 8% greater than the base case cost estimate for accident cleanup and DECON of the reference PWR following a scenario 2 accident. The additional cost is due primarily to the cost of construction of a separate facility to house the filter/demineralizer system and the cost of deferred decontamination of the fuel and auxiliary buildings.

15.2.3 Safety Impacts of Onsite Waste Storage During Accident Cleanup and Decommissioning at the Reference PWR

The safety impacts considered in the analyses of accident cleanup and decommissioning presented in Chapter 14 and Appendix J of this report include occupational and public safety impacts from both onsite (accident cleanup and decommissioning) and offsite (radioactive waste and spent fuel transportation activities. Because impacts to the public from onsite activities are estimated to be extremely small, and because these impacts are not considered to be significantly influenced by the possibility of onsite waste storage, public safety impacts from onsite activities are not considered further in this analysis. Furthermore, nonradiological safety impacts from accident cleanup and decommissioning activities are also not considered. Therefore, the safety impacts considered are as follows:

- occupational radiation doses to workers performing onsite accident cleanup and decommissioning activities
- occupational radiation doses to transportation workers during the offsite shipment of radioactive wastes and spent fuel
- radiation doses to members of the public resulting from the offsite shipment of radioactive wastes and spent fuel.

Comparisons of estimated safety impacts for each of the onsite waste alternatives considered in this chapter with safety impacts for the base case are shown in Table 15.2-4 for those cases that involve SAFSTOR (cases 1, 2, and 3), and in Table 15.2-5 for case 4 that involves partial DECON. As with the cost comparisons of Section 15.2.2, all of the safety comparisons are for accident cleanup and decommissioning of the reference PWR following the scenario 2 accident.

15.2.3.1 Safety Impacts of Onsite Waste Storage for Cases Involving Accident Cleanup and SAFSTOR

As shown in Table 15.2-4 the external occupational radiation dose to onsite decommissioning workers during accident cleanup and SAFSTOR at the reference PWR with no onsite waste storage is estimated to total about 5600 man-rem. This includes about 4620 man-rem during accident cleanup, about 430 man-rem during preparations for safe storage, about 230 man-rem during 100 years of continuing care, and approximately 300 man-rem during deferred decontamination, as shown in Tables J.2-11 and J.3-10 of Appendix J. Onsite

TABLE 15.2-4. Estimated Safety Impacts of Accident Cleanup and SAFSTOR at the Reference PWR as a Function of Onsite Waste Storage

Group/Activity	Estimated Radiation Dose (man-rem)(a)			
	No Onsite Waste Storage(b)	Case 1: LLW Stored Onsite	Case 2: Process Solids & Spent Fuel Stored Onsite	Case 3: All Wastes Stored Onsite
Decommissioning Workers/Onsite Decommissioning Activities				
Accident Cleanup	4620	4620	4620	4620
SAFSTOR	960	960	890	890
Totals, Decommissioning Workers	5580	5580	5510	5510
Transportation Workers/Offsite Waste & Spent Fuel Shipments				
Accident Cleanup	31	12	19	0
SAFSTOR	33	23	36	26
Totals, Transportation Workers	64	35	55	26
Public/Offsite Waste and Spent Fuel Shipments				
Accident Cleanup	3.2	1.4	1.8	0
SAFSTOR	3.1	2.2	3.5	2.6
Totals, Public	6.3	3.6	5.3	2.6

(a) Doses are for accident cleanup and SAFSTOR following a scenario 2 accident; rounded to two significant figures.
(b) Accident cleanup doses are based on information in Sections 14.2.2 and 14.2.3. SAFSTOR doses are based on information in Sections 14.3.2 and 14.3.3. A safe storage period of 100 years is assumed.

storage of the LLW generated during accident cleanup and SAFSTOR (case 1) is anticipated to have only very minor effects on these activities; thus, the occupational radiation doses to workers during case 1 accident cleanup and SAFSTOR are assumed to be about the same as for these activities with no onsite waste storage.

No change in the occupational dose to accident cleanup workers is anticipated as a result of the onsite storage of process solids and spent fuel (both cases 2 and 3). During preparations for safe storage with onsite

TABLE 15.2-5. Estimated Safety Impacts of Accident Cleanup and Partial DECON at the Reference PWR as a Function of Onsite Waste Storage

Group/Activity	Estimated Radiation Dose (man-rem)[a]: Base Case: No Onsite Waste Storage[b]	Estimated Radiation Dose (man-rem)[a]: Case 4: Partial DECON with Onsite Storage of Spent Fuel
Decommissioning Workers/Onsite Decommissioning Activities		
Accident Cleanup	4620	4620
DECON	3060	2950
Totals, Decommissioning Workers	7680	7570
Transportation Workers/Offsite Waste and Spent Fuel Shipments		
Accident Cleanup	31	27
DECON	200	200
Totals, Transportation Workers	230	230
Public/Offsite Waste and Spent Fuel Shipments		
Accident Cleanup	3.2	2.5
DECON	19	19
Totals, Public	22	21

(a) Doses are for accident cleanup and SAFSTOR following a scenario 2 accident; rounded to two significant figures.
(b) Accident cleanup doses are based on information in Sections 14.2.2 and 14.2.3. DECON doses are based on information in Sections 14.3.2 and 14.3.3.

storage of process solids and spent fuel, the external occupational doses to decommissioning workers are reduced by about 120 man-rem, the estimated dose for offsite shipment of the process solids and spent fuel (see Table G.3-1 of Appendix G). Thus, the total dose for preparations for safe storage is about 310 man-rem. The occupational dose during continuing care is estimated to be the same for cases 2 and 3 as it is for the base case. The dose associated with offsite shipment of the process solids and spent fuel during deferred decontamination is estimated to be about a factor of 2 lower than the estimated dose for offsite shipment during prepartions for safe storage. Thus, the total accumulated occupational dose during deferred decontamination is increased by only 60 man-rem compared to the base case, resulting in a dose of about 360 man-rem. The estimated total external occupational dose to decommissioning workers during all phases of SAFSTOR at the reference PWR with onsite storage of process solids and spent fuel (cases 2 and 3) is therefore slightly reduced from its estimated value with no onsite waste storage (the base case).

Doses to transport workers during accident cleanup are estimated by adjusting the dose information presented in Tables J.2-14 and J.2-15 of Appendix J to account for the wastes shipped offsite during accident cleanup for each onsite waste storage case. For case 1, the onsite storage of LLW reduces the dose to 12 man-rem, the estimated dose from shipment of the process solids and spent fuel. For case 2, the dose is 19 man-rem, the dose from shipment of the LLW. For case 3, there is no offsite shipment of waste during accident cleanup.

Estimated radiation doses from shipment of radioactive materials during SAFSTOR of the reference PWR are shown in Table J.3-12 of Appendix J for the base case of no onsite waste storage. For case 1 SAFSTOR, the LLW from accident cleanup and preparations for safe storage is held onsite for the 100-year safe storage period and is then shipped with the LLW from deferred decontamination. This results in a total transportation worker dose of 23 man-rem, assuming that the 100-year storage period results in a factor of 10 reduction in the dose from the LLW generated during accident cleanup and preparations for safe storage. (This dose reduction factor is based on ^{137}Cs being the dominant radioisotope.) For case 2 SAFSTOR, the total dose to transportation workers includes the doses from shipment of the LLW generated during preparations for safe storage and during deferred decontamination (the same as for the base case) plus the dose from shipment of the process solid wastes and spent fuel stored onsite for the 100-year safe storage period. The dose from shipment of the process solid wastes is assumed to be reduced by a factor of 10 as a result of 100 years of storage. Because of the presence of some long-lived radionuclides, the radioactivity in the spent fuel does not decay at the same rate as that in the waste materials. It is assumed that the transportation worker dose that results from spent fuel shipment during deferred decontamination is about half of that from spent fuel shipment prior to the 100-year storage period. For case 2 SAFSTOR, the total dose to transportation workers is estimated to be about 36 man-rem. For case 3 SAFSTOR, the total dose to transportation workers includes the dose from the shipment of wastes generated during deferred decontamination plus that from shipment of the LLW, process solid wastes, and spent fuel stored onsite during the 100-year storage period. The total dose to transportation workers for case 3 SAFSTOR is estimated to be about 26 man-rem.

The doses to the public from transportation activities during accident cleanup and SAFSTOR at the reference PWR are calculated in the same manner as the transportation worker doses presented above, based on the public dose information presented in Tables J.2-14, J.2-15, and J.3-12 of Appendix J. Doses to the public from transportation activities during accident cleanup and SAFSTOR following the scenario 2 accident with no onsite waste storage total about 6.3 man-rem. Estimated corresponding doses during case 1, case 2, and case 3 accident cleanup and SAFSTOR are about 3.6 man-rem, 5.3 man-rem, and 2.6 man-rem, respectively, and are about 57%, 84%, and 41% of those for the base case.

15.2.3.2 Safety Impacts of Case 4: Partial DECON with Onsite Storage of Spent Fuel

As shown in Table 15.2-5, the external occupational radiation dose to onsite decommissioning workers during accident cleanup and DECON at the reference PWR with no onsite waste storage is estimated to total almost 7700 man-rem. This includes about 4620 man-rem during accident cleanup and about 3060 man-rem during DECON. No change in occupational radiation dose to accident cleanup workers is anticipated as a result of the onsite storage of spent fuel. For the partial DECON case in which decontamination of the auxiliary and fuel buildings is deferred until after the period of spent fuel storage, the dose to DECON workers is reduced by about 120 man-rem during initial decontamination activities, but a dose of about 10 man-rem is incurred during deferred decontamination. (The dose during deferred decontamination is estimated assuming a storage period of 100 years and a factor of 10 reduction in contamination levels.)

Doses to transport workers during case 4 accident cleanup are estimated by subtracting the dose from spent fuel shipment, shown in Table J.2-15 of Appendix J, from the total dose to transport workers for the no onsite waste storage case. Because the doses from spent fuel shipment and from shipment of radioactive wastes from decontamination of the fuel and auxiliary buildings are small, the dose to transportation workers during DECON is estimated to be approximately the same (to two significant figures) for both the base case and the partial DECON case.

Doses to the public from transportation activities during case 4 accident cleanup are estimated by subtracting the dose from spent fuel shipment, shown in Table J.2-15, from the total dose to the public for the no onsite waste storage case. As indicated above for transportation workers, the dose to the public during DECON activities is estimated to be approximately the same for both the base case and case 4.

15.3 CONCLUSIONS

Of the three alternative approaches to decommissioning following accident cleanup (i.e., DECON, SAFSTOR, or ENTOMB), both SAFSTOR and partial DECON appear to be practical at the reference PWR if temporary onsite storage of accident cleanup and decommissioning wastes or spent fuel were to be necessary. DECON of the total facility and ENTOMB both have characteristics that appear to make them generally unsuitable for post-accident decommissioning with onsite waste storage.

The estimated cost impacts of temporary onsite storage of accident cleanup and decommissioning wastes (including spent fuel) at the reference PWR following a scenario 2 accident are summarized in Tables 15.2-1 and 15.2-2. Onsite storage of LLW has virtually no effect on the total costs (in constant dollars) of accident cleanup and decommissioning. However, some costs that would normally occur prior to the safe storage period are delayed until deferred decontamination. Onsite storage of process solid wastes and of spent fuel increases the total cost of accident cleanup and decommissioning by a few percent. The increased cost is due primarily to increased expenditures during preparations for accident cleanup if a separate facility must be provided for the demineralizer system rather than installing it in the spent fuel pool and, for partial DECON, to deferred decontamination of the auxiliary and fuel buildings. In the analyses of this chapter, the costs directly associated with interim storage of spent fuel (i.e., the costs of operating and maintaining the fuel pool and of the operating staff and security force during the safe storage period) are assumed to be operational costs rather than decommissioning costs.

The estimated radiological safety impacts of onsite waste storage during accident cleanup and decommissioning at the reference PWR are summarized in Tables 15.2-4 and 15.2-5. As shown in the tables, occupational radiation doses are estimated to be essentially unaffected by onsite waste storage. Radiation doses to transport workers and the public from offsite waste shipments are estimated to be reduced by about a factor of 2 for cases involving SAFSTOR with onsite waste storage but to be essentially unaffected for the case of partial DECON with onsite storage of spent fuel. It should be noted that the estimated safety impacts to the public of accident cleanup and decommissioning are small, with or without onsite storage of decommissioning wastes.

The conclusions reported here regarding the cost and safety impacts of onsite waste storage are based on accident cleanup and decommissioning requirements following a scenario 2 accident. Similar conclusions are anticipated for onsite storage of accident cleanup and decommissioning wastes following a scenario 1 or scenario 3 accident.

REFERENCES

1. R. I. Smith, G. J. Konzek, and W. E. Kennedy, Jr., Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station, NUREG/CR-0130, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, June 1978.

2. U.S. Code of Federal Regulations, Title 10, Part 73, "Physical Protection of Plants and Materials," (10 CFR 73), U.S. Government Printing Office, Washington, D.C., December 1980.

CHAPTER 16

POST-ACCIDENT CLEANUP AND DECOMMISSIONING AT A REFERENCE BWR

The technical requirements, estimated costs, and safety impacts of post-accident decommissioning, including accident cleanup, of a large boiling water reactor power station (BWR) are summarized in this chapter. The technical approach used to perform the BWR accident cleanup and decommissioning analysis is described in Section 16.1. A brief description of the reference BWR is given in Section 16.2. Accident scenarios that provide the basis for the conceptual evaluation of accident cleanup and decommissioning are described in Section 16.3. Activities, manpower requirements, and costs of accident cleanup are summarized in Sections 16.4 and 16.5. Activities, manpower requirements, and costs of decommissioning following accident cleanup are summarized in Sections 16.6 and 16.7. Public and occupational safety impacts of post-accident cleanup and decommissioning are summarized in Section 16.8. Details of the analysis of post-accident cleanup and decommissioning of the reference BWR are given in Appendix K of Volume 2.

The BWR analysis is not performed to the same level of detail as the PWR analysis reported in Chapters 10 through 14. Where the results of BWR accident cleanup are judged not to be significantly different from those of PWR accident cleanup, the PWR results are used. Where the results of BWR post-accident decommissioning are judged not to be significantly different from the results of normal-shutdown decommissioning of the reference BWR reported in Reference 1, the normal-shutdown results are used. Significant differences between BWR and PWR accident cleanup requirements and between decommissioning requirements for the post-accident and the normal-shutdown cases are identified and discussed. It is believed that this approach has resulted in reasonable estimates of time, manpower requirements, costs, and safety impacts of BWR accident cleanup and decommissioning while reducing the analysis effort required to obtain the results. To trace the logic and justify the assumptions used in making the BWR analysis, the reader must refer to the appropriate sections in Chapter 8 and in Chapters 10 through 14 where the bases and assumptions for the PWR analysis are discussed.

16.1 TECHNICAL APPROACH

The overall approach used to perform this analysis of BWR post-accident cleanup and decommissioning is described in this section.

The reference BWR for this analysis is WNP-2 being built near Richland, Washington, by the Washington Public Power Supply System. The choice of WNP-2 as the reference reactor is made to facilitate comparisons between the requirements and costs of post-accident decommissioning and of normal-shutdown decommissioning and to provide consistency with the previously published report[1] of BWR decommissioning following normal shutdown. The use of WNP-2 as the reference BWR for this study should not be construed as implying anything about the reliability and/or safety of this reactor relative to other BWRs in operation or under construction. Its use is only to facilitate comparisons with the earlier, nonaccident, decommissioning study.

The three accident scenarios that provide a basis for the BWR post-accident analysis are basically similar to those for the PWR, discussed in Chapter 8. Because of differences in the BWR and PWR containment configurations, the accident consequences (i.e., the quantities and distributions of radioactive surface contamination and/or contaminated water present in the plant following an accident) are different for the two reactors. For the BWR, the radioactive surface contamination and the contaminated water resulting from an accident are assumed to be largely confined to the dry well and supression pool located inside the primary containment vessel.

The study approach used to evaluate BWR post-accident cleanup and decommissioning is the same as that described in Chapter 4 for PWR post-accident cleanup and decommissioning. Accident cleanup activities at the reference BWR precede the actual decommissioning of the plant and are assumed to be essentially independent of the alternative chosen for completing the decommissioning. The alternatives for decommissioning are DECON (immediate decontamination to unrestricted release), SAFSTOR (safe storage with deferred decontamination to unrestricted release), and ENTOMB (entombment of radioactive materials with decay to unrestricted release). The key study bases listed in Section 4.2 of Chapter 4 are assumed to apply to both the PWR and BWR analyses. Many of the assumptions made for the PWR are considered to be applicable to the BWR. Differences in assumptions are noted, where necessary.

Wherever possible, the BWR analysis described in this chapter makes use of the PWR accident cleanup and decommissioning analysis described in Chapters 10 through 14 and the BWR normal-shutdown decommissioning analysis of Reference 1. The goals of accident cleanup, described in Section 10.1, and the tasks, described in Section 10.4, that must be completed to achieve these goals are assumed to be the same for both PWR and BWR accident cleanup. The procedures and techniques used for accident cleanup and decommissioning are assumed to be generally similar for both the PWR and the BWR. These procedures and techniques are described in detail in Appendices D, E, and G. Differences in time schedules, manpower requirements, costs, and occupational doses resulting from the application of similar techniques and procedures are the result of differences in plant layout and in physical and radiological conditions inside the plant at the time accident cleanup and decommissioning activities take place.

16.2 CHARACTERISTICS OF THE REFERENCE BWR POWER STATION

The reference BWR is WPPSS Nuclear Project No. 2 (WNP-2) being built near Richland, Washington, by the Washington Public Power Supply System (WPPSS). It is a 3320-MWt (1155-MWe) boiling water reactor of the BWR/5 class and the Mark-II containment design. The choice of WNP-2 as the reference BWR is made to facilitate comparisons between the requirements of post-accident cleanup and decommissioning given in this study and the requirements of normal-shutdown decommissioning given in an earlier study of BWR decommissioning,[1] and is not intended to imply anything about the reliability and/or safety of this reactor relative to other BWRs in operation or under construction.

The components and structures of primary interest for post-accident cleanup and decommissioning are the reactor vessel, the primary containment, and the reactor building. These components and structures are described briefly in the following subsection. Additional details of the BWR nuclear power generation system and of plant structures are given in Appendix K of Volume 2. More complete descriptions of the reference BWR are given in Reference 1 and in the WPPSS Nuclear Project No. 2 Final Safety Analysis Report.[2]

16.2.1 Reactor Vessel, Primary Containment, and Reactor Building

The BWR nuclear power generation system is shown schematically in Figure 16.2-1. The pressure in a typical BWR reactor pressure vessel is maintained at about 6.9×10^3 kPa. At this pressure, water boils to form steam at about 285°C. The steam, generated by direct boiling in the nuclear core, passes through steam separators and dryers in the top of the reactor pressure vessel to remove trace amounts of entrained water and then goes directly to the turbine generator. Exhaust steam from the turbine is condensed, routed through a cleanup process to remove trace radioactive contaminants resulting

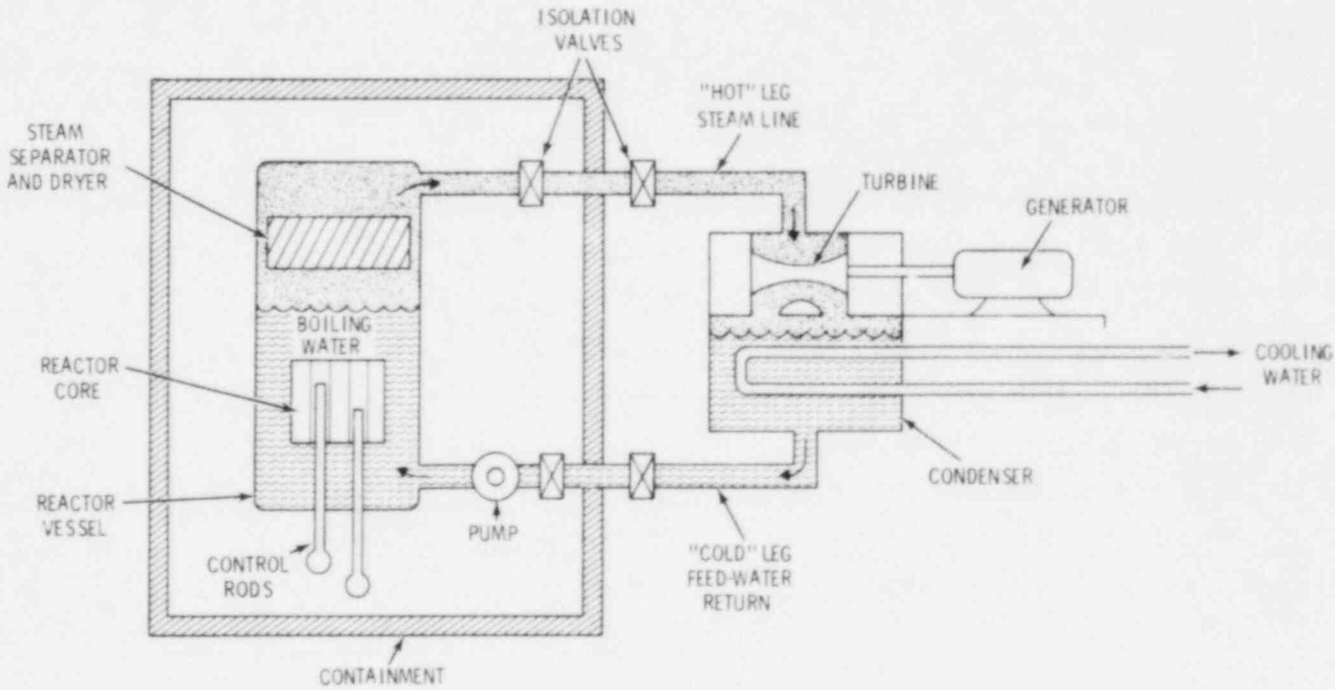


FIGURE 16.2-1. Schematic of BWR Nuclear Power Generation System

from fission product leakage through pinhole openings in fuel rods and from neutron activation of corrosion products, and returned to the reactor vessel. Makeup water is added to the coolant as needed to compensate for losses that occur in the turbine, the condenser, and the water purification system.

The reactor vessel is a right circular cylinder with a permanently attached hemispheric bottom and a removable hemispheric top, as illustrated in Figure 16.2-2. The vessel is made of carbon steel about 0.171 m thick, with the inside clad with stainless steel about 3 mm thick. The vessel top head is secured to the reactor vessel by 108 studs and nuts. The housings for the control rod drives and for reactor instrumentation extend upward through penetrations in the bottom of the reactor vessel. The approximate dimensions of the vessel are 22.2 m in height and 6.7 m in outer diameter. The mass of the vessel is nearly 750 Mg, empty.

A vertical section of the reactor building is shown in Figure 16.2-3. The building encloses the primary containment within which the reactor vessel is located, and also houses new and spent fuel pools, refueling equipment, and emergency core cooling systems.

The primary containment is a free-standing steel pressure vessel surrounded by a reinforced concrete biological shield and designed to withstand the peak transient pressures that might occur in a loss-of-coolant accident. Inside the primary containment, a drywell encloses the reactor vessel and its recirculation loops and is connected through ducts to a lower-level suppression chamber that stores a large pool of water. Under accident conditions, valves in the main steam lines from the reactor to the turbine generators are designed to close automatically. Steam escaping from the reactor system would then collect in the drywell, increasing pressure and forcing the air-steam mixture in the drywell down into the suppression pool where most of the steam would be condensed. The suppression pool also serves as one source of water for the emergency core cooling system. Additional details of the primary containment are presented in Section K.1.2 of Appendix K.

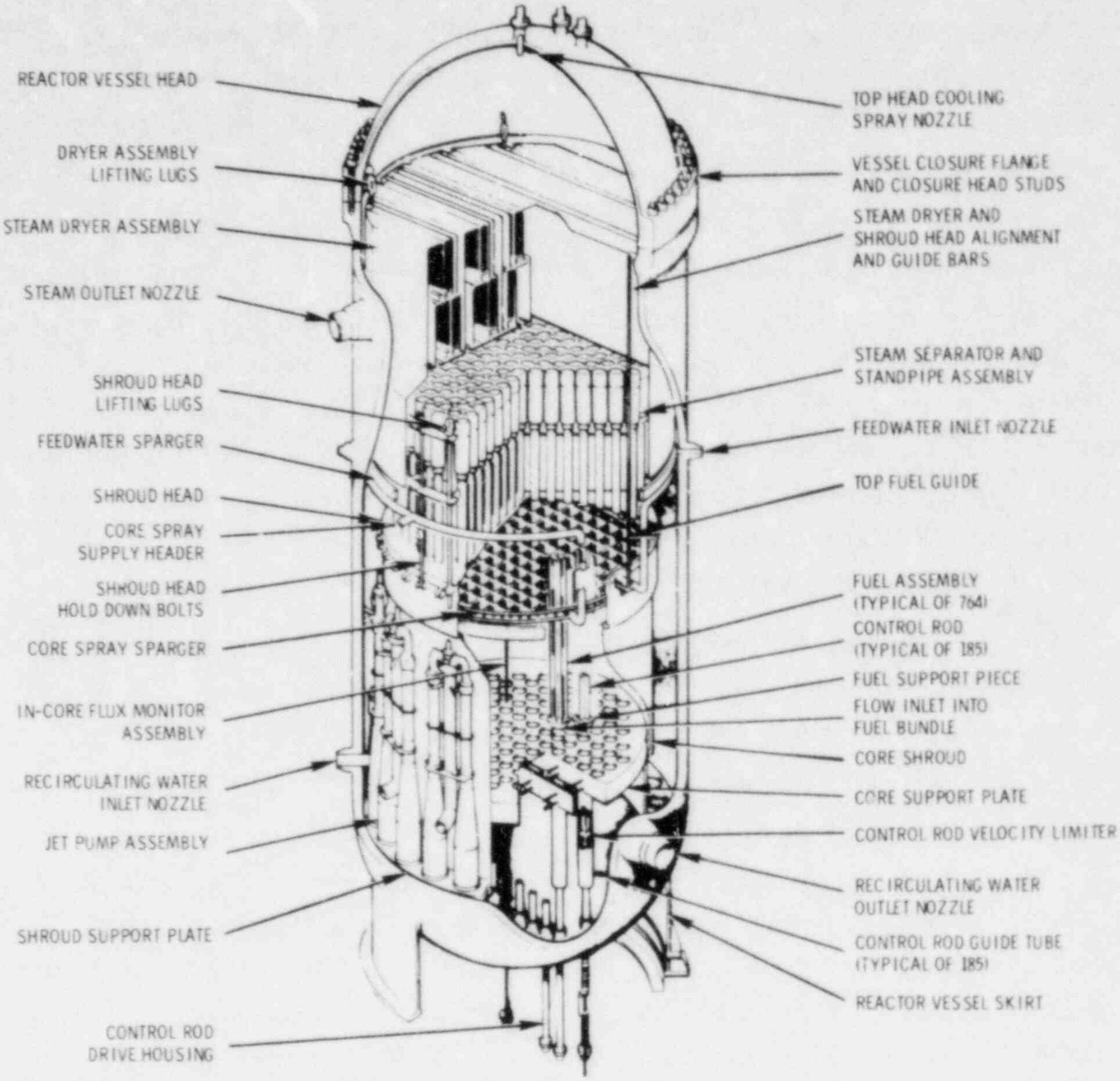


FIGURE 16.2-2. BWR Reactor Vessel and Internals

The reactor building provides a secondary containment system for the BWR. The building is rectangular in plan and elevation. The maximum exterior dimensions are 41.9 m by 52.9 m in plan, 70.2 m above grade, and 10.6 m below grade to the bottom of the foundation mat. The building is constructed of reinforced concrete up to the refueling floor level at elevation 185.0 m. Above this level, the building is constructed of insulated metal siding and

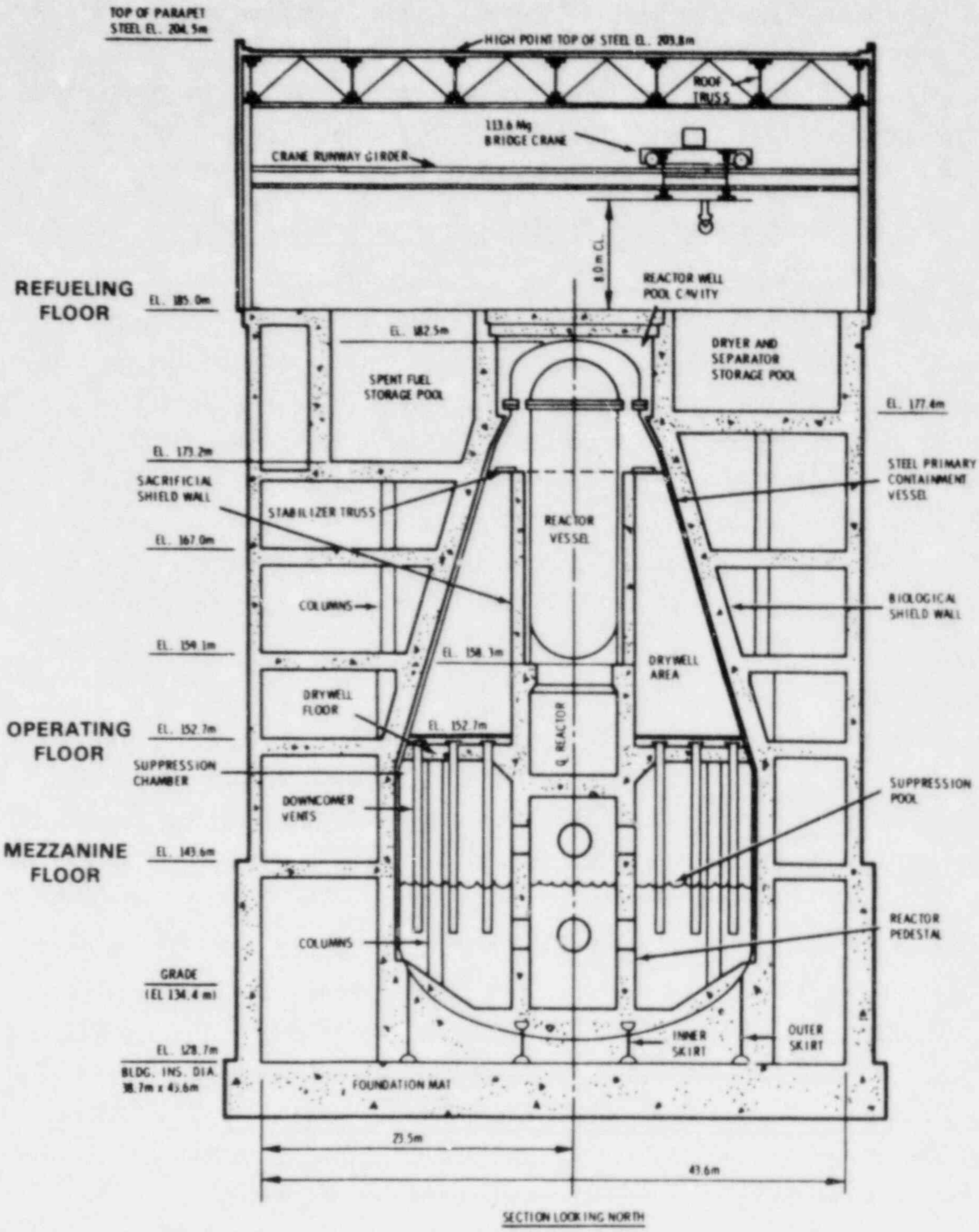


FIGURE 16.2-3. Vertical Section of BWR Reactor Building

roof decking supported by structural steel. Access to the primary containment is through a personnel lock and a 3.8-m-diameter equipment hatch at the operating floor (152.7 m) level. Additional details of the reactor building and of the systems and equipment that it contains are given in Section K.1.1 of Appendix K.

16.2.2 Radioactive Contamination Resulting from Normal Operation

Radionuclide inventories at the time of accident cleanup and decommissioning result from both normal reactor operation prior to the accident and from accident-generated fission product contamination. Information about the radioactive contamination from both of these sources is needed to assess the level of effort necessary for accident cleanup and decommissioning and to evaluate the occupational and public safety impacts of cleanup and decommissioning activities. Radioactive inventories and average radiation exposure rates due to contamination resulting from normal reactor operation are summarized in this section. Radioactive inventories and average radiation exposure rates resulting from the postulated reactor accidents are described in Section 16.3.

Estimated radionuclide inventories from neutron-activated materials in the reference BWR after 30 effective full-power years (EFPY) of operation are given in Chapter 7 of Reference 1. Neutron-activated materials include the reactor vessel and vessel internals, biological shield concrete, and corrosion products from structural components of the reactor coolant system that become activated by neutron absorption during reactor operation.

Estimated radioactive exposure rates (external exposure from gamma radiation) at selected locations in the reference BWR due to contamination resulting from normal operation are given in Appendix D of Reference 1. The exposure rates for the reference BWR are a composite of data from seven commercial BWRs that had operated for 3 to 8 years. Estimated exposure rates due to contamination from normal operation vary over a range from 0.001 to 12 R/hr, depending on location. The highest exposure rates are contact rates at external surfaces of the reactor vessel, the reactor water cleanup pumps, and the heat exchangers. General area exposure rates inside the primary containment vary from about 0.04 to 0.6 R/hr. General area exposure rates in the reactor building (the secondary containment) vary over a range from about 0.001 to 0.75 R/hr. The highest general area exposure rates due to contamination resulting from normal operation in the BWR reactor building are in the vicinity of the heat exchangers.

16.3 REFERENCE BWR ACCIDENT SCENARIOS AND RESULTANT CONTAMINATION LEVELS

The three accident scenarios that provide the bases for post-accident cleanup and decommissioning cost estimates for the reference BWR are discussed in this section. The postulated BWR accident scenarios are basically similar to those for the PWR, discussed in Section 8.3 of Chapter 8, with some differences because of differences in the BWR and PWR containment configurations.

The postulated BWR accident scenarios, listed in increasing order of the difficulty of post-accident cleanup and decommissioning, are:

1) A small loss-of-coolant accident (LOCA) in which emergency core cooling functions to cool the core and limit the release of radioactivity. The accident is postulated to result in 10% fuel cladding failure, no fuel melting, moderate contamination inside the containment vessel, no significant radioactive contamination in the reactor building, and no significant physical damage.

2) A small LOCA in which emergency core cooling is delayed, resulting in 50% fuel cladding failure and a small amount of fuel melting. The consequence scenario includes extensive radioactive contamination inside the containment vessel, minor contamination in the reactor building, and minor physical damage to equipment inside the containment.

3) A major LOCA in which emergency core cooling is delayed, resulting in 100% fuel cladding failure and significant fuel melting and core damage. The consequence scenario includes extensive radioactive contamination inside the containment vessel and in the reactor

building, and major physical damage to structures and equipment. The scenario 3 accident is also postulated to result in some contamination in the radwaste building.

Parameters that characterize conditions at the reference BWR 1 year after the postulated accidents are listed in Table 16.3-1.

The accident scenarios used as bases for this study are illustrative of a range of accident cleanup and decommissioning requirements, costs, and occupational doses. Scenarios are not restricted to accidents that have occurred, but the scenarios chosen are believed to be credible with respect to initiating circumstances, and are in agreement with scenarios currently considered as design basis by the NRC.

As shown in Figure 16.2-3, the primary containment encloses the reactor vessel, the reactor coolant recirculating loops, and the pressure suppression system. The pressure suppression system includes the drywell, suppression pool, a connecting vent system between the drywell and and the suppression pool, a containment cooling system, and valves and other service equipment. In the event of a pipe rupture or valve failure inside the primary containment, reactor water and steam would be released to the drywell air space. The resulting increase in drywell pressure would force a mixture of air, steam, and water through the vents into the suppression pool water, resulting in a rapid pressure reduction in the drywell. As a consequence of the accident and of the use of the suppression pool as a source of water for the emergency core cooling system (ECCS), the suppression pool water would become highly contaminated with radioactivity.

There are 171 penetrations through the primary containment,[1] ranging in diameter from 19.1 mm to 3.81 m, that could serve as potential paths for the release of radioactive material to the reactor building or to other areas outside of containment. The largest-diameter penetrations are for the equipment hatch and the personnel lock at the operating floor level and for the main steam lines. Smaller-diameter penetrations are for piping and instrumentation associated with the reactor water cleanup system, the residual heat removal system, etc. Two automatic isolation valves (one on each side of

TABLE 16.3-1. Reference BWR Accident Parameters

Parameter	Parmeter Value[a] Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
Percent of Fuel Cladding Failure	10	50	100
Percent of Fuel Melting	0	5	50
Volume of Suppression Pool Water (m^3)	3160[b]	3160[b]	3160[b]
Total Fission Product Radioactivity in Suppression Pool Water (Ci)	2.5×10^4	3.5×10^5	2.2×10^6
Average Fission Product Radioactivity in Suppression Pool Water (Ci/m^3)	8	110	700
Volume of Reactor Building Sump Water (m^3)	0	0	500
Total Fission Product Radioactivity in Reactor Building Sump Water (Ci)	0	0	3×10^5
Average Fission Product Radioactivity in Reactor Building Sump Water (Ci/m^3)	--	--	700
Total Fission Product Radioactivity Plated Out on Containment Vessel Surfaces (Ci)[c]	5.2	73	460
Average Fission Product Radioactivity on Containment Vessel Surfaces (Ci/m^2)			
• Floors	0.005	0.07	0.44
• Walls	0.00005	0.0007	0.0044
Average Gamma Radiation Exposure Rate at Operating Floor Level Inside Containment (R/hr)			
• Contribution from Plateout	0.052	0.720	4.6
• Contribution from Suppression Pool Water	0.006	0.070	0.5
• Total Exposure Rate	0.058	0.790	5.1
Total Fission Product Radioactivity Plated Out on Reactor Building Surfaces (Ci)	0	10	82
Average Fission Product Radioactivity on Reactor Building Surfaces (Ci/m^2)			
• Floors	--	0.001	0.008
• Walls	--	0.00001	0.00008
Average Gamma Radiation Exposure Rate at Refueling Floor level in Reactor Building (R/hr)[d]			
• Contribution from Plateout	--	0.002	0.020
• Contribution from Sump Water	--	-	0.0
• Total Exposure Rate	--	0.002	0.020

(contd on next page)

TABLE 16.3-1. (contd)

Parameter	Parameter Value(a) Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
Average Gamma Radiation Exposure Rate at Operating Floor Level in Reactor Building(e) (R/hr)			
• Contribution from Plateout	--	0.010	0.083
• Contribution from Sump Water	--	--	0.002
• Total Exposure Rate	--	0.010	0.085
Average Gamma Radiation Exposure Rate at Service Floor Level in Reactor Building(f) (R/hr)			
• Contribution from Plateout	--	0.010	0.083
• Contribution from Sump Water	--	--	30
• Total Exposure Rate	--	0.010	30
Damage to Fuel Core	Slight damage to some fuel elements as a result of fuel swelling and cladding rupture.	Oxidation of fuel cladding. Melting and fusing together of stainless steel fittings on center fuel elements. Cracking and crumbling of some fuel pellets. Melting of fuel in localized areas of central core.	Cracking, crumbling, and melting of fuel pellets. Melting and fusing together of stainless steel parts on adjacent fuel assemblies. Molten fuel present over much of core radius. Fuel and cladding fragments carried throughout water recirculation system.
Damage to Containment Vessel and Equipment	No significant physical damage.	Most electrical equipment and some valves inoperable due to water damage and corrosion. Minor structural damage.	Pipes and cable conduits dented or ripped away. Loss of electrical and other services. Recirculation system pump motors inoperable due to damage to electrical components and corrosion.
Damage to Reactor Building and Equipment	No significant physical damage.	No significant physical damage	Contamination of building ventilation system. Some electrical equipment and some valves inoperable due to water damage and corrosion. Minor structural damage. Bridge crane and refueling platform inoperable due to damage to electrical components and corrosion.
Contamination of Radwaste Building	--(g)	--(g)	Plateout on building surfaces. Reactor water cleanup demineralizer system grossly contaminated. General area radiation exposure levels about 50 mR/hr.

(a) Values refer to conditions approximately 1 year after the accident.
(b) Based on maximum water volume specified in Section C.2.1 of Reference 1.
(c) Plateout values are after washdown of walls by condensing moisture.
(d) The refueling floor level is the 185.0-m level. See Figure 16.2-3.
(e) The operating floor level is the 152.7-m level. See Figure 16.2-3.
(f) The service floor level is the 134.4-m level. See Figure 16.2-3.
(g) Contamination of radwaste building is postulated only for the scenario 3 accident.

the containment boundary) are provided in each main steam line. Each isolation valve is powered by both air pressure and a spring force. One function of these valves is to limit the release of radioactive materials outside of containment by isolating the reactor coolant pressure boundary in case of a gross release of radioactive material from the fuel to the reactor cooling water and steam. In this study, releases of radioactivity to the reactor building are postulated for both the scenario 2 and the scenario 3 accidents. The relatively large release of radioactivity and of contaminated water (approximately two reactor coolant system volumes) postulated for the scenario 3 accident is unlikely but is analyzed to provide cost and safety information on cleanup and decommissioning following such a severe accident.

The fission product source inventory for the reference BWR (e.g., the inventory of fission product radioactivity in the BWR core at the time of the hypothetical accidents) is assumed to be the same as that for the reference PWR. This source inventory, shown in Table 8.1-1 of Chapter 8, is taken from the Reactor Safety Study[3] and was calculated by means of the ORIGEN code[4] for a 1000-MWe (3200-MWt) three-region PWR core at a time when the three regions have average burnups of 8800, 17,600, and 26,400 megawatt-days per metric ton of uranium charged. BWRs typically operate at a lower specific power density than PWRs. However, because of the lower enrichment of BWR fuel, the average thermal neutron flux for both PWRs and BWRs operating at the same power levels is approximately the same. Thus, fission-product generation and transmutation by neutron absorption are approximately equivalent in both types of reactors when operating at the same power levels. The source inventory in Table 8.1-1 is therefore a good approximation for a 1000-MWe BWR with the same operating history as a 1000-MWe PWR.

Estimated inventories of radioactivity released from damaged reactor fuel during the reference PWR accidents are shown in Tables 8.2-2, 8.2-3, and 8.2-4 of Chapter 8. Assumed percentages of fuel cladding failure and fuel melting are the same for the reference BWR accidents as they are for the reference PWR accidents. Therefore, the radioactivity inventories given in Tables 8.2-2, 8.2-3, and 8.2-4 can also be used for the reference BWR accidents. These inventories form the bases for estimates of fission product contamination and radiation exposure rates shown in Table 16.3-1. The average radiation exposure rates shown in Table 16.3-1 provide adequate bases for estimating occupational radiation dose rates to cleanup and decommissioning workers following the postulated accidents.

Models and assumptions used to estimate fission product contamination and radiation exposure rates for the reference BWR accident scenarios are discussed in Appendix K of Volume 2.

16.4 ACTIVITIES AND MANPOWER REQUIREMENTS FOR ACCIDENT CLEANUP

The first activities that follow reactor accidents that result in severe radioactive contamination and possible physical damage to structures and equipment are designed to bring the accident under control and to stabilize the facility to prevent further releases of radioactivity. Initial stabilization is followed by accident cleanup, which is followed by either refurbishment or decommissioning of the reactor facility. This section provides a summary of activities and manpower requirements for accident cleanup. Activities and manpower requirements for decommissioning that follows accident cleanup are described in Section 16.6.

BWR accident cleanup has many similarities to PWR accident cleanup, which is described in Chapter 10 of this report. Chapter 10 should be referred to for the bases and assumptions behind the activities and requirements summarized here. These include such items as design of processing systems, special equipment, discharge of processed water, defueling, and waste disposal. Details of the rationale for accident cleanup are presented in Section 10.1 and in Section E.1 of Appendix E.

The goals of BWR accident cleanup are the same as those of PWR accident cleanup, namely:

1) to reduce the initial high levels of radioactive contamination present on building surfaces and in accident water, thereby reducing the radiation dose received by workers engaged in cleanup and decommissioning operations

2) to safely defuel the reactor, placing the fuel in a configuration that is safe from nuclear criticality and/or fuel meltdown

3) to collect and package for disposal the large quantities of water-soluble and otherwise readily dispersible radioactivity present in the plant.

To achieve these goals, the accident cleanup tasks postulated for the reference BWR, similar to those postulated for the reference PWR, are:

- processing of the contaminated water generated by the accident (and by decontamination operations)
- initial decontamination of building surfaces and decontamination or disposal of some equipment
- defueling of the reactor and storage of the fuel in the spent fuel pool
- cleanup of the reactor coolant recirculation system
- solidification and packaging of wastes from accident cleanup operations.

The contaminated water generated by the accident includes the suppression pool water (the suppression pool contains most of the radioactivity released in the accident) and water that collects in sumps, on floors, and in the basement of the reactor building.

Contaminated building surfaces include those in the reactor building and those inside the containment vessel. Decontamination of the inside of the containment vessel (i.e., of surfaces and equipment in the drywell and of suppression pool surfaces after the water is removed) is not essential to the task of defueling the reactor, since defueling operations are carried out from the refueling floor. However, decontamination of the containment vessel contributes to the first goal of accident cleanup and is essential for reducing radiation dose rates inside the containment vessel prior to reactor decommissioning or to reactor refurbishment and restart.

Activities and manpower requirements for BWR accident cleanup are summarized in this section. Preparations for accident cleanup are discussed in Section 16.4.1. Accident cleanup in the radwaste building, postulated to be required following the scenario 3 accident, is discussed in Section 16.4.2. Accident cleanup in the reactor building and containment vessel is discussed in Section 16.4.3. Additional details of activities and manpower requirements for accident cleanup are presented in Section K.3 of Appendix K.

16.4.1 Preparations for Accident Cleanup

Planning and preparation activities for BWR accident cleanup, similar to those for PWR accident cleanup, include:

- venting of radioactive gases (e.g., krypton-85)
- reactor building and containment vessel entry and data acquisition
- preparation of documentation for regulatory agencies
- design, fabrication, and installation of special equipment
- development of detailed work plans and procedures
- selection and training of accident cleanup staff
- removal of accumulated spent fuel from the spent fuel storage pool.

Descriptions of these activities are given in Section 10.2 of Chapter 10 and Section E.2 of Appendix E.

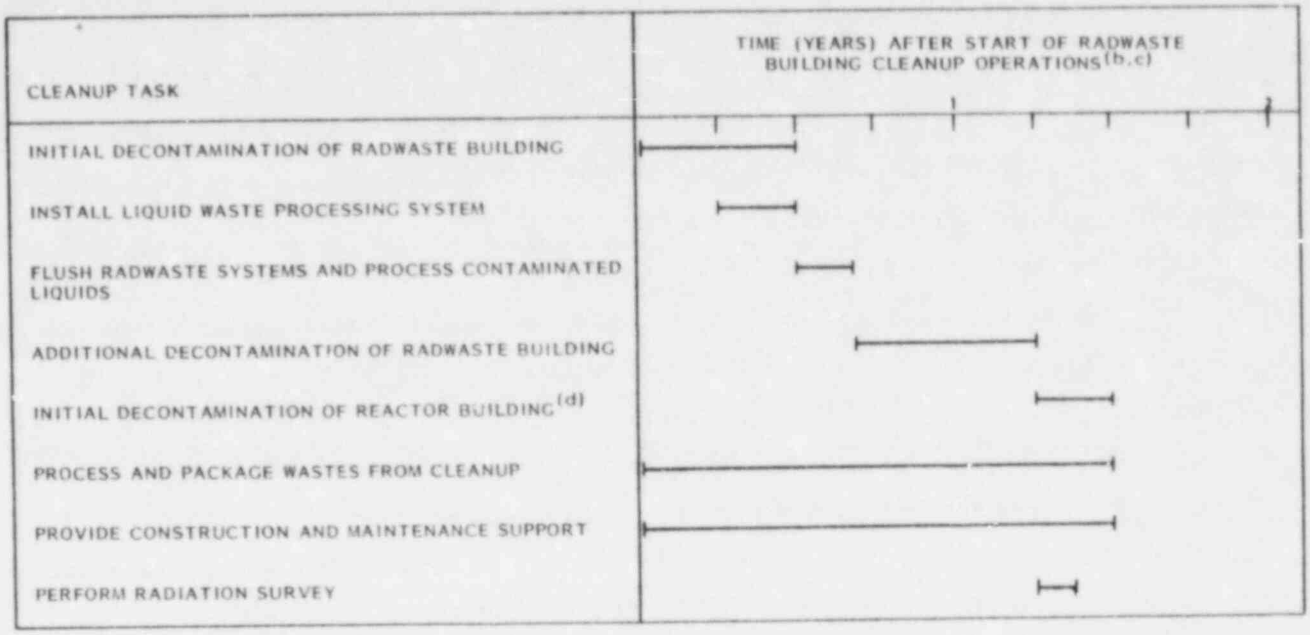
Time and manpower requirements for preparations for BWR accident cleanup are expected to be about the same as those for preparations for PWR accident cleanup. As for PWR accident cleanup, it is postulated for BWR accident cleanup that the filter/demineralizer system for processing accident water is placed in the spent fuel pool. To make space available in the pool for the filter/demineralizer system, the accumulated spent fuel from refuelings during normal operations is removed and shipped offsite. A minimum of 15 months is assumed to be required to discharge the accumulated spent fuel and ship it to an independent spent fuel storage installation (ISFSI), based on the assumptions that the pool contains 1-1/3 fuel cores at the time of the reactor accident and that 2 spent fuel rail casks are available for continuous use in transporting the fuel.

Planning and preparations activities that precede BWR accident cleanup are assumed to require 1.5 years following the scenario 1 accident, 2 years following the scenario 2 accident, and 3 years following the scenario 3 accident. The postulated utility staff organization for preparations for BWR accident cleanup is the same as that for preparations for PWR accident cleanup shown in Figure 10.2-1 of Chapter 10. Estimated utility staff labor requirements for preparations for BWR accident cleanup are shown in Table K.3-1 of Appendix K. A total of 369 man-years of utility staff labor is estimated to be required following the scenario 1 accident, 552 man-years are estimated to be required following the scenario 2 accident, and 972 man-years are estimated to be required following the scenario 3 accident. These labor requirements do not include contractor support staff to provide engineering support for preparations for accident cleanup. Engineering support staff labor requirements are estimated to be 30 man-years for preparations for cleanup following the scenario 1 accident, 80 man-years following the scenario 2 accident, and 180 man-years following the scenario 3 accident.

16.4.2 Accident Cleanup in the Radwaste Building

Fission product contamination of the radwaste building, including radioactive plateout on building and equipment surfaces and contamination of the reactor water cleanup system, is postulated for the scenario 3 accident. The radwaste building contains many systems essential to maintaining the reactor in a safe shutdown condition until defueling and to providing the services needed for efficient performance of accident cleanup operations in the reactor building and containment vessel. Decontamination is necessary to permit routine access by plant personnel to perform required operational and maintenance tasks without excessive occupational exposures.

Accident cleanup in the radwaste building is postulated to take place during preparations for cleanup in the reactor building and containment vessel and to require approximately 1.5 years to complete. A sequence and schedule for accident cleanup in the radwaste building is shown in Figure 16.4-1.



(a) SCHEDULE DETAILS ARE GIVEN IN FIGURE K.3-1 OF APPENDIX K.

(b) CLEANUP IN THE RADWASTE BUILDING IS ASSUMED TO TAKE PLACE DURING PREPARATIONS FOR CLEANUP IN THE REACTOR BUILDING AND THE CONTAINMENT.

(c) THE TOTAL TIME REQUIREMENT FOR THESE CLEANUP OPERATIONS IS ESTIMATED TO BE ABOUT 1.5 YEARS.

(d) DECONTAMINATION OF SURFACE AND EQUIPMENT, INSTALLATION OF LOCAL SHIELDING, AND REFURBISHMENT OF SYSTEMS AT REFUELING FLOOR LEVEL INSIDE REACTOR BUILDING TO PREPARE FOR REMOVAL OF ACCUMULATED FUEL FROM SPENT FUEL POOL.

FIGURE 16.4-1. Sequence and Schedule for Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident[a]

Accident cleanup tasks include:

- initial decontamination of some areas of the building to permit temporary installation of a demineralizer system for processing contaminated liquids
- installation of the demineralizer in a shielded area of the building
- operation of the demineralizer system to process contaminated liquids from the reactor water cleanup system
- additional decontamination of the radwaste building
- processing and packaging of wastes from cleanup operations
- a comprehensive radiation survey of the radwaste building.

An additional task shown in Figure 16.4-1 is some initial decontamination of the reactor building, primarily at the refueling floor level, to permit worker access to this building for the purpose of discharging the accumulated fuel assemblies from the spent fuel pool.

Accident cleanup in the radwaste building is accomplished by a staff of cleanup workers that is added to the staff for preparations for accident cleanup. The cleanup staff includes decontamination crews, crews that provide construction and maintenance support, and waste processing and waste packaging crews. Estimated cleanup worker labor requirements for accident cleanup in the radwaste building total 190 man-years and are shown in Table 16.4-1. These requirements include only the labor to complete the accident cleanup tasks and do not include operations and site support personnel who are included in the staff labor requirements for preparations for cleanup. The cleanup worker requirements shown in the table are adjusted to include the additional manpower necessary to maintain compliance with the occupational radiation dose limit, taken in this study as 5 rem/year per person.[5]

TABLE 16.4-1. Estimated Cleanup Worker Requirements for Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident[a]

Worker Category	Adjusted Worker Requirement[b] (man-yr)
Cleanup Operations Supervisor	1.5
Crew Leader	28.1
Utility Operator	42.5
Laborer	43.5
Craftsman	46.2
Health Physics Technician	28.1
Total	189.9

(a) These requirements include only the labor to complete the accident cleanup tasks and do not include operations and site support personnel who are included in the staff labor requirements for preparations for cleanup.
(b) Adjusted worker requirement to comply with an occupational radiation dose limit of 5 rem/yr. Details are given in Table K.3-4 of Appendix K.

Details of this adjusted labor requirement for compliance with occupational dose restrictions are given in Section K.3.2 of Appendix K.

16.4.3 Accident Cleanup in the Reactor Building and Containment Vessel

Accident cleanup in the BWR reactor building and containment vessel is postulated to include the following tasks (not specifically in the order listed):

- installation of the filter/demineralizer system in the spent fuel pool
- processing of contaminated liquids
- decontamination of the reactor building and containment vessel
- defueling of the reactor
- cleanup of the reactor water recirculation and reactor water cleanup systems
- treatment and disposal or storage of wastes from cleanup operations.

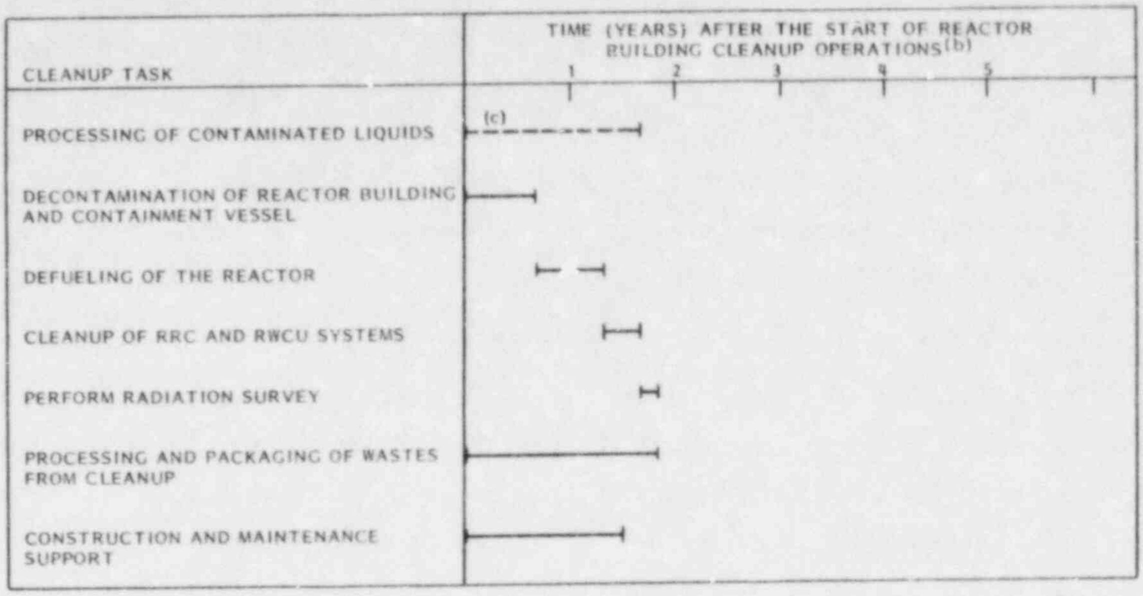
Procedures used for accident cleanup in the reference BWR are assumed to be similar to those postulated for accident cleanup in the reference PWR and described in Section 10.4.1 of Chapter 10 and Section E.4.1 of Appendix E. The filter/deminerlizer system is postulated to be similar to that used for processing PWR accident water and described in Section E.4.1.1 of Appendix E. As for the PWR, it is postulated that the filter/demineralizer system is installed in the spent fuel pool to permit ease of access and provide radiation shielding. Some preliminary decontamination and installation of temporary shielding is required in the reactor building following the scenario 2 and scenario 3 accidents to limit occupational doses to workers engaged in the installation and operation of the filter/demineralizer system.

The total building and equipment surface area requiring decontamination inside the BWR containment vessel is much smaller than the building and equipment surface area inside the PWR containment building. However, decontamination operations inside the BWR containment vessel are rendered more difficult because of the large amount of equipment in the vessel and the restrictions on the movements of personnel imposed by this equipment. This also makes it more difficult to control occupational radiation exposures during accident cleanup. Accident cleanup procedures are assumed to reduce general area radiation exposure rates to values comparable to those shown in Table 10.4-1 (Chapter 10) for the PWR.

BWR defueling operations are postulated to be similar to PWR defueling operations described in Section 10.4.1.3 of Chapter 10 and to require the use of special tools for the removal of damaged fuel assemblies. To remove the fuel from the BWR, the steam separator and dryer must first be removed from the reactor vessel. Because BWR defueling can be accomplished from the refueling floor outside the containment vessel, the radiation dose rates to workers and the difficulties associated with work in radiation areas are less for BWR defueling than for PWR defueling. However, because more fuel assemblies must be removed from the BWR than from the PWR (764 BWR assemblies versus 193 PWR assemblies), defueling of the reference BWR is postulated to take approximately twice as long as defueling of the reference PWR for similar accident scenarios, as discussed in Section K.3.3.2 of Appendix K.

Schedules for accident cleanup in the reactor building and containment vessel following the postulated BWR accidents are shown in Figures 16.4-2, 16.4-3, and 16.4-4. Accident cleanup is estimated to require approximately 1.7 years following the scenario 1 accident, 3.3 years following the scenario 2 accident, and 5.3 years following the scenario 3 accident. Details of accident cleanup schedules and estimated cleanup worker requirements for completion of individual accident cleanup tasks are presented in Section K.3.3.2 of Appendix K.

The postulated utility staff organization for accident cleanup in the BWR reactor building and containment vessel is the same as the staff organization for PWR accident cleanup shown in Figure 10.4-4 of Chapter 10. The utility staff includes a plant operations branch and several site support branches (e.g., engineering, health and safety, security, contracts and accounting, and quality assurance) as well as the staff actually involved in cleanup of the

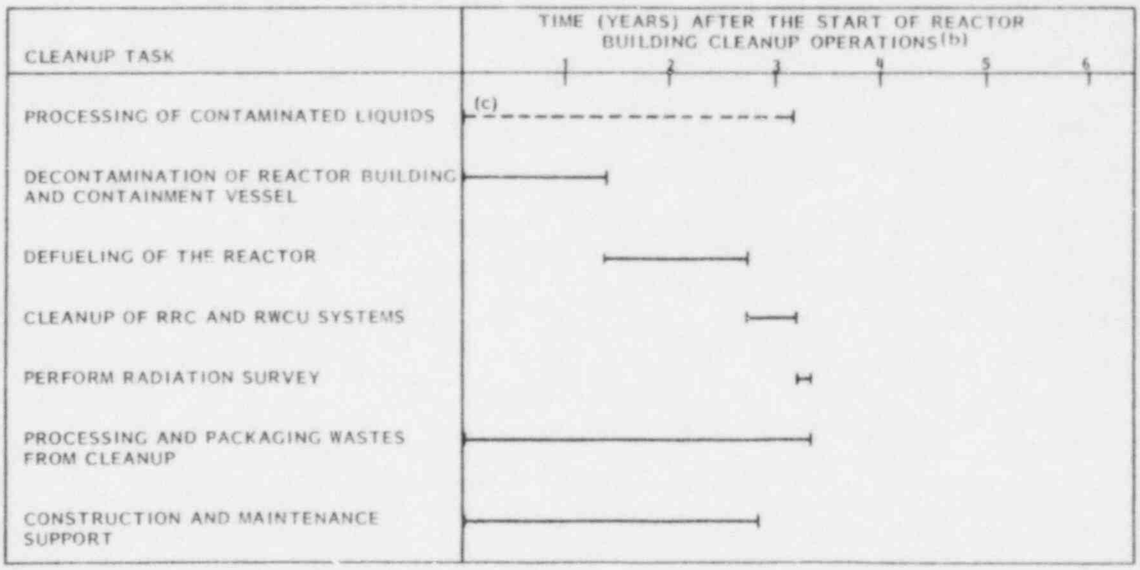


(a) SCHEDULE DETAILS ARE GIVEN IN FIGURE K.3-2 OF APPENDIX K.

(b) THE TOTAL TIME REQUIREMENT FOR ACCIDENT CLEANUP IN THE REACTOR BUILDING FOLLWING THE SCENARIO 1 ACCIDENT IS 1.7 YEARS.

(c) AS REQUIRED DURING THIS TIME PERIOD.

FIGURE 16.4-2. Sequence and Schedule for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 1 Accident[a]

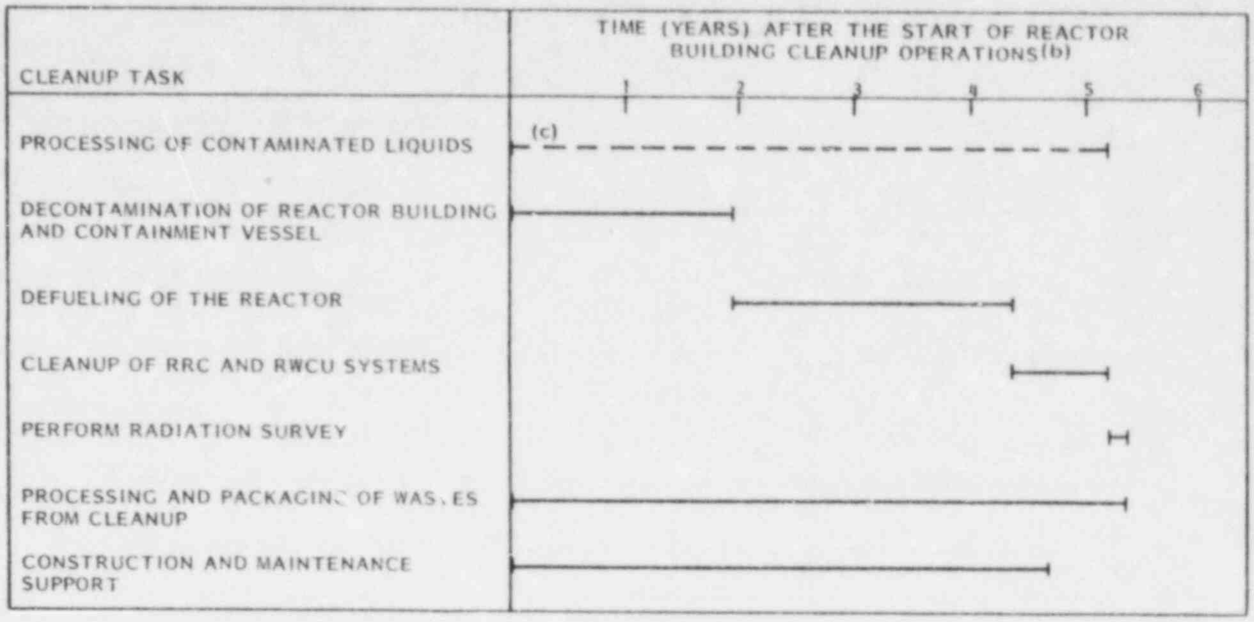


(a) SCHEDULE DETAILS ARE GIVEN IN FIGURE K.3-3 OF APPENDIX K.

(b) THE TOTAL TIME REQUIREMENT FOR ACCIDENT CLEANUP IN THE REACTOR BUILDING FOLLOWING THE SCENARIO 2 ACCIDENT IS 3.3 YEARS.

(c) AS REQUIRED DURING THIS TIME PERIOD.

FIGURE 16.4-3. Sequence and Schedule for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 2 Accident[a]



(a) SCHEDULE DETAILS ARE GIVEN IN FIGURE K.3-4 OF APPENDIX K.

(b) THE TOTAL TIME REQUIREMENT FOR ACCIDENT CLEANUP IN THE REACTOR BUILDING FOLLOWING THE SCENARIO 3 ACCIDENT IS 5.3 YEARS.

(c) AS REQUIRED DURING THIS TIME PERIOD.

FIGURE 16.4-4. Sequence and Schedule for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 3 Accident[a]

reactor building and containment vessel. Estimated utility staff labor requirements for BWR accident cleanup are shown in Table 16.4-2. These labor requirements are 657 man-years for cleanup following the scenario 1 accident, 1613 man-years for cleanup following the scenario 2 accident, and 3658 man-years for cleanup following the scenario 3 accident. The accident cleanup staff labor requirements (the man-years for personnel engaged in cleanup operations inside the reactor building and containment vessel) shown in Table 16.4-2 have been adjusted upward by appropriate factors to ensure that the estimated occupational radiation dose for individual workers does not exceed 5 rem/year.[5] An explanation of the adjustment factors used to obtain man-years for accident cleanup staff labor is given in Section K.3.3.4 of Appendix K.

TABLE 16.4-2. Estimated Utility Staff Labor Requirements for Accident Cleanup in the Reactor Building Following the Postulated BWR Accidents

Position	Utility Staff Labor Requirements (man-years) for Cleanup Following		
	Scenario 1 Accident[a]	Scenario 2 Accident[b]	Scenario 3 Accident[c]
Plant Superintendent	1.7	3.3	5.3
Assistant Plant Superintendent	1.7	3.3	5.3
Consultants	5.1	19.8	53.0
Secretaries and Word Processors	13.6	33.0	106.0
Site Support Staff			
Health and Safety Supervisor	1.7	3.3	5.3
Health Physicist	1.7	3.3	5.3
Senior Health Physics Technician	13.6	26.4	63.6
Health Physics Technician[d]	13.6	26.4	63.6
Protective Equipment Attendant	6.8	26.4	63.6
Industrial Safety Specialist	1.7	3.3	5.3
Industrial Safety Technician	3.4	6.6	10.6
Security Supervisor	1.7	3.3	5.3
Security Snift Supervisor	6.8	13.2	21.2
Security Patrolman	81.6	158.4	254.4
Contacts & Accounting Supervisor	1.7	3.3	5.3
Accountant	1.7	3.3	10.6
Contracts Specialist	1.7	3.3	5.3
Insurance Specialist	1.7	3.3	10.6
Procurement Specialist	1.7	3.3	5.3
Clerk	3.4	13.2	31.8
Quality Assurance Supervisor	1.7	3.3	5.3
Quality Assurance Engineer	3.4	6.6	10.6
Quality Assurance Technician	3.4	6.6	10.6
Construction Engineering Supervisor	1.7	3.3	5.3
Engineer	10.2	26.4	63.6
Estimator	1.7	6.6	21.2
Draftsman	3.4	13.2	31.8
Subtotals	170.0	366.3	715.5
Plant Operations Staff			
Plant Operations Supervisor	1.7	3.3	5.3
Plant Chemist	1.7	3.3	5.3
Chemist	3.4	6.6	10.6

(contd on next page)

TABLE 16.4-2. (contd)

Position	Utility Staff Labor Requirements (man-years) for Cleanup Following Scenario 1 Accident[a]	Scenario 2 Accident[b]	Scenario 3 Accident[c]
Plant Operations Staff (contd)			
Reactor Operations Engineer	1.7	3.3	5.3
Engineer	3.4	6.6	10.6
Reactor Operations Shift Supervisor	6.8	13.2	21.2
Senior Reactor Operator	13.6	26.4	42.4
Reactor Operator	27.2	52.8	84.8
Utility Operator	27.2	52.8	84.8
Technician	27.2	66.0	127.2
Craft Supervisor	1.7	3.3	5.3
Crew Foreman	6.8	13.2	21.2
Craftsman[e]	13.6	39.6	63.6
Warehouseman	6.8	26.4	42.4
Tool Crib Attendant	6.8	26.4	42.4
Subtotals	149.6	343.2	572.4
Accident Cleanup Staff			
Cleanup Superintendent	1.7	3.3	5.3
Radioactive Shipment Specialist	1.7	3.3	5.3
Clerk	1.7	3.3	10.6
Shift Supervisor	6.8	13.2	21.2
Crew Leader[f]	30.5	91.1	227.0
Utility Operator[f]	110.8	266.9	679.0
Laborer[f]	62.0	153.3	374.9
Craftsman[f]	61.2	199.5	585.2
Health Physics Technician[f]	39.1	109.9	291.9
Subtotals	315.5	843.8	2200.4
Totals	657.2	1612.7	3657.9

(a) Based on an estimated cleanup time requirement of 1.7 years.
(b) Based on an estimated cleanup time requirement of 3.3 years.
(c) Based on an estimated cleanup time requirement of 5.3 years.
(d) Additional health physics technicians counted as part of accident cleanup staff.
(e) Additional craftsmen counted as part of accident cleanup staff.
(f) Cleanup staff labor requirements are adjusted to limit individual radiation doses to 5 rem/yr.

The staff labor requirements shown in Table 16.4-2 do not include contractor labor to provide engineering support services during accident cleanup operations. Engineering support staff labor requirements are estimated to be 17 man-years (10 man-years per year) during cleanup following the scenario 1 accident, 66 man-years (20 man-years per year) during cleanup following the scenario 2 accident, and 159 man-years (30 man-years per year) during cleanup following the scenario 3 accident.

16.5 COSTS OF ACCIDENT CLEANUP

The costs of accident cleanup in the reference BWR following the postulated accidents are summarized in this section. These costs are developed in detail in Section K.4 of Appendix K and are based on unit costs discussed in Appendix I. Costs are in early-1981 dollars and include a 25% contingency. Costs of decommissioning following accident cleanup are summarized in Section 16.7.

As discussed in earlier sections of this study, accident cleanup activities would be similar whether the reactor is refurbished for restart or decommissioned. Hence the costs of accident cleanup presented in this chapter are considered to be a good representation independent of the ultimate use of the plant. Costs of activities related to refurbishment and restart of a reactor, beyond the accident cleanup activities described here, are not included in this cost summary.

The costs of accident cleanup are based on the key study bases and assumptions listed in Section 4.2 of Chapter 4. Additional bases and assumptions used to develop accident cleanup costs are given in Chapter 11 where PWR accident cleanup costs are discussed. Chapter 11 should be referenced for the bases and assumptions behind the BWR accident cleanup costs described in this section.

Total estimated costs and estimated time requirements for BWR accident cleanup are shown in Table 16.5-1. Accident cleanup following the scenario 1 accident at the reference BWR is estimated to cost $128 million and to require 3.2 years for completion. Accident cleanup following the scenario 2 accident at the reference BWR is estimated to cost $228 million and to require 5.3

TABLE 16.5-1. Summary of Time and Cost Estimates for Accident Cleanup at the Reference BWR Following the Postulated Accidents

	Scenario 1 Accident		Scenario 2 Accident		Scenario 3 Accident	
	Time (years)	Cost ($ millions)[a]	Time (years)	Cost ($ millions)[a]	Time (years)	Cost ($ millions)[a]
Preparations for Accident Cleanup	1.5	30.1	2.0	49.7	3.0	90.3
Accident Cleanup in the Radwaste Building	--[b]	--[b]	--[b]	--[b]	--[c]	13.1[d]
Accident Cleanup in the Reactor Building & Containment	1.7	98.4	3.3	178.5	5.3	317.5
Totals	3.2	128.5	5.3	228.2	8.3	420.9

(a) Costs are in early-1981 dollars and include a 25% contingency.
(b) Accident cleanup in the radwaste building is not postulated following the scenario 1 and scenario 2 accidents.
(c) Accident cleanup in the radwaste building following the scenario 3 accident is postulated to be completed during preparations for cleanup in the reactor building.
(d) Includes the costs of cleanup worker labor, waste management, equipment, supplies, and services for accident cleanup in the radwaste building. Management and support staff costs and incidental costs (e.g., energy, insurance, etc.) are included in the costs of preparations for accident cleanup.

years for completion. Accident cleanup following the scenario 3 accident at the reference BWR is estimated to cost $421 million and to require 8.3 years for completion. These costs and times include those for planning and preparation as well as for the actual cleanup operations. Accident cleanup costs for the scenario 3 accident include the costs of accident cleanup in the radwaste building as well as the costs of accident cleanup in the reactor building and containment vessel.

Accident cleanup costs are shown by cost category in Table 16.5-2 to illustrate the relative importance of individual cost items. The major cost item for accident cleanup is labor. Staff labor costs account for about 40 to 50% of accident cleanup costs, depending on accident scenario. Engineering support costs are an additional labor cost. Engineering support costs constitute more than 90% of the specialty contractor costs in Table 16.5-2.

Other major accident cleanup costs include energy costs, waste management costs, and the costs of special equipment and facilities. Significant quantities of electrical energy and fuel oil are required to heat buildings, operate the systems to keep the reactor in a safe shutdown condition following an accident, and provide support services during accident cleanup operations. Costs of waste management are based on disposal assumptions discussed in Section 10.4.1.5 of Chapter 10 and in Section E.4.1.5 of Appendix E. The major contributor to waste management costs is the cost of shipment and disposal of the damaged fuel removed from the reactor during defueling operations.

TABLE 16.5-2. Summary of Accident Cleanup Costs at the Reference BWR by Cost Category

Cost Category	Accident Cleanup Following Scenario 1 Accident		Accident Cleanup Following Scenario 2 Accident		Accident Cleanup Following Scenario 3 Accident	
	Estimated Costs ($ millions)	Percent of Total	Estimated Costs ($ millions)	Percent of Total	Estimated Costs ($ millions)	Percent of Total
Preparations for Accident Cleanup						
Utility Staff Labor	16.199	53.7	24.381	49.0	43.771	48.6
Waste Management	0.188	0.6	0.444	0.9	0.569	0.6
Energy	4.845	16.1	6.460	13.0	9.690	10.7
Special Equipment and Facilities	2.074	6.9	4.324	8.7	7.593	8.4
Miscellaneous Supplies	0.094	0.3	0.125	0.3	0.188	0.2
Speciality Contractors	3.923	13.0	10.231	20.6	22.846	25.3
Nuclear Insurance & License Fees	2.821	9.4	3.751	7.5	5.610	6.2
Subtotals for Preparations for Cleanup	30.144	100.0	49.716	100.0	90.267	100.0
Accident Cleanup in the Radwaste Building						
Cleanup Worker Labor					8.040	61.3
Waste Management					1.005	7.7
Special Tools and Equipment					1.500	11.4
Miscellaneous Supplies					1.094	8.3
Special Contractors					1.488	11.3
Subtotals for Cleanup in the Radwaste Building					13.127	100.0
Accident Cleanup in the Reactor Building and Containment						
Operations and Support Staff Labor	15.168	15.4	34.381	19.3	66.136	20.8
Accident Cleanup Staff Labor	14.218	14.5	38.116	21.4	100.915	31.8
Waste Management[e]	7.979	8.1	19.188	10.8	27.341	8.6
Disposal of Fuel from Reactor Defueling[e]	42.020	42.7	42.145	23.5	42.395	13.3
Energy	6.203	6.3	12.[illegible]	6.7	18.773	5.9
Special Tools and Equipment	3.781	3.8	7.813	4.4	17.063	5.4
Miscellaneous Supplies	2.571	2.6	8.910	5.[illegible]	12.891	4.1
Specialty Contractors	2.665	2.7	9.351	5.[illegible]	21.771	6.9
Nuclear Insurance & License Fees	3.750	3.8	6.540	3.7	0.259	3.2
Subtotals for Cleanup in the Reactor Building & Containment	98.355	100.0	178.462	100.0	317.544	100.0
Total Accident Cleanup Costs	128.5		228.2		420.9	

(a) Costs are in early-1981 dollars and include a 25% contingency.
(b) Number of figures shown is for computational accuracy only and does not imply precision to the nearest one thousand dollars.
(c) Costs are based on assumed time periods of 1.5 years for preparations for cleanup following the scenario 1 accident, 2 years for preparations for cleanup following the scenario 2 accident, and 3 years for preparations for cleanup following the scenario 3 accident.
(d) Accident cleanup in the radwaste building following the scenario 3 accident is assumed to be accomplished during preparations for cleanup in the reactor building. Management and support staff costs and other incidental costs are included in the costs of preparations for cleanup.
(e) Costs for disposal of fuel are shown separately from other waste management costs.

Major equipment items needed for accident cleanup include the filter/demineralizer system for processing accident water, facilities for the interim storage of wastes, and the special tools needed to defuel the reactor.

In general, accident cleanup costs at the reference BWR are estimated to be comparable to accident cleanup costs at the reference PWR, with total cleanup costs for the BWR slightly higher than total cleanup costs for the PWR. Costs that are higher for the BWR than for the PWR include:

a) Support staff labor costs, which are related to time requirements for the completion of accident cleanup.

b) Accident cleanup staff labor costs, which are higher for the BWR because of the greater labor requirement for defueling the reactor.

c) Waste management costs, which are higher because disposal costs for the BWR fuel core are higher than disposal costs for the PWR fuel core.

Costs that are higher for the PWR than for the BWR include:

a) Energy costs (energy costs are site and reactor-specific)

b) Accident cleanup costs in the auxiliary and fuel buildings, which are higher than accident cleanup costs in the radwaste building.

As discussed in Section 11.6 of Chapter 11, the costs of accident cleanup are sensitive to various factors that include:

- the potential for delay in accident cleanup activities due to various causes such as greater core damage or contamination than expected, inability to dispose of radioactive wastes, financial difficulties, social and political concerns, regulatory constraints, etc.

- the need for complicated and expensive equipment for processing radioactive wastes resulting from the accident or for defueling the reactor

- the need to construct additional facilities such as buildings to house equipment or storage facilities for temporary onsite storage of radioactive wastes
- differences in plant design and plant location.

The sensitivity of PWR accident cleanup costs to these various factors is discussed in Section 11.6. Similar considerations are expected to apply to BWR accident cleanup costs, and the interested reader is referred to Section 11.6.

16.6 ACTIVITIES AND MANPOWER REQUIREMENTS FOR DECOMMISSIONING

This section summarizes the technical requirements and manpower needs for post-accident decommissioning at the reference BWR via the DECON, SAFSTOR, and ENTOMB alternatives. BWR post-accident decommissioning has many similarities to PWR post-accident decommissioning, which is discussed in Chapter 12 of this report. Chapter 12 should be referred to for the bases and assumptions behind the activities and requirements discussed here.

The actual decommissioning of an accident-damaged BWR begins following the completion of accident cleanup activities. During the accident cleanup campaign, some tasks that would be part of normal decommissioning are completed and other tasks are partially completed. Examples of tasks that are completed during cleanup include defueling the reactor, decontamination of the reactor water recirculation system, and a comprehensive radiation survey of the plant. Examples of tasks that are partially completed include the decontamination of building surfaces in the reactor building and the containment vessel and the removal and segmentation of reactor vessel internals.

Accident cleanup also results in some new tasks that must be completed during decommissioning. These new tasks include the removal of new equipment installed to process accident water and the decommissioning of temporary onsite waste storage structures specially constructed for the interim storage of accident cleanup wastes.

Many decommissioning tasks are common to both post-accident and normal-shutdown decommissioning. However, changes in the physical and radiological condition of the plant resulting from an accident can result in substantial changes in time and manpower requirements for post-accident decommissioning. Radiation doses to workers during post-accident decommissioning are likely to be higher than those following normal shutdown because of increased contamination of equipment and building surfaces. Physical damage to the plant may compromise some systems and equipment needed for the performance of decommissioning tasks, thus necessitating repairs or substitutions and increasing the time and cost of decommissioning.

Activities and manpower requirements for BWR post-accident decommissioning are summarized in this section. Preparations for decommissioning are summarized in Section 16.6.1. Decommissioning via the DECON, SAFSTOR, and ENTOMB alternatives is discussed in Sections 16.6.2, 16.6.3, and 16.6.4, respectively. Additional details of activities and manpower requirements for BWR post-accident decommissioning are given in Section K.5 of Appendix K. The activities and requirements for normal shutdown decommissioning, described in detail in Reference 1, provide a basis for the decontamination and dismantlement requirements, schedules, and manpower estimates summarized here. Changes have been made in schedules and manpower estimates, where appropriate, to account for differences in post-accident and normal-shutdown decommissioning.

16.6.1 Planning and Preparation Activities

Planning and preparation activities for BWR post-accident decommissioning, similar to those for PWR post-accident decommissioning, include:

- satisfying regulatory requirements
- gathering and analyzing data
- developing detailed work plans and procedures
- designing, procuring, and testing special equipment
- selecting and training staff
- selecting specialty contractors
- installation of HEPA filters.

Descriptions of these activities are given in Section K.5.1 of Appendix K and in Section H.2 of Reference 1.

Planning and preparation for post-accident decommissioning is assumed to take place during the final 1.5 years of accident cleanup. In addition to key supervisory personnel who are added to the staff to direct the planning effort, these activities utilize personnel who are available from the accident cleanup staff.

16.6.2 DECON Activities and Manpower Requirements

The decontamination and dismantlement activities during post-accident DECON at the reference BWR are similar to the activities during DECON following normal shutdown, described in Appendix I of Reference 1. These activities include:

- decontamination of the surfaces of process systems and equipment
- disassembly and disposal of neutron-activated components, including the reactor vessel and vessel internals
- disassembly and disposal of contaminated equipment, including ductwork, piping, and pool liners
- removal of contaminated concrete
- packaging and shipment of radioactive wastes to a waste disposal site
- a final radiation survey.

Some of these activities are initiated during accident cleanup. However, the bulk of this work is carried out during DECON, particularly the removal of large equipment components and of contaminated structural material.

Radioactive contamination levels in the reactor building and the containment vessel during post-accident DECON exceed those that would be present following normal shutdown by amounts that depend on the severity of the accident and on the particular location in the reactor building or the containment. To reduce radiation doses to decommissioning workers to practicable levels, the major access routes used by these workers and "hot spots" outside of the access routes that can affect worker doses are cleaned up or shielded.

This task is undertaken at the start of DECON to obtain the maximum dose-reduction benefit, using the same methods postulated for accident cleanup.

The requirements for post-accident DECON in the radwaste and control building, the turbine-generator building, and in other plant structures are assumed to be about the same as for normal-shutdown DECON.

Some radioactive contamination of onsite structures used for interim storage of the radioactive wastes from accident cleanup is expected because of package failures, smearable contamination on package surfaces, etc. Therefore, these structures require decontamination before DECON is completed.

The schedule for DECON at the reference BWR following a scenario 2 accident is shown in Figure 16.6-1. DECON begins in the reactor building and the containment vessel, which comprises the major effort by the decommissioning staff. The work proceeds through the turbine-generator building, the radwaste and control building, and other structures as staff become available and as

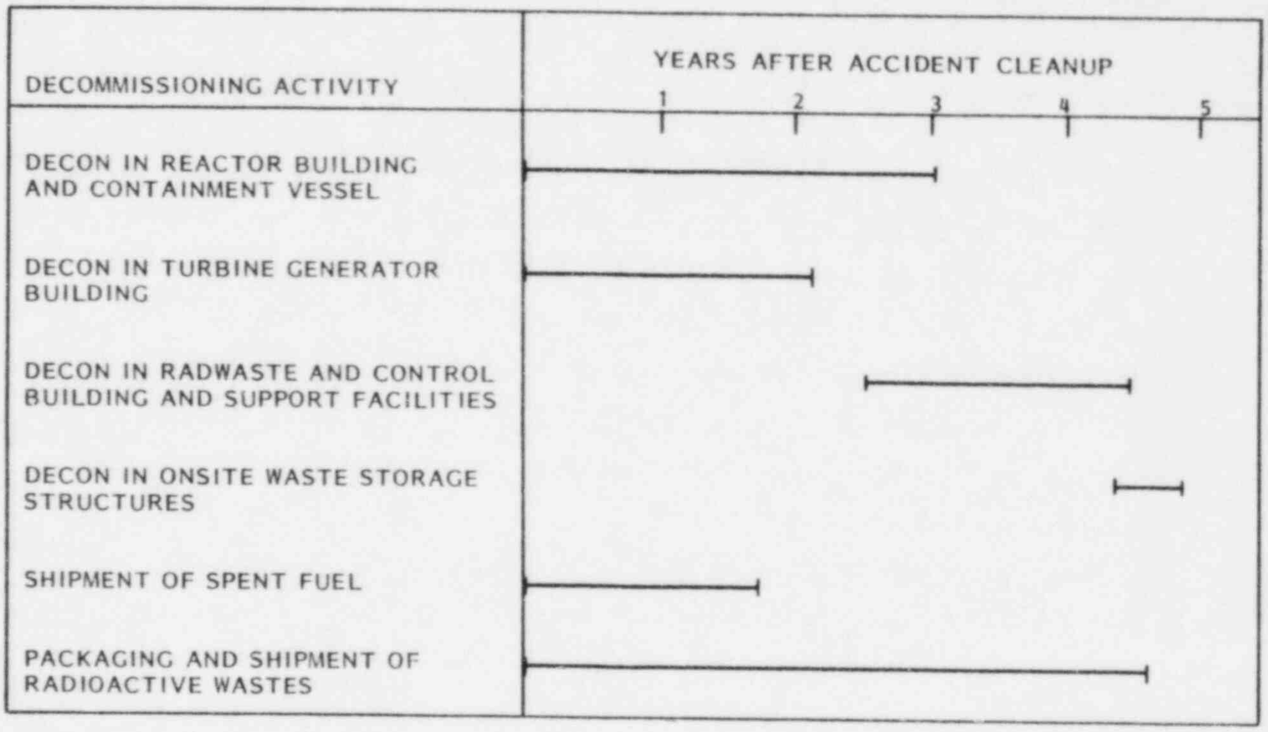


FIGURE 16.6-1. Overall Schedule and Sequence for DECON at the Reference BWR Following a Scenario 2 Accident

the various systems in these other buildings complete their required service functions. DECON following a scenario 2 accident and the subsequent accident cleanup is estimated to require about 4.8 years for completion. Variations in accident severity, within the range of accident scenarios considered in this study, are estimated to change the duration of post-accident DECON at the reference BWR by about ± 0.2 years.

The postulated utility staff organization for BWR decommissioning is the same as the staff organization for PWR decommissioning shown in Figure 12.2-2 of Chapter 12. The utility staff includes several support branches as well as the staff actually engaged in DECON in the reactor building and other plant structures. Estimated utility staff labor requirements for DECON at the reference BWR following a scenario 2 accident are shown in Table 16.6-1. The total staff labor requirement for DECON following a scenario 2 accident is estimated to be about 1070 man-years. Decommissioning worker labor requirements shown in Table 16.6-1 have been adjusted upward by appropriate factors to ensure that the estimated occupational radiation dose for individual workers does not exceed 5 rem/year.

Because management and support staff labor requirements are a function of project duration and the duration of DECON does not vary substantially with accident severity, management and support staff labor requirements do not vary greatly with accident severity. Decommissioning worker requirements vary with accident severity because of estimated variations in the occupational radiation doses received by the decommissioning workers and the individual radiation dose limitations. Decommissioning worker manpower requirements are estimated to be about one-half as great following a scenario 1 accident as following a scenario 2 accident, and about 2 times as great following a scenario 3 accident as following a scenario 2 accident. The estimated total staff labor requirement for DECON following a scenario 1 accident is approximately 670 man-years and for DECON following a scenario 3 accident is approximately 1850 man-years.

TABLE 16.6-1. Overall Staff Labor Requirements for DECON at the Reference BWR Following a Scenario 2 Accident

Position	Staff Labor Requirement (man-years) in Decommissioning Phase:[a] Planning and Preparation	DECON	Total Staff Labor Required (man-years)
Management and Support Staff			
Decommissioning Superintendent	1.5	5.1[b]	6.6
Secretary	3.0	14.7[b]	17.7
Clerk	1.0	9.6	10.6
Decommissioning Engineer	1.5	5.1[b]	6.6
Assistant Decommissioning Engineer	1.5	4.8	6.3
Radioactive Shipment Specialist	0	4.8	4.8
Procurement Specialist	0	4.8	4.8
Tool Crib Attendant	0	9.6	9.6
Reactor Operator[c]	0	38.4	38.4
Security Supervisor	0	4.8	4.8
Security Shift Supervisor	0	19.2	19.2
Security Patrolman	0	57.6	57.6
Contracts & Accounting Supervisor	0	5.1[b]	5.1
Health & Safety Supervisor	0	5.1[b]	5.1
Health Physicist	0	4.8	4.8
Protective Equipment Attendant	0	9.6	9.6
Industrial Safety Specialist	0	4.8	4.8
Quality Assurance Supervisor	0	5.1[b]	5.1
Quality Assurance Engineer	0	4.8	4.8
Quality Assurance Technician	0	19.2	19.2
Consultant (Safety Review)	0	2.4	2.4
Instrument Technician[d]	0	19.2	19.2
Maintenance Mechanic[d]	0	19.2	19.2
Warehouseman	0	9.6	9.6
Subtotals	8.5	287.4	295.9
Decommissioning Workers			
Shift Engineer	0	9.6	9.6
Crew Leader[e]	0	103.1	103.1
Utility Operator[e]	0	224.5	224.5
Laborer[e]	0	154.0	154.0
Craft Supervisor	0	19.2	19.2
Craftsman[e]	0	146.7	146.7
Senior Health Physics Technician	0	19.2	19.2
Health Physics Technician[e]	0	95.7	95.7
Subtotals	0	772.0	772.0
Totals	8.5	1059.4	1067.9

(a) Rounded to the nearest 0.1 man-year.
(b) Includes an additional 4 months following active decommissioning to complete the documentation and other unspecified license and contract termination requirements.
(c) Based on two operators per shift in the control room, three shifts per day, 7 days per week.
(d) Based on one per shift, three shifts per day, 7 days per week to maintain essential services.
(e) From Table K.5-7 of Appendix K.

16.6.3 SAFSTOR Activities and Manpower Requirements

Post-accident SAFSTOR includes preparations for safe storage of the accident-damaged facility, continuing care for a specified period during which the radioactivity within the plant is allowed to decay, and eventual deferred decontamination of the facility. An advantage of SAFSTOR is that it satisfies the requirements for protection of the public while reducing initial commitments of time, money, occupational radiation dose, and offsite waste disposal space compared to DECON. Disadvantages of SAFSTOR include the need to maintain the nuclear license during a period of safe storage and the absence of personnel familiar with the plant and the accident to assist in deferred decontamination. The decay of radioactive contamination within the stored facility is slower following an accident than it is following normal shutdown because post-accident radioactive decay is dominated by ^{137}Cs with a 30-year half-life rather than by ^{60}Co with a 5.27-year half-life.

16.6.3.1 Preparations for Safe Storage

Activities during post-accident preparations for safe storage at the reference BWR are similar to the activities during preparations for safe storage following normal shutdown, described in Appendix J of Reference 1. These activities include:

- decontamination, deactivation, and sealing of systems, equipment items, and plant areas
- fixation of surface contamination
- transfer of contaminated materials
- decontamination and isolation of contaminated plant areas
- installation of barriers and monitoring systems needed during the period of continuing care.

Some of these activities, particularly the chemical decontamination of water treatment and recirculation systems and the initial decontamination of building surfaces and equipment, are initiated during accident cleanup that precedes decommissioning. Additional decontamination and shielding of "hot spots" in the reactor building and the containment vessel, beyond that required for decommissioning following normal shutdown, is required at the start of preparations for safe storage to reduce the radiation dose to workers engaged in decommissioning operations. The methods used for building decontamination during post-accident preparations for safe storage are generally the same as those used during accident cleanup described in Appendix E.

The requirements for post-accident preparations for safe storage in the turbine-generator building, the radwaste and control building, and site and support facilities are assumed to be about the same for post-accident SAFSTOR as they are for normal-shutdown SAFSTOR. The requirements for preparations for safe storage of onsite structures for interim storage of radioactive wastes at the reference BWR are assumed to be similar to those for preparations for safe storage of these facilities at the reference PWR, described in Appendix G of Volume 2.

The schedule for preparations for safe storage at the reference BWR following a scenario 2 accident is shown in Figure 16.6-2. As with DECON, the preparations for safe storage phase of SAFSTOR begins in the reactor building and the containment vessel and proceeds through the other buildings as staff are available and as the various systems involved complete their required service functions. Preparations for safe storage following a scenario 2 accident and the subsequent accident cleanup are estimated to require about 2.8 years for completion. Variations in accident severity, within the range of accident scenarios considered in this study, are estimated to change the duration of preparations for safe storage by about $\pm$0.1 years.

The organization and the individual functions of the decommissioning staff for preparations for safe storage are the same as those for DECON. (See Section 12.2.4 of Chapter 12.) Estimated utility staff labor requirements for preparations for safe storage at the reference BWR following a scenario 2 accident are shown in Table 16.6-2. The total staff labor requirement for preparations for safe storage following a scenario 2 accident is estimated to be about 425 man-years.

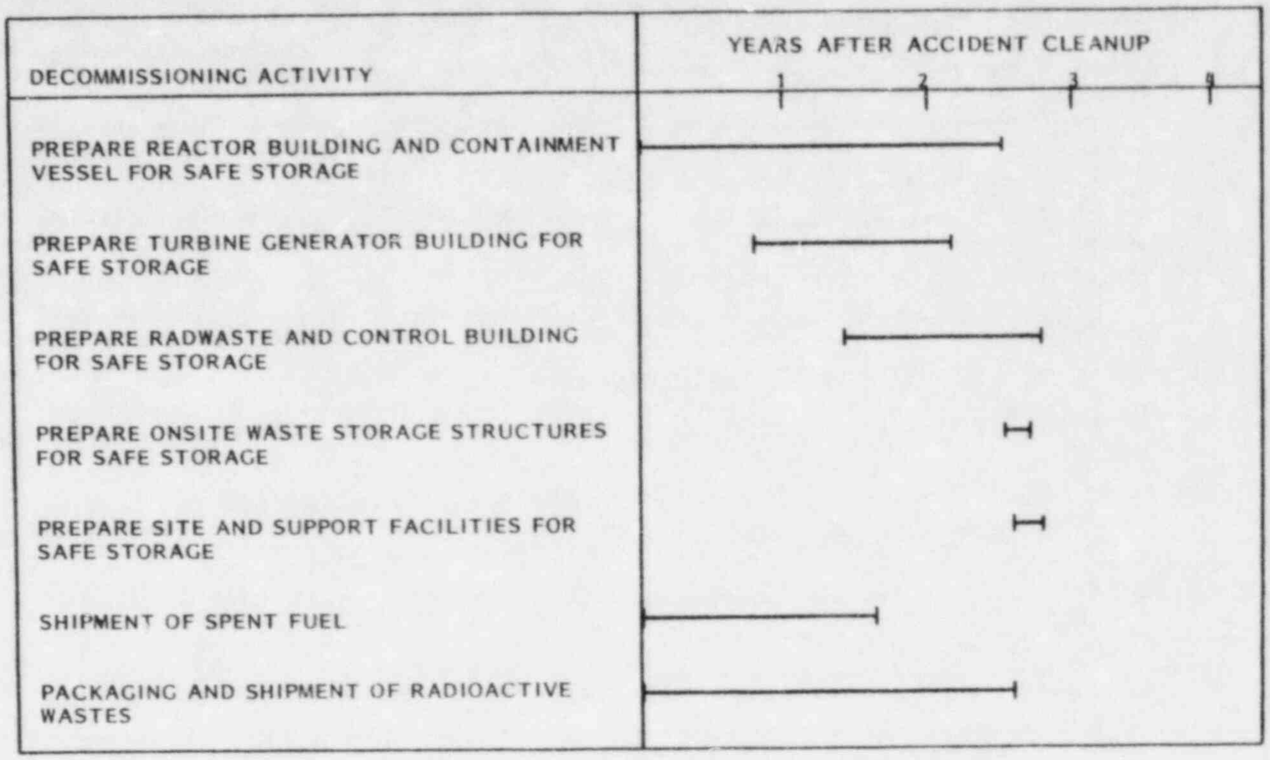


FIGURE 16.6-2. Overall Schedule and Sequence for Preparations for Safe Storage at the Reference BWR Following a Scenario 2 Accident

As is the case for DECON, decommissioning worker requirements for preparations for safe storage vary with accident severity because of estimated variations in the occupational radiation doses received by decommissioning workers and the individual radiation dose limitations. However, the variation in decommissioning worker requirements with accident severity is not as great for preparations for safe storage as it is for DECON, because for preparations for safe storage, the decommissioning workers spend a smaller fraction of their time inside the containment vessel where radiation dose rates are generally the highest. For preparations for safe storage, decommissioning worker manpower requirements are estimated to be about two-thirds as great following a scenario 1 accident as following a scenario 2 accident, and about 1.5 times as great following a scenario 2 accident. The estimated total staff labor requirement for preparations for safe storage following a scenario 1 accident is approximately 340 man-years, and for preparations for safe storage following a scenario 3 accident is approximately 560 man-years.

TABLE 16.6-2. Overall Staff Labor Requirements for Preparations for Safe Storage at the Reference BWR Following a Scenario 2 Accident

Position	Staff Labor Requirement (man-years) in Decommissioning Phase:(a) Planning and Preparation	Preparations for Safe Storage	Total Staff Labor Required (man-years)
Management and Support Staff			
Decommissioning Superintendent	1.5	3.1(b)	4.6
Secretary	3.0	8.7(b)	11.7
Clerk	1.0	5.6	6.6
Decommissioning Engineer	1.5	3.1(b)	4.6
Assistant Decommissioning Engineer	1.5	2.8	4.3
Radioactive Shipment Specialist	0	2.8	2.8
Procurement Specialist	0	2.8	2.8
Tool Crib Attendant	0	5.6	5.6
Reactor Operator(c)	0	22.4	22.4
Security Supervisor	0	2.8	2.8
Security Shift Supervisor	0	11.2	11.2
Security Patrolman	0	33.6	33.6
Contracts & Accounting Supervisor	0	3.1(b)	3.1
Health & Safety Supervisor	0	3.1(b)	3.1
Health Physicist	0	2.8	2.8
Protective Equipment Attendant	0	5.6	5.6
Industrial Safety Specialist	0	2.8	2.8
Quality Assurance Supervisor	0	3.1(b)	3.1
Quality Assurance Engineer	0	2.8	2.8
Quality Assurance Technician	0	11.2	11.2
Consultant (Safety Review)	0	1.4	1.4
Instrument Technician(d)	0	11.2	11.2
Maintenance Mechanic(d)	0	11.2	11.2
Warehouseman	0	5.6	5.6
Subtotals	8.5	168.4	176.9
Decommissioning Workers			
Shift Engineer	0	5.6	5.6
Crew Leader(e)	0	36.4	36.4
Utility Operator(e)	0	84.7	84.7
Laborer(e)	0	40.9	40.9
Craft Supervisor	0	11.2	11.2
Craftsman(e)	0	36.8	36.8
Senior Health Physics Technician	0	11.2	11.2
Health Physics Technician(e)	0	21.8	21.8
Subtotals	0	248.6	248.6
Totals	8.5	417.0	425.5

(a) Rounded to the nearest 0.1 man-year.
(b) Includes an additional 4 months following active decommissioning to complete the documentation and other unspecified license and contract termination requirements.
(c) Based on two operators per shift in the control room, three shifts per day, 7 days per week.
(d) Based on one per shift, three shifts per day, 7 days per week to maintain essential services.
(e) From Table K.5-12 of Appendix K.

16.6.3.2 Continuing Care and Deferred Decontamination

Continuing care commences immediately upon conclusion of preparations for safe storage and continues until deferred decontamination of the plant. Continuing care activities include security, surveillance, and maintenance functions. The level of effort required during continuing care at the reference BWR is assumed to be approximately the same following post-accident preparations for safe storage as it is following normal-shutdown preparations for safe storage. Continuing care requirements following normal-shutdown preparations for safe storage at the reference BWR are described in Section J.4 of Reference 1.

In this study, two potential continuing care periods are considered: 30 years and 100 years. The annual labor requirement is estimated to be less than 1.5 man-year/year. (See Section J.4.6 of Reference 1.) Thus, the total cumulative labor requirement for the 30-year or the 100-year safe storage period is conservatively estimated to be 45 man-years or 150 man-years, respectively.

The level of effort required to perform the work of deferred decontamination at the conclusion of the continuing care period is assumed to be about the same as that required for DECON, described in Section 16.6.2. A number of dismantlement tasks, such as the draining and decontamination of contaminated liquid systems and the removal of some radioactive materials, are accomplished during preparations for safe storage. During deferred decontamination, the time not expended on these tasks is offset by the time required to familiarize the work force with the facility, remove the locks and barriers installed to secure the plant, and restore essential services. Therefore, it is assumed that the basic decommissioning requirement for efficient performance of the decontamination tasks and the time required for deferred decontamination are the same as for DECON.

The actual decommissioning worker requirements for deferred decontamination are controlled by the limit on radiation dose to decommissioning workers. A method for estimating actual decommissioning worker requirements, based on the residual radioactivity remaining in the plant at the conclusion of the continuing care period, is described in Section K.5.3.5 of

Appendix K. The overall staff labor requirements for deferred decontamination following a scenario 2 accident, including the adjusted decommissioning worker requirements, are estimated to total about 680 man-years after 30-year safe storage and about 490 man-years after 100-year safe storage. Following a scenario 1 accident, overall staff labor requirements are estimated to total about 480 man-years after 30-year safe storage and about 380 man-years after 100-year safe storage. Following a scenario 3 accident, overall staff labor requirements for deferred decontamination are estimated to total about 1080 man-years after 30-year safe storage and about 690 man-years after 100-year safe storage.

16.6.4 ENTOMB Activities and Manpower Requirements

ENTOMB appears to be less acceptable following a reactor accident than following normal shutdown of the reactor because: 1) the residual radioactivity levels in the facility following an accident, even after substantial reactor cleanup activities, are significantly higher than following normal shutdown, and 2) the post-accident radionuclide inventory decays more slowly than the normal-shutdown inventory because of the large quantities of ^{137}Cs (with a 30-year half-life) released by the accident. Post-accident ENTOMB requires the continuation of the facility's nuclear license for a period of continuing care until the entombment structure is reopened and the materials inside are surveyed and either released for unrestricted use or packaged and shipped to a waste disposal site.

16.6.4.1 Entombment Activities and Requirements

Activities for post-accident ENTOMB at the reference BWR are similar to ENTOMB activities following normal shutdown, described in Appendix K of Reference 1. Entombment of radioactive materials is assumed to take place within the confines of the steel primary containment vessel and the surrounding concrete biological shield. Sufficient space is not available within the containment vessel for all of the radioactive materials in the plant; therefore, some radioactive waste must still be shipped to a shallow-land disposal site. After the material to be entombed is placed inside the containment vessel, all openings through the biological shield are filled with cast-in-place, reinforced concrete, and the removable concrete shield plugs are grouted in place. The reactor building is sealed and left in place to provide a secondary barrier for all-weather protection and enhanced security of the entombment structure.

All plant areas outside of the entombment barrier are decontaminated to allow unrestricted release if desired. Prior to the placement of other radioactive components and wastes inside the entombment structure, the reactor vessel internals containing long-lived activation products (e.g., ^{59}Ni, ^{94}Nb) are removed and shipped offsite to a nuclear waste repository. Any radioactive wastes not entombed are packaged and shipped offsite for disposal.

The shielded waste storage facility (i.e., the canyon and caisson facility) constructed onsite to house accident-cleanup wastes is postulated to be entombed rather than shipping the wastes offsite and decontaminating the facility. Entombment involves sealing the cover blocks in place and decontaminating the upper parts of the storage structure.

The schedule for ENTOMB at the reference BWR following a scenario 2 accident is shown in Figure 16.6-3. As with the other decommissioning alternatives, ENTOMB begins in the reactor building and the containment vessel and proceeds through the other buildings as staff are available and as the various systems in these other buildings complete their required service functions. ENTOMB following a scenario 2 accident and the subsequent accident cleanup is estimated to require about 4.4 years for completion. Variations in accident severity, within the range of accident scenarios considered in this study, are estimated to change the duration of ENTOMB by about $\pm$0.2 years.

Estimated utility staff labor requirements for ENTOMB at the reference BWR following the scenario 2 accident are shown in Table 16.6-3. The total staff labor requirement for ENTOMB following a scenario 2 accident is estimated to be about 880 man-years. The estimated staff labor requirement for ENTOMB following the scenario 1 accident is approximately 560 man-years, and for ENTOMB following the scenario 3 accident is approximately 1490 man-years.

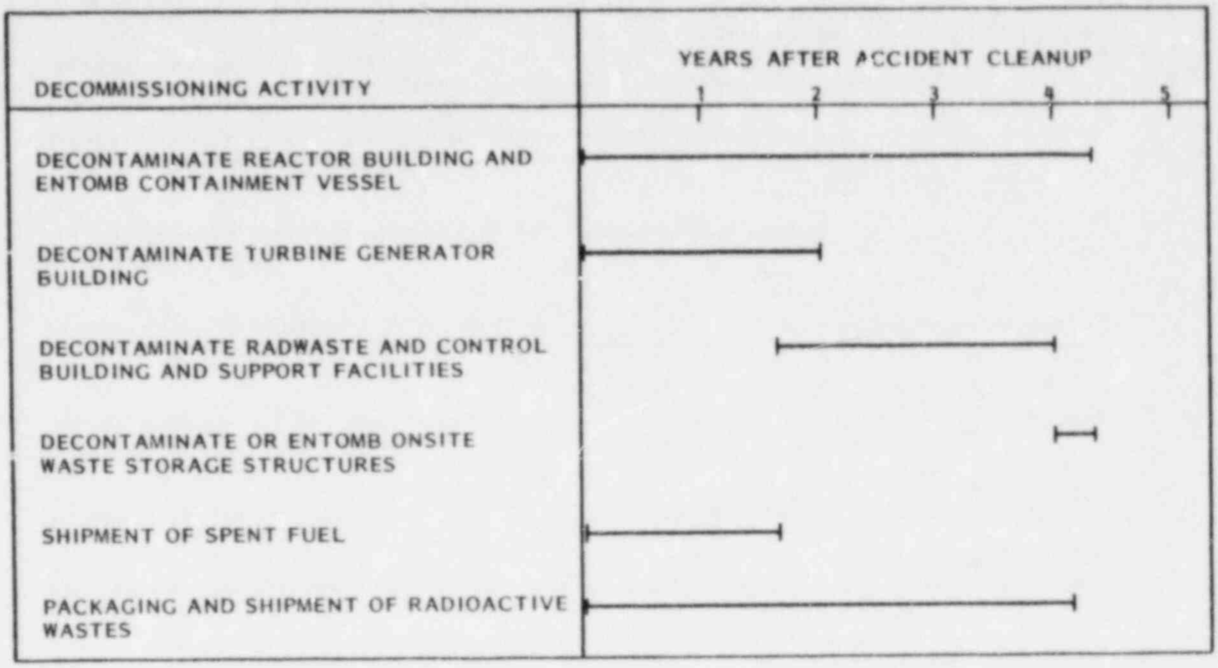


FIGURE 16.6-3. Overall Schedule and Sequence for ENTOMB at the Reference BWR Following a Scenario 2 Accident

16.6.4.2 Continuing Care and Possible Deferred Decontamination

The initial decommissioning activities for ENTOMB are followed by a period of continuing care that includes security, surveillance, and maintenance. The level of effort for the continuing care phase of ENTOMB is judged to be less than that for the continuing care phase of SAFSTOR because of the increased amount of preparation of the entombed facility. Continuing care costs for ENTOMB are estimated to be about half of those for SAFSTOR. Assuming that labor accounts for the same fraction of the total costs of continuing care for both ENTOMB and SAFSTOR, the annual continuing care labor requirement for ENTOMB is about 0.8 man-year/year. Thus, the total cumulative labor requirement for 100 years of continuing care for the ENTOMB alternative is estimated to be about 80 man-years.

Deferred decontamination, if required for the entombment alternative, is anticipated to require a greater level of effort than deferred decontamination for SAFSTOR. Although there is less radioactive material to remove from

TABLE 16.6-3. Overall Staff Labor Requirements for ENTOMB at the Reference BWR Following a Scenario 2 Accident

Position	Staff Labor Requirement (man-years) in Decommissioning Phase:(a) Planning and Preparation	ENTOMB	Total Staff Labor Required (man-years)
Management and Support Staff			
Decommissioning Superintendent	1.5	4.7(b)	6.2
Secretary	3.0	13.5(b)	16.5
Clerk	1.0	8.8	9.8
Decommissioning Engineer	1.5	4.7(b)	6.29
Assistant Decommissioning Engineer	1.5	4.4	5.9
Radioactive Shipment Specialist	0	4.4	4.4
Procurement Specialist	0	4.4	4.4
Tool Crib Attendant	0	8.8	8.8
Reactor Operator(c)	0	35.2	35.2
Security Supervisor	0	4.4	4.4
Security Shift Supervisor	0	17.6	17.6
Security Patrolman	0	52.8	52.8
Contracts & Accounting Supervisor	0	4.7(b)	4.7
Health & Safety Supervisor	0	4.7(b)	4.7
Health Physicist	0	4.4	4.4
Protective Equipment Attendant	0	8.8	8.8
Industrial Safety Specialist	0	4.4	4.4
Quality Assurance Supervisor	0	4.7(b)	4.7
Quality Assurance Engineer	0	4.4	4.4
Quality Assurance Technician	0	17.6	17.6
Consultant (Safety Review)	0	2.2	2.2
Instrument Technician(d)	0	17.6	17.6
Maintenance Mechanic(d)	0	17.6	17.6
Warehouseman	0	8.8	8.8
Subtotals	8.5	263.6	272.1
Decommissioning Workers			
Shift Engineer	0	8.8	8.8
Crew Leader(e)	0	70.7	70.7
Utility Operator(e)	0	175.2	175.2
Laborer(e)	0	109.3	109.3
Craft Supervisor	0	17.6	17.6
Craftsman(e)	0	129.7	129.7
Senior Health Physics Technician	0	17.6	17.6
Health Physics Technician(e)	0	74.7	74.7
Subtotals	0	603.6	603.6
Totals	8.5	867.2	875.7

(a) Rounded to the nearest 0.1 man-year.
(b) Includes an additional 4 months following active decommissioning to complete the documentation and other unspecified license and contract termination requirements.
(c) Based on two operators per shift in the control room, three shifts per day, 7 days per week.
(d) Based on one per shift, three shifts per day, 7 days per week to maintain essential services.
(e) From Table K.5-17 of Appendix K.

the plant (because of some offsite disposal during the initial phase of ENTOMB), the removal of this material is complicated by having to break into the entombment structure and by the more-or-less random placement of this material within the structure. The methods used for deferred decontamination following ENTOMB are similar to those for DECON, described in Section K.5.2 of Appendix K.

16.7 COSTS OF DECOMMISSIONING

The costs of decommissioning at the reference BWR following the postulated accidents and subsequent accident cleanup are summarized in this section. Costs are given in early-1981 dollars and include a 25% contingency. These costs, based on unit cost information given in Appendix I, are developed in detail in Section K.6 of Appendix K. This section is derived from Chapter 13, which should be referred to for the bases and assumptions behind the costs in this section.

Cost estimates are made for each of the three decommissioning alternatives: DECON, SAFSTOR, and ENTOMB. Detailed cost estimates are prepared only for decommissioning following a scenario 2 accident, with cost estimates for decommissioning following the other two scenarios obtained by adjustment of the scenario 2 decommissioning costs. Assumptions made to develop cost estimates for post-accident decommissioning at the reference PWR, given in Chapter 13, also apply to cost estimates for post-accident decommissioning at the reference BWR.

16.7.1 DECON Costs

The estimated costs of post-accident DECON at the reference BWR are shown in Table 16.7-1. The total cost of DECON following a scenario 2 accident is estimated to be about $86 million. Total costs of DECON following the scenario 1 and scenario 3 accidents are estimated to be about $67 million and $119 million, respectively.

The major cost item for post-accident decommissioning is staff labor. Labor costs account for from 43 to 66% of DECON costs, depending on accident scenario. Management and support staff costs are a function of the time requirement for the completion of DECON and are relatively constant with

TABLE 16.7-1. Summary of Estimated Costs of Post-Accident DECON at the Reference BWR

Cost Category	Following Scenario 1 Accident[a] Estimated Costs ($ millions)[c]	Following Scenario 1 Accident[a] Percent of Total	Following Scenario 2 Accident[b] Estimated Costs ($ millions)[c]	Following Scenario 2 Accident[b] Percent of Total	Following Scenario 3 Accident[a] Estimated Costs ($ millions)[c]	Following Scenario 3 Accident[a] Percent of Total
Staff Labor						
Management & Support Staff	10.1		10.6		10.9	
Decommissioning Workers	13.2		26.5		52.6	
Total Staff Labor Costs	23.3	43	37.1	54	63.5	66
Waste Management						
Neutron-Activated Materials	1.7		1.7		1.4	
Contaminated Materials	9.8		10.3		10.8	
Radioactive Wastes	2.2		2.3		2.4	
Total Waste Management Costs	13.7	25	14.4	21	14.6	15
Energy	8.9	17	9.3	13	9.7	10
Special Tools and Equipment	2.6	5	2.6	4	2.6	3
Miscellaneous Supplies	2.7	5	2.7	4	2.7	3
Specialty Contractors	0.4	1	0.4	1	0.4	1
Nuclear Insurance & License Fees	2.0	4	2.0	3	2.0	2
Subtotals	53.6	100	68.6	100	95.5	100
Contingency (25%)	13.4		17.1		23.9	
Totals, DECON Costs	67.0		85.7		119.4	

(a) No detailed analysis performed for DECON following scenario 1 or scenario 3 accidents; estimates shown are derived by difference from those for scenario 2.
(b) From Table K.6-1 of Appendix K.
(c) Individually rounded to the nearest $0.1 million; costs adjusted to early 1981.

accident severity since the DECON time requirement does not vary greatly for the accident scenarios considered in this study. Decommissioning worker labor costs increase significantly with accident severity, because of the increase in manpower required to maintain compliance with occupational radiation dose standards.

Other major DECON costs include waste management costs and energy costs. Details of these cost items are given in Section K.6.1 of Appendix K.

16.7.2 SAFSTOR Costs

The estimated costs of post-accident SAFSTOR at the reference BWR are shown in Table 16.7-2. Costs are included for the three phases of SAFSTOR:

- preparations for safe storage
- continuing care (i.e., the safe storage period)
- deferred decontamination.

Total SAFSTOR costs at the reference BWR following a scenario 2 accident are estimated to be about \$104 million with a 30-year safe storage period and about \$94 million with a 100-year safe storage period. All costs are given in constant 1981 dollars, with no escalation for inflationary effects included. Deferred decontamination accounts for the majority of SAFSTOR costs, with preparations for safe storage making up about one-third of the total. The greater cost of SAFSTOR with 30-year safe storage results from the greater cost of deferred decontamination for this option. Relatively high radiation levels still remain in the reactor building and the containment vessel after 30 years of safe storage. To maintain compliance with occupational dose limits, labor requirements and costs of deferred decontamination after 30 years of safe storage are significantly greater than they are for deferred decontamination after 100 years of safe storage.

Approximate costs of SAFSTOR following a scenario 1 accident total about \$85 million with 30-year safe storage and about \$78 million with 100-year safe storage. Approximate costs of SAFSTOR following a scenario 3 accident total about \$138 million with 30-year safe storage and about \$120 million with 100-year safe storage. The cost differences between SAFSTOR costs for the three accident scenarios result primarily from deferred decontamination costs, with a small impact from preparations for safe storage costs.

TABLE 16.7-2. Summary of Estimated Costs of Post-Accident SAFSTOR at the Reference BWR

Cost Category	Following Scenario 1 Accident[a] Estimated Costs ($ millions)[c]	Following Scenario 1 Accident[a] Percent of Total	Following Scenario 2 Accident[b] Estimated Costs ($ millions)[c]	Following Scenario 2 Accident[b] Percent of Total	Following Scenario 3 Accident Estimated Costs ($ millions)[c]	Following Scenario 3 Accident Percent of Total
Preparations for Safe Storage						
Staff Labor						
Management & Support Staff	6.2		6.4		6.6	
Decommissioning Workers	5.8		8.8		13.1	
Total Staff Labor Costs	12.0	54	15.2	58	19.7	64
Waste Management	1.7	8	1.8	7	1.9	6
Energy	5.2	23	5.4	21	5.6	18
Special Tools and Equipment	0.5	2	0.5	2	0.5	2
Miscellaneous Supplies	1.6	7	1.6	6	1.6	5
Specialty Contractors	0.2	1	0.2	1	0.2	1
Nuclear Insurance & License Fees	1.2	5	1.2	5	1.2	4
Subtotal	22.4	100	25.9	100	30.7	100
Contingency (25%)	5.6		6.4		7.7	
Totals, Preparations for Safe Storage Costs	28.0		32.3		38.4	
Annual Continuing Care Costs	0.10		0.10		0.10	
Deferred Decontamination Costs						
After 30-Year Safe Storage	54.3		69.1		96.7	
After 100-Year Safe Storage	40.2		51.2		71.6	
Totals, SAFSTOR Costs						
With 30-Year Safe Storage	85.3		104.4		138.1	
With 100-Year Safe Storage	78.2		93.5		120.0	

(a) No detailed analysis performed for SAFSTOR following scenario 1 or scenario 3 accidents; estimates shown are derived by difference from those of scenario 2.
(b) From Table K.6-6 of Appendix K.
(c) Individually rounded to the nearest $0.1 million, except annual continuing care costs rounded to the nearest $10,000; costs adjusted to early 1981.

Labor costs account for more than half of the total costs of SAFSTOR. Energy costs are another important SAFSTOR cost item. Waste management costs are of lesser importance for SAFSTOR than they are for DECON. Details of SAFSTOR costs are given in Section K.6.2 of Appendix K.

16.7.3 ENTOMB Costs

The estimated costs of post-accident ENTOMB at the reference BWR are shown in Table 16.7-3. The total estimated cost of ENTOMB following a scenario 2 accident is $67 million. Total ENTOMB costs following a scenario 1 and scenario 3 accident are estimated to be about $52 million and $93 million, respectively.

Continuing care of the entombed plant is estimated to cost $50,000 annually, independent of the accident that is postulated to have occurred.

Labor costs, which account from 47 to 69% of ENTOMB costs, are the major cost item for this decommissioning alternative. Waste management costs for ENTOMB are estimated to be about half the corresponding value for DECON. Details of ENTOMB costs are given in Section K.6.3 of Appendix K.

16.8 SAFETY IMPACTS OF POST-ACCIDENT CLEANUP AND DECOMMISSIONING

The safety impacts of post-accident cleanup and decommissioning at the reference BWR are summarized in this section. Safety impacts from accident cleanup and decommissioning include: 1) radiation doses to the public from routine or accidental atmospheric releases of radioactivity during accident cleanup and decommissioning, 2) radiation doses to and industrial accidents involving workers performing the cleanup and decommissioning tasks, and 3) radiation doses to and accidents involving the transportation workers and the public during shipment of radioactive materials from the site. The evaluation of safety impacts employs 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 the safety impacts is believed to be conservative. The discussion of PWR safety in Chapter 14 should be referenced for the bases and assumptions behind the calculated doses in this section.

TABLE 16.7-3. Summary of Estimated Costs of Post-Accident ENTOMB at the Reference BWR

Cost Category	Following Scenario 1 Accident Estimated Costs ($ millions)[c]	Following Scenario 1 Accident Percent of Total	Following Scenario 2 Accident[b] Estimated Costs ($ millions)[c]	Following Scenario 2 Accident[b] Percent of Total	Following Scenario 3 Accident Estimated Costs ($ millions)[c]	Following Scenario 3 Accident Percent of Total
Staff Labor						
Management & Support Staff	9.3		9.8		10.1	
Decommissioning Workers	10.2		20.7		41.0	
Total Staff Labor Costs	19.5	47	30.5	57	51.1	69
Waste Management						
Neutron-Activated Materials	1.5		1.5		1.2	
Contaminated Materials	3.2		3.6		4.0	
Radioactive Wastes	2.1		2.2		2.2	
Total Waste Management Costs	6.8	16	7.3	14	7.4	10
Energy	8.1	20	8.5	16	8.9	12
Special Tools & Equipment	2.6	6	2.6	5	2.6	3
Miscellaneous Supplies	2.5	6	2.5	5	2.5	3
Specialty Contractors	0.2	1	0.2	1	0.2	1
Nuclear Insurance & License Fees	1.8	4	1.8	3	1.8	2
Subtotals	41.5	100	53.4	100	74.5	100
Contingency (25%)	10.4		13.3		18.6	
Totals, ENTOMB Costs	51.9		66.7		93.1	
Annual Continuing Care Costs	0.05		0.05		0.05	

(a) No detailed analysis performed for ENTOMB following scenario 1 or scenario 3 accidents; estimates shown are derived by difference from those of scenario 2.
(b) From Table K.6-13 of Appendix K.
(c) Individually rounded to the nearest $0.1 million, except annual continuing care costs rounded to the nearest $10,000; costs adjusted to early 1981.

The safety impacts of accident cleanup are summarized in Section 16.8.1. The safety impacts of decommissioning that follows accident cleanup are summarized in Section 16.8.2. Safety assessment details are presented in Section K.7 of Appendix K. These safety assessments are based on information about activities and manpower requirements for accident cleanup and decommissioning of the reference BWR that are presented in Sections K.3 and K.5, respectively, of Appendix K.

16.8.1 Accident Cleanup Safety

Radiological and nonradiological safety impacts from routine activities and from potential industrial and transportation accidents during accident cleanup at the reference BWR are summarized in Table 16.8-1. Safety impacts are estimated for accident cleanup following each of the three reactor accident scenarios. The principal source of radiation dose to the public is the atmospheric release of radionuclides from the facility during routine

TABLE 16.8-1. Summary of Safety Analysis for Accident Cleanup at the Reference BWR

Type of Safety Concern	Source of Safety Concern	Units	Accident Scenario 1	Accident Scenario 2	Accident Scenario 3
Public Safety					
Radiation Dose	Accident Cleanup Activities(a)	man-rem	6	20	40
	Transportation	man-rem	3	5	11
Occupational Safety					
Serious Lost-Time Injuries	Accident Cleanup Activities(c)	total no.	0.54	1.0	2.3
	Transportation(d)	total no.	0.31	0.54	1.3
Fatalities	Accident Cleanup Activities(c)	total no.	0.0038	0.0072	0.016
	Transportation(d)	total no.	0.019	0.032	0.076
Radiation Dose	Accident Cleanup Activities(e)	man-rem	1 490	4 170	11 940
	Transportation(f)	man-rem	28	50	120

(a) 50-yr committed dose equivalent to the bone, for the total population within 80 km of the site, from atmospheric releases during normal cleanup activities. Assumed to be about the same as for accident cleanup at the reference PWR. Doses resulting from postulated accidents are not included.
(b) 50-yr committed dose equivalent to the total body, for the population along the transport route. From Table K.7-3 of Appendix K.
(c) From Table K.7-2.
(d) From Table K.7-4.
(e) From Table K.7-1.
(f) From Table K.7-3.

cleanup activities. Contamination control measures and HEPA filters in plant ventilation systems limit the dose by reducing the levels of radioactivity in the air leaving the plant. Radiation doses to the public from the offsite transportation of accident cleanup wastes to authorized disposal sites are estimated using the assumption that each shipment contains enough radioactive material to result in the maximum exposure rates allowed by Department of Transportation regulations.[6]

Potential lost-time injuries to workers are primarily due to accident cleanup activities, although about one-third of the injuries are estimated to result from transportation accidents. Because the risk of death from a transportation accident is much higher than the risk of death from an industrial accident, the potential for worker fatalities, which is estimated to be small, results almost entirely from transportation tasks. Essentially no worker fatalities are predicted to occur from accident cleanup activities. Occupational radiation doses during accident cleanup result almost entirely from routine onsite activities. The occupational dose from accident cleanup following a scenario 2 accident is estimated to be about a factor of 3 larger than the dose from cleanup following a scenario 1 accident and about a factor of 3 less than the dose from cleanup following a scenario 3 accident.

16.8.2 Decommissioning Safety

Radiological and nonradiological safety impacts from routine activities and from potential industrial and transportation accidents during decommissioning that follows accident cleanup are summarized in Table 16.8-2. Decommissioning safety impacts are calculated for the DECON, SAFSTOR, and ENTOMB alternatives following a scenario 2 reactor accident. No detailed analyses of the safety impacts of decommissioning following the scenario 1 or scenario 3 accidents are made in this study. However, the effects on public and occupational safety resulting from variations in accident severity are discussed in Section K.7 of Appendix K.

The principal source of radiation dose to the public during decommissioning is the transport of radioactive materials from the reactor station to authorized disposal facilities. The estimated dose to the public from onsite decommissioning activities is small.

TABLE 16.8-2. Summary of Safety Analysis for Decommissioning at the Reference BWR Following a Scenario 2 Accident

Type of Safety Concern	Source of Safety Concern	Unit	DECON	SAFSTOR With Deferred Decontamination After: 30 Years	100 Years	ENTOMB
Public Safety						
Radiation Dose	Decommissioning Activities(a)	man-rem	0.05	0.05	0.05	0.04
	Transportation(b)	man-rem	16	2(c)	2(c)	7
	Continuing Care	man-rem	--	neg.(d)	neg.	neg.
Occupational Safety						
Serious Lost-Time Injuries	Decommissioning Activities(e)	total no.	1.9	2.2	2.2	1.7
	Transportation(f)	total no.	1.9	1.9	1.9	0.87
	Continuing Care	total no.	--	neg.	neg.	neg.
Fatalities	Decommissioning Activities(e)	total no.	0.0094	0.012	0.012	0.0085
	Transportation(f)	total no.	0.11	0.11	0.11	0.051
	Continuing Care	total no.	--	neg.	neg.	neg.
Radiation Dose	Decommissioning Activities(g)	man-rem	3181	2417	1137	2531
	Transportation(h)	man-rem	170	24(i)	24(i)	78
	Continuing Care(j)	man-rem	--	65	120	neg.

(a) 50-year committed dose equivalent to the lung, for the total population within 80 km of the site, from atmospheric releases during normal decommissioning activities. Assumed to be about the same as for normal-shutdown decommissioning at the reference BWR. Doses resulting from postulated accidents are not included.
(b) 50-year committed dose equivalent to the total body, for the population along the transport route. From Table K.7-8 of Appendix K.
(c) Includes only preparations for safe storage.
(d) neg. = negligible. Impacts of continuing care expected to be negligible compared to those of decommissioning activities.
(e) From Table K.7-7.
(f) From Table K.7-9.
(g) From Tables K.7-5 and K.7-6.
(h) From Table K.7-8.
(i) Includes only preparations for safe storage.
(j) From Table K.7-6.

Approximately 4 lost-time injuries to workers are predicted to result from industrial-type accidents during decommissioning. Fatalities appear to be unlikely during decommissioning, with the greatest risk associated with transportation accidents. Occupational radiation doses during decommissioning result primarily from routine onsite activities. Approximately two-thirds of the occupational dose from decommissioning activities results from operations in the reactor building and the containment vessel. Occupational doses resulting from a scenario 2 accident are estimated to be increased by about a factor of 2 following a scenario 3 accident and to be reduced by a similar factor following a scenario 1 accident.

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CHAPTER 17

COMPARISONS OF POST-ACCIDENT DECOMMISSIONING WITH NORMAL-SHUTDOWN DECOMMISSIONING

In this chapter, manpower requirements, occupational radiation doses, waste volumes, and costs for post-accident decommissioning at the reference PWR and the reference BWR are compared with the same parameters for decommissioning following normal shutdown. The scenario 2 accident is used as the basis for the post-accident parameter values. The previous PWR[1] and BWR[2] decommissioning studies provide information about the safety and costs of decommissioning following normal shutdown. For ease of comparison, decommissioning costs shown in References 1 and 2 are adjusted to the early-1981 cost base used in this study. The factors used for adjusting costs from the original 1978 data base to the 1981 data base are described in Section I.7 of Appendix I (Volume 2).

For the post-accident case, decommissioning is preceded by accident cleanup. The accident cleanup activities remove most of the accident-generated waste; however, there are still greater levels and more widespread contamination present after accident cleanup than after normal shutdown.

For the post-accident case, certain tasks are completed during accident cleanup that would normally be part of the decommissioning activities (e.g., decontamination of the reactor coolant system and comprehensive radiation surveys). Significant portions of other tasks that are part of normal decommissioning are also performed during accident cleanup (e.g., removal and segmentation of reactor vessel internals, surface decontamination inside the containment structure, and disposal of fuel racks). The planning and preparation period that precedes the decommissioning can be shortened for the post-accident case because of preparations activities that take place prior to and during accident cleanup. Thus, accident cleanup can have the effect of reducing some of the normal decommissioning requirements and costs.

Accident cleanup also results in some new decommissioning tasks that are not included in normal-shutdown decommissioning. Systems and structures, such as the demineralizer system used to process accident water and facilities for the interim storage of radioactive wastes, that are installed specially for accident cleanup must be decommissioned. These new tasks increase the requirements and costs of post-accident decommissioning relative to normal-shutdown decommissioning.

Post-accident decommissioning and normal-shutdown decommissioning via the DECON alternative are compared in Section 17.1. Comparisons based on the SAFSTOR alternative are presented in Section 17.2. Comparisons based on the ENTOMB alternative are presented in Section 17.3.

17.1 COMPARISONS OF POST-ACCIDENT DECON WITH NORMAL-SHUTDOWN DECON

Comparisons of manpower requirements, occupational radiation doses, waste management requirements, and costs between post-accident decommissioning following a scenario 2 accident and normal-shutdown decommissioning via the DECON alternative are presented in Table 17.1-1. Estimated staff labor requirements and occupational radiation doses for post-accident DECON are approximately 2 to 3 times larger than estimated staff labor requirements and occupational radiation doses for DECON following normal shutdown. Volumes of radioactive waste requiring disposal are estimated to be about 5% larger for post-accident DECON than they are for DECON following normal shutdown. The total costs of post-accident DECON are estimated to be about 33% larger than the costs of DECON following normal shutdown.

Decontamination and dismantlement operations for post-accident DECON are very similar to those for DECON following normal shutdown. However, average radiation dose rates experienced by workers engaged in decommissioning operations inside the containment structures of the reference reactors are estimated to be 2 to 3 times higher for post-accident DECON following a scenario 2 accident than they are for normal-shutdown DECON. Consequently, the number of decommissioning workers required to ensure compliance with occupational dose limits is larger for post-accident DECON than for normal-shutdown DECON. Staff labor requirements shown in Table 17.1-1 include both

TABLE 17.1-1. Comparison of Post-Accident and Normal-Shutdown Decommissioning (DECON Alternative)

Parameter	Reference PWR		Reference BWR	
	Post-Accident DECON[a]	Normal-Shutdown DECON[b]	Post-Accident DECON[a]	Normal-Shutdown DECON[c]
Staff Labor (man-years)	790	300	1 070	610
Occupational Dose (man-rem)	3 260	1 200	3 350	1 960
Waste Volume (m^3)[d]	18 800	17 900	20 000	18 900
Cost ($ millions)[e,f]	68	51	86	64

(a) Values are for DECON following cleanup after a scenario 2 accident.
(b) Values are from Reference 1 (NUREG/CR-0130).
(c) Values are from Reference 2 (NUREG/CR-0672).
(d) Waste volumes do not include the volume of the final fuel core.
(e) Costs are in early-1981 dollars and include a 25% contingency.
(f) Costs do not include the cost of disposal of the final fuel core.

decommissioning workers and support staff. Decommissioning worker requirements are adjusted to ensure that occupational radiation doses to individual workers do not exceed 5 rem/year.[3]

Waste management requirements for post-accident DECON following a scenario 2 accident are only slightly larger than those for DECON following normal shutdown. The volumes of activated and contaminated material requiring disposal are comparable for the two cases. (Slightly more contaminated concrete is packaged for disposal during post-accident decommissioning than during decommissioning following normal shutdown.) The major difference is the greater volume of radioactive waste (e.g., disposable clothing, plastic covers, tools, construction materials, immobilized decontamination solutions, and miscellaneous trash) requiring disposal as a result of the additional decontamination required during post-accident decommissioning.

The major factor affecting the increase in costs for post-accident DECON is the larger labor requirement for post-accident decommissioning.

17.2 COMPARISONS OF POST-ACCIDENT SAFSTOR WITH NORMAL-SHUTDOWN SAFSTOR

Comparisons of manpower requirements, occupational radiation doses, waste management requirements, and costs between post-accident decommissioning following a scenario 2 accident and normal-shutdown decommissioning based on the SAFSTOR alternative are presented in Table 17.2-1. Comparisons are shown for preparations for safe storage, continuing care, and deferred decontamination at the end of the continuing care period. A continuing care period of 100 years is used as the basis for these comparisons. Similarities and differences between post-accident and normal-shutdown SAFSTOR are discussed in the following subsections.

17.2.1 Preparations for Safe Storage

The objective of preparations for safe storage is to place the reactor facility in a condition that satisfies the requirements for public safety during the period of storage. Preparations for safe storage encompasses those activities that accomplish this objective. The major activities include:

- removal of loose, readily removable contamination and fixation of residual surface contamination to the maximum extent possible by spray painting
- deactivation and securing of equipment and systems not needed during continuing care
- transfer of contaminated equipment from areas being decontaminated to other secured areas
- installation of equipment and systems needed for plant security and surveillance.

These activities are carried out in a manner that minimizes initial commitments of time, money, occupational radiation dose, and waste disposal. Since activities inside the containment building are minimal, differences in the requirements for post-accident and normal-shutdown preparations for safe storage are not great.

TABLE 17.2-1. Comparison of Post-Accident and Normal-Shutdown Decommissioning (SAFSTOR Alternative)

Parameter	Reference PWR Post-Accident SAFSTOR[a]	Reference PWR Normal-Shutdown SAFSTOR[b]	Reference BWR Post-Accident SAFSTOR[a]	Reference BWR Normal-Shutdown SAFSTOR[c]
Preparations for Safe Storage				
Staff Labor (man-years)	190	115	425	380
Occupational Dose (man-rem)	440	420	840	400
Waste Volume (m^3)[d]	430	390	1 250	1 050
Cost ($ millions)[e,f]	17	14	32	29
Continuing Care (100-Year Period)[g]				
Staff Labor (man-years)	200	200	150	135
Occupational Dose (man-rem)	225	14	120	10
Cost ($ millions)[e]	11	10	10	8
Deferred Decontamination				
Staff Labor (man-years)	320	300	490	420
Occupational Dose (man-rem)	300	1	320	1
Waste Volume (m^3)	18 600	1 390	19 500	620
Cost ($ millions)[e]	45	41	51	36

(a) Values are for SAFSTOR following cleanup after a scenario 2 accident.
(b) Values are from Reference 1 (NUREG/CR-0130).
(c) Values are from Reference 2 (NUREG/CR-0672).
(d) Waste volumes do not include the volume of the final fuel core.
(e) Costs are in early-1981 dollars and include a 25% contingency.
(f) Costs do not include the cost of disposal of the final fuel core.
(g) Values are total values for a 100-year continuing care period.

17.2.2 Continuing Care

Continuing care (i.e., the safe storage period of SAFSTOR) commences when preparations for safe storage are completed and continues until deferred decontamination of the facility. Activities carried out during the safe storage period include security, surveillance, and maintenance functions. The level of effort required for continuing care following an accident is judged to be approximately the same as that required following normal reactor shutdown. However, the occupational exposure rate to maintenance and surveillance personnel is higher for post-accident continuing care than it is for normal-shutdown continuing care. Reasons for the higher occupational exposure during post-accident continuing care are discussed in Section 17.2.3.

17.2.3 Deferred Decontamination

The decay of radioactive contamination within the stored facility is considerably slower following a postulated reactor accident than it is following normal reactor shutdown. For the post-accident radionuclide inventory, the controlling radionuclides are ^{137}Cs and ^{90}Sr with 30-year half-lives, whereas for normal shutdown the controlling radionuclide is ^{60}Co with a 5.27-year half-life. A continuing care period of 100 years is used as the basis for comparisons between post-accident and normal-shutdown SAFSTOR activities. After 100 years, the radioactivities of ^{137}Cs and ^{90}Sr decay by about a factor of 10, whereas the radioactivity of ^{60}Co decays by a factor of almost 1 million.

The basic operations performed during deferred decontamination are the same as those performed during DECON. Therefore, the level of effort to efficiently perform the work during deferred decontamination is approximately the same as it is for DECON. However, decommissioning worker requirements are controlled by radiation dose limitations. The worker requirement for deferred decontamination following an accident is greater than it is for deferred decontamination following normal shutdown until the radiation dose rates decay to levels permitting decommissioning activities to proceed without a requirement for additional labor to comply with radiation dose limitations (about 60 years for the accident scenarios of this study).

For post-accident deferred decontamination, radioactive decay does not significantly reduce the amount of contaminated material requiring packaging and disposal prior to unrestricted release of the facility. Waste management requirements for post-accident deferred decontamination are about the same as they are for post-accident DECON. For the normal shutdown case, the amount of contaminated material requiring disposal is substantially reduced as a result of radioactive decay during the continuing care period. Therefore, the waste management requirements for post-accident deferred decontamination are greater than those for normal-shutdown deferred decontamination.

17.3 COMPARISONS OF POST-ACCIDENT ENTOMB WITH NORMAL-SHUTDOWN ENTOMB

Comparisons of manpower requirements, occupational radiation doses, waste management requirements, and costs between post-accident ENTOMB following a scenario 2 accident and normal shutdown ENTOMB are presented in Table 17.3-1. Values shown in the table are for the entombment phase of ENTOMB.

TABLE 17.3-1. Comparison of Post-Accident and Normal-Shutdown Decommissioning (ENTOMB Alternative)[a]

	Reference PWR		Reference BWR	
Parameter	Post-Accident ENTOMB[b]	Normal-Shutdown ENTOMB[c]	Post-Accident ENTOMB[b]	Normal-Shutdown ENTOMB[d]
Staff Labor (man-years)	680	--[h]	880	630
Occupational Dose (man-rem)	2610	1020	2610	1760
Waste Volume (m^3)[e]	8150	--[h]	8350	8040
Cost ($ millions)[f,g]	52	36	67	58

(a) Values are for the entombment phase of ENTOMB.
(b) Values are for ENTOMB following cleanup after a scenario 2 accident.
(c) Values are from Reference 4 (NUREG/CR-0130 Addendum).
(d) Values are from Reference 2 (NUREG/CR-0672).
(e) Waste volumes do not include the volume of the final fuel core.
(f) Costs are in early-1981 dollars and include a 25% contingency.
(g) Costs do not include the cost of disposal of the final fuel core.
(h) Information not available from Reference 4.

In terms of the activities and requirements for decommissioning, ENTOMB at a nuclear power reactor is quite similar to DECON. Thus, as a first approximation, post-accident decommissioning by ENTOMB compares to normal-shutdown ENTOMB in the same way that post-accident decommissioning by DECON compares to normal-shutdown DECON.

A major difference between the DECON and ENTOMB alternatives is that the facility and site can be released for unrestricted use on completion of DECON, whereas ENTOMB is followed by a period of continuing care until the onsite radioactivity either decays to unrestricted release levels or is removed from the site via eventual deferred decontamination. In this respect, ENTOMB is much less attractive as a decommissioning alternative following a reactor accident than following normal shutdown because of: 1) the higher levels of the entombed radioactivity resulting from accident-generated contamination in the plant, and 2) the slower decay of the post-accident radionuclide inventory, which is controlled by ^{90}Sr and ^{137}Cs (with 30-year half-lives) rather than by ^{60}Co (with a 5.27-year half-life) as is the normal-shutdown radionuclide inventory. Therefore, selection of ENTOMB as the decommissioning alternative to be used following a reactor accident implies a commitment to a much longer period of retention of the entombed plant or to an eventual deferred decontamination involving significantly greater time and manpower commitments and expenditures than selection of ENTOMB following normal shutdown.

REFERENCES

1. R. I. Smith, G. J. Konzek, and W. E. Kennedy, Jr., Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station, NUREG/CR-0130, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, June 1978.

2. H. D. Oak, et al., Technology, Safety and Cost of Decommissioning a Reference Boiling Water Reactor Power Station, NUREG/CR-0672, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, June 1980.

3. U.S. Code of Federal Regulations, Title 10, Part 20.101, "Radiation Dose Standards for Individuals in Restricted Areas," (10 CFR 20.101), U.S. Government Printing Office, Washington, D.C., 1980.

4. R. I. Smith and L. M. Polentz, Technology, Safety and Cost of Decommissioning a Reference Pressurized Water Reactor Power Station, NUREG/CR-0130, Addendum, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, July 1979.

CHAPTER 18

FACILITATION OF POST-ACCIDENT CLEANUP AND DECOMMISSIONING

Methods and procedures are described in this study for the conceptual cleanup and decommissioning of reference LWRs that have been involved in serious accidents. The purpose of this chapter is to identify some design features, construction methods, and regulatory procedures that could expedite and simplify the cleanup and decommissioning operations at an accident-damaged reactor facility.

The primary purpose of the facilitation of accident cleanup and decommissioning is to reduce occupational and public radiation doses during these operations, while a secondary purpose is to reduce the costs. Ideally, if introduced during the design or construction stage, a facilitation procedure will also reduce radiation doses to workers and to the public during normal operation and routine maintenance periods. Reductions in cost will not occur in all cases where a design change is made to facilitate cleanup and decommissioning activities, and costs will often be increased rather than decreased.

Radiation doses, quantities of radioactive waste, and even the costs of accident cleanup and decommissioning can also be reduced by the careful planning of decontamination procedures and by the use of special tools and techniques. In the case of cleanup and decommissioning operations that must be performed in high-radiation areas, reductions in the performance time for these operations and consequent reductions in worker exposure can often be achieved by rehearsing an operation in a nonradiation area, sometimes with a mockup of the equipment or facility to be decontaminated, prior to entry into the radiation area for actual performance of the work.

Design considerations and innovative ideas that have been identified in a previous study(1) of reactor decommissioning following normal shutdown are also applicable to the facilitation of post-accident cleanup and decommissioning. Table 18.0-1, reproduced from Reference 1, lists some of these considerations and possible solutions.

TABLE 18.0-1. PWR Design Considerations and Innovative Ideas Related to Improvements in Decommissioning Technology[a]

Design Consideration	Possible Solution (Idea)[b]
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 advantagenus facility location.
4. Reduce personnel radiation exposure while changing radioactive 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) Reproduced from Reference 1.
(b) 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.

A facilitation report[2] prepared as part of this series of NRC-sponsored studies on the decommissioning of nuclear facilities deals specifically with design features, special equipment, and construction methods useful in the facilitation of decommissioning light water reactors. While the facilitation issues discussed in Reference 2 are intended to be directly applicable to the decommissioning of a nuclear reactor following normal shutdown, they also apply to the facilitation of post-accident cleanup and decommissioning. Facilitation methods discussed in Reference 2 include the following:

- improved documentation
- improved access to contaminated equipment
- substitution of materials in the pressure vessel internals
- design of the biological shield for easy removal
- techniques for improved protection of concrete and improved removal of contaminated concrete
- improved shielding of decommissioning and maintenance personnel
- techniques for reductions in radioactive waste volumes
- remote maintenance and decommissioning equipment
- primary coolant system decontamination
- special decommissioning tools and techniques.

Accident cleanup usually involves work in areas of relatively high radioactive contamination. To reduce worker exposure to radiation during the early stages of accident cleanup, design modifications should facilitate the use of remote decontamination techniques, such as spray systems for the washdown of contaminated surfaces, and of remotely controlled manipulators. Use should be made of protective coatings wherever possible (e.g., on concrete surfaces) to facilitate surface decontamination. Controls for containment building systems needed during accident cleanup should be located external to the building (e.g., in the auxiliary or fuel buildings or the reactor control room) whenever possible. Attention should be given to locating personnel entries and walkways and critical equipment away from areas within the containment building where contaminated water could accumulate as the result of an accident.

A major source of occupational dose identified in this study is the worker dose received during defueling operations. Specific attention should be given to the design of fuel cores and of fuel removal equipment to facilitate the removal of damaged fuel, thereby reducing the time spent by cleanup workers in the performance of this operation. Provision could be made in the design and construction of the reactor vessel for remote examination of the fuel core using fiber optics systems and miniature TV cameras.

Experience gained in the accident cleanup of TMI-2 (containment building decontamination is beginning as this report is being written in the summer of 1982) will undoubtedly provide valuable insights into design modifications and construction practices that could facilitate the cleanup and decommissioning of an accident-damaged nuclear reactor. Programs have been established in conjunction with TMI-2 cleanup activities to gather information about electrical and mechanical components and systems that were subjected to unusual stresses during the accident.[3] A goal of these programs is to recommend changes that would improve the reliability of the systems and services needed for accident cleanup.

An instrumentation and electrical program was established at TMI-2 in May 1980. The purpose of this program is to examine the status, determine failure modes and changes in operating characteristics, and analyze the impact on system safety of failures of electrical components and electrical equipment such as:

- area radiation monitor instrumentation
- polar crane electrical components
- cables and cable terminations
- neutron detectors
- thermocouples.

The program will recommend standards and qualifications for electrical equipment to improve the survivability and reliability of this equipment in case of an accident.

Several recommendations have already come from this program. These include:

- a recommendation that MOS transistors not be used in any application in a nuclear reactor
- a recommendation for the use of conformal coating of assembled printed wiring boards to improve their performance in humid environments
- a recommendation for improvements in the design, testing, and installation of electrical equipment.

Several simple circuit design changes have also been identified.

The mechanical component information and examination program at TMI-2 began in the summer of 1981. The objective of this program is to examine the condition and performance of selected TMI-2 mechanical components to identify improvements in safety and cost-effectiveness. Components to be evaluated include pumps, valves, seals, fans, ductwork, pipe and cable tray supports, electrical distribution panels, and concrete surfaces.

Management of the wastes from cleanup and decommissioning of an accident-damaged reactor represents a major cost item and a significant source of worker exposure to radiation. As discussed in Section 5.3.3 of Chapter 5, some of these wastes (such as ion exchange media and evaporator bottoms from the processing of contaminated liquids) may not be suitable for disposal at a shallow-land burial ground. At TMI-2, temporary shielded storage for these wastes has been provided onsite. Ultimate disposal will require that the wastes be handled a second time with resulting additional occupational radiation doses and additional costs. No regulatory framework has yet been developed to specifically address the treatment and disposal of wastes from accident cleanup and decommissioning of a nuclear reactor. Regulatory attention should be given to defining waste disposal criteria that will minimize the impacts of waste management on costs and occupational exposures for accident cleanup and decommissioning.

REFERENCES

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2. E. B. Moore, Jr., Facilitation of Decommissioning Light Water Reactors, NUREG/CR-0569, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, December 1979.

3. TMI-2 Seminar, December 4-5, 1981, San Francisco, California.

CHAPTER 19

GLOSSARY

Abbreviations, acronyms, symbols, terms, and definitions directly related to PWR accident cleanup and decommissioning are defined and explained in this chapter. The chapter is divided into two parts, with the first part containing abbreviations, acronyms, and symbols, and the second part containing terms and definitions (including those used in special context for this study). Common terms covered adequately in standard dictionaries and commonly used chemical symbols are not included.

19.1 ABBREVIATIONS, ACRONYMS, AND SYMBOLS

Abbreviations and Acronyms

AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable[a]
ANI	American Nuclear Insurers
ANSI	American National Standards Institute
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations[a]
Ci	Curie[a]
cpm	Counts Per Minute
CRDM	Control Rod Drive Mechanism
CVCS	Chemical Volume Control System
DF	Decontamination Factor[a]
DOE	Department of Energy
DOT	Department of Transportation
dpm (or d/m)	Disintegrations per Minute[a]
ECCS	Emergency Core Cooling System
EDTA	Ethylenediamine Tetraacetic Acid

(a) See Section 19.2 for additional information or explanation.

EEI	Edison Electric Institute
EFPY	Effective Full Power Year(s)[a]
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
FES	Final Environmental Statement
FSAR	Final Safety Analysis Report
GPU	General Public Utilities Corporation
GVW	Gross Vehicle Weight
HEPA	High Efficiency Particulate Air (filter)[a]
HP	Health Physicist[a]
HTO[a]	Tritiated Water
HVAC	Heating, Ventilation, and Air Conditioning
ICRP	International Commission on Radiological Protection
ISFSI	Independent Spent Fuel Storage Installation
LLD	Lower Limit of Detection
LLW	Low-Level Waste[a]
LOCA	Loss of Coolant Accident[a]
LSA	Low Specific Activity
LWR	Light Water Reactor
MAELU	Mutual Atomic Energy Liability Underwriters
MAERP	Mutual Atomic Energy Reinsurance Pool
mCi	Millicurie, see also Ci (Curie)
MeV	Million Electron Volts
MPC	Maximum Permissable Concentration
mR	Milliroentgen, see also R (Roentgen)
mrad	Millirad, see also rad
mrem	Millirem, see also rem
MWD/MTU	Thermal Megawatt Day per Metric Ton of Uranium
MWe	Megawatts, electric
MWt	Megawatts, thermal
NEIL	Nuclear Electric Insurance Limited
NEPA	National Environmental Policy Act

(a) See Section 19.2 for additional information or explanation.

NML	Nuclear Mutual Limited
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OPG	Oxalic-Peroxide-Gluconic (solution)
PEIS	Programmatic Environmental Impact Statement
PWR	Pressurized Water Reactor
PVC	Polyvinyl Chloride
QA	Quality Assurance
QC	Quality Control
R	Roentgen[a]
rad[a]	Radiation Absorbed Dose
RCS	Reactor Coolant System[a]
rem[a]	Roentgen Equivalent Man
RPV	Reactor Pressure Vessel
RRC	Reactor Water Recirculation (system)
RWCU	Reactor Water Cleanup (system)
SDS	Submerged Demineralizer System
TLD	Thermoluminescent Detector (Dosimeter)[a]
TMI-2	Three Mile Island Nuclear Station, Unit 2
TRU	Transuranic[a]
VES	Vinyl Ester Styrene[a]
WPPSS	Washington Public Power Supply System

Symbols

α	Alpha Radiation[a]
β	Beta Radiation[a]
γ	Gamma Radiation[a]
$\bar{\chi}/Q'$	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.

(a) See Section 19.2 for additional information or explanation.

19.2 GLOSSARY DEFINITIONS

Absorbed Dose:	The amount of energy imparted by ionizing radiation to a unit mass of irradiated material at the place of interest. Also known as dose or dosage, it is defined in terms of rems or rads.
Acceptable Residual Radioactive Contamination Levels:	Those levels of radioactive contamination remaining at a decommissioned facility or on its site that are acceptable to the NRC for termination of the facility operating license and unrestricted release of the site.
Accident Cleanup:	The decontamination, defueling, and waste disposal activities undertaken following an accident and prior to commencement of procedures required for either restart or decommissioning of the reactor. In this study, accident cleanup includes all activities that culminate in defueling the reactor and cleanup of the reactor coolant system.
Accident Water:	The radioactively contaminated water occurring at a nuclear reactor station specifically as the result of a reactor accident.
Activation:	The induction of radioactivity in material irradiated with nuclear particles, usually neutrons produced by a nuclear reactor.
Activity:	Sometimes used for the term "radioactivity." (See Radioactivity.)
ALARA:	An operating philosophy to maintain worker exposure to ionizing radiation As Low As is Reasonably Achievable.
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 (alpha emitters). 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.
Anticontamination Clothing:	Special clothing worn in a radioactively contaminated area to prevent personal contamination.

Background:	Radiation originating from sources other than the source of interest (i.e., the nuclear plant). Background radiation includes natural radiation (e.g., cosmic rays and radiation from naturally radioactive elements) as well as man-made radiation (e.g., fallout from atmospheric weapons testing).
Beta Decay:	Radioactive decay in which a beta particle is emitted. This transformation changes only the atomic number of the nucleus, raising or lowering Z by one for emission of a negative or positive beta particle, respectively.
Beta Particle:	An electron, of either positive or negative charge, emitted by an atomic nucleus in a nuclear transformation.
Boron:	A neutron-absorbing element used in some nuclear reactor systems to control the reactivity of the nuclear fuel.
Breakthrough Volume:	The processing volume passed through an ion-exchange system that necessitates the replacement or regeneration of the ion-exchange media.
Burial Ground:	An area specifically designated for shallow subsurface disposal of solid radioactive wastes to temporarily isolate the waste from man's environment.
Burnup, Specific:	The total energy released per unit mass of a nuclear fuel. It is commonly expressed in megawatt-days per ton.
Capacity Factor:	The ratio of the electricity actually produced by a nuclear power plant to the electricity that would be produced if the reactor operated continuously at design capacity.
Cask:	A tightly sealing, heavily shielded, reusable shipping container for radioactive materials.
Cask Liner:	A tightly sealing, disposable metal container used inside a cask for shipping radioactive materials.
Chelating Agent:	An organic compound used to complex certain metal ions to prevent them from precipitating in neutral or alkaline wash solutions. A chelating agent has two or more groups that attach to a single ion to form a stable (usually 5- or 6-member) ring.

Chemical Decontamination: The use of chemical solutions to dissolve or suspend radioactive contaminants to remove them from surfaces.

Cladding: Metal material enclosing the uranium fuel to form the exterior of a nuclear reactor fuel rod.

Cleanup: See Accident Cleanup.

Code of Federal Regulations (CFR): A codification of the general rules by the executive departments and agencies of the federal government.

Cold Leg: The section of reactor coolant system piping through which the coolant returns to the core.

Complexing Agent: A chemical that combines with some ion to form a stable compound that no longer behaves like the original ion. The usual result of the complexing is to increase the mobility of the complexed ion.

Contact Maintenance: "Hands-on" maintenance, or maintenance performed by direct contact of personnel with the equipment. Typically, most nonradioactive maintenance is contact maintenance.

Containment: The structure housing the nuclear reactor, specially constructed of reinforced concrete and designed to withstand internal pressure and external collisions. It is fitted with gas-tight seals designed to contain radioactivity within the structure and to permit release of radioactive materials only under controlled conditions.

Contamination: Undesired (e.g., radioactive or hazardous) material that is deposited on the surface of, or internally ingrained into, structures or equipment, or that is mixed with another material.

Contamination, Fixed: Radioactivity remaining on a surface after repeated decontamination attempts fail to significantly reduce the contamination level. Survey meter readings made on the surface generally indicate the level of fixed contamination.

Contamination, Removable: That fraction of the radioactive contamination present on a surface that can be transferred to a smear test paper by rubbing with moderate pressure.

Continuing Care Period: The surveillance and maintenance phase of SAFSTOR or ENTOMB, with the facility secured against intrusion.

Control Rods: An array of tubes, containing material that absorbs neutrons, which are inserted into the core of a nuclear reactor to control or halt nuclear fission.

Controlled Area: Any specific region of a nuclear facility into which personnel entry is regulated by a physical barrier or administrative procedure.

Core: The central portion of a nuclear reactor containing the fuel elements.

Count Rate: The measured rate of the detection of ionizing events using a specific radiation detection device.

Criticality: The condition in which an arrangement of fissionable material (e.g., nuclear fuel in a reactor core) undergoes nuclear fission at a self-sustaining rate.

Curie (Ci): A unit of radioactivity that equals 3.7×10^{10} nuclear transformations per second. Several fractions of the curie are in common usage.

- Millicurie, abbreviated mCi. One-thousandth of a curie (3.7×10^{7} d/s).
- Microcurie, abbreviated µCi. One-millionth of a curie (3.7×10^{4} d/s).
- Nanocurie, abbreviated nCi. One-billionth of a curie (37 d/s).

Decay, Radioactive: A spontaneous nuclear transformation in which charged particles and/or gamma radiation are emitted.

Decommissioning: The measures taken following a nuclear facility's operating life to remove the facility and its site safely from service and dispose of the radioactive residue. The level of any residual radioactivity remaining in the facility or on the site after decommissioning must be low enough to allow unrestricted use of the facility/site.

DECON: A decommissioning alternative that involves the immediate removal of all radioactive material to levels considered acceptable for unrestricted release of the property.

Decontamination: The removal of radioactivity from structures, equipment, or material by chemical and/or mechanical means.

Decontamination Agents:	Chemical or cleansing materials used to effect decontamination.
Decontamination Factor (DF):	The ratio of the initial amount (i.e., concentration or quantity) of an undesired material to the final amount resulting from a treatment process.
Deferred Decontamination:	Those actions required after the continuing care period of SAFSTOR to disassemble and remove sufficient radioactive or contaminated materials from the facility and site to permit unrestricted release of the property.
Defueling:	The removal of the nuclear fuel from the reactor vessel of a nuclear power station.
Demineralizer Systems:	Processing systems in which ion-exchange materials are used to remove impurities from water.
Deminimus 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 design and safety analysis.
Desorb:	To remove materials that have been adsorbed on another material.
Disintegration, Nuclear:	The spontaneous (radioactive) transformation of an atom of one element to that of another, characterized by a definite half-life and the emission of particles or radiation from the nucleus of the first element.
Disintegration Rate:	The rate at which disintegrations (i.e., nuclear transformations) occur, in events per unit time (e.g., disintegrations per minute, dpm).
Disposal:	The disposition of materials with the intent that they will not enter man's environment in sufficient amounts to cause a significant 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).

Dose Commitment:	The integrated dose that results unavoidably from an intake of radioactive material, starting at the time of intake and continuing (at a decreasing dose rate) to a specified later time.
Dose, Equivalent:	Expresses the amount of ionizing radiation that is effective in the human body, in units of rems. Modifying factors associated with human tissue and body are taken into account. Equivalent dose is the product of absorbed dose, a quality factor, and a distribution factor. Referred to as Dose in this study.
Dose, Occupational:	An individual's exposure to ionizing radiation (above background) as a result of his employment, expressed in rems.
Dose Rate:	The radiation dose delivered per unit time, expressed in units of rems per hour.
Dosimeter:	A device, such as a film badge or an ionization chamber, that measures radiation dose.
Effective Full Power Year (EFPY):	The product of the fraction of rated power achieved times the number of years of operation. (Operation for 40 years at 75% of rated power is equivalent to 30 EFPY of operation.)
ENTOMB:	A decommissioning alternative that involves the encasement and maintenance of radioactive material or property in a strong and structurally long-lived material (e.g., concrete) to assure retention until the radioactivity decays to a level considered acceptable for unrestricted release.
Evaporator Bottoms:	The residue left in an evaporator after the liquids have been vaporized and removed.
Exclusive-Use Trucks:	Trucks used only to transport radioactive materials from a single shipper.
Exposure:	A measure of the ionization produced in air by x-ray 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 air in the volume element. The special unit of exposure is the roentgen. (See Roentgen.)

Facility:	The physical complex of buildings and equipment on a plant site.
Fission:	The splitting of a heavy atomic nucleus into two or more nearly equal 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 absorption of gamma rays, neutrons, or other particles.
Fission Products:	The lighter atomic nuclides (fission fragments) formed by the fission of heavy atoms. It also refers to the nuclides formed by the fission fragments' radioactive decay.
Fuel Assembly:	A bundle of fuel rods housed in a fixed geometry in a metal channel. During operation, water circulated through the assembly is heated by the nuclear reaction to produce steam.
Fuel Rod:	One of many metal tubes containing uranium fuel for a nuclear reactor.
Gamma Radiation:	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 uranium. The rays are similar to x-rays, but are nuclear in origin, i.e., they originate from within the nucleus of the atom.
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 it has absorbed.
Half-Life, Effective:	The time required for radioactivity contained in a biological system (such as a man or animal) to be reduced 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 form. Each radionuclide has a unique half-life. Measured half-lives vary from millionths of a second to billions of years.

Halogen: Any of a group of five chemically related nonmetallic elements that includes fluorine, chlorine, bromine, iodine, and astatine.

Health Physicist: A person trained to perform radiation surveys, oversee radiation monitoring, estimate the degree of radiation hazard, and advise on operating procedures for minimizing radiation exposures.

High Efficiency Particulate Air (HEPA) Filter: An air filter capable of removing at least 99.9% of the particulate material in an air stream.

High-Specific-Activity Waste: Waste having higher radioactivity than wastes which are routinely generated at nuclear power plants and disposed of by routine shallow-land burial techniques.

Hot Leg: The section of reactor coolant system piping through which the heated coolant flows away from the reactor core to provide heat for the generation of electricity.

Hot Spot: An area of radioactive contamination of higher than average concentration.

HTO: Chemical symbol for a molecule of water in which one of the ordinary hydrogen atoms has been replaced by an atom of tritium (tritiated water).

Immobilization: Treatment and/or emplacement of materials (e.g., radioactive contamination) so as to impede their movement. Usually refers to the fixing or solidification of radioactive wastes by any of several possible means.

Inhibitor: A chemical added to an acid wash solution to inhibit the corrosive reaction. Inhibitors are usually organic polar compounds having a carbon chain or ring with hydrogen atoms attached, and a polar group such as amino (NH_2^-), sulfonic (SO_3^-), or carboxy (CO_2^-).

Ion Exchange: A chemical process involving the selective adsorption or desorption of certain chemical ions in a solution onto a chemical compound or solid material.

Isotope: Any of two or more forms of an element having the same or very closely related chemical properties but different radioactive properties. Isotopes of an element have the same atomic number but different atomic weights.

License, Nuclear: Written authorization issued to the licensee by the appropriate regulatory body (i.e., the NRC for nuclear power reactors) to perform specific activities related to the possession and use of byproduct, source, or special nuclear material.

Licensed Material: Byproduct material, source material, or special nuclear material received, possessed, used, or transferred under a license issued by the NRC or a state regulatory agency.

Licensee: The holder of a license to perform specific activities related to the possession and use of byproduct, source, or special nuclear material.

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 1 year to millions of years.

Loss of Coolant Accident (LOCA): An accident at a nuclear reactor involving the loss of coolant from the primary reactor coolant system, which may result in overheating of the reactor core and the subsequent spread of gross radioactive contamination within the reactor containment.

Low-Activity Waste: Radioactive waste that is similar to wastes routinely generated at nuclear power plants and disposed of by routine shallow-land burial techniques.

Low-Level Waste (LLW): Wastes containing low but not hazardous quantities of radionuclides and requiring little or no biological shielding; low-level wastes generally contain no more than 10 nanocuries of transuranic material per gram of waste.

Man-rem: Used as a unit measure of population radiation dose, 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 implied.

Maximum-Exposed Individual: The hypothetical member of the public who receives the maximum radiation dose to an organ of reference. For the common case where exposures from airborne radionuclides result in the highest radiation exposure, this individual resides at the location of the highest airborne radionuclide concentration and eats food grown at that location.

Neutron Activation: See Activation.

Noble Gases: Any of a family of gases that do not readily react chemically with other elements. The noble gases include helium, neon, krypton, xenon, and radon.

Nuclear Reaction: A reaction involving a change in an atomic nucleus, such as fission, fusion, particle capture, or radioactive decay.

Nuclear Steam Supply System: A contractual term designating those components of the nuclear power plant furnished by the nuclear steam supply system supplier. Generally includes those systems most closely associated with the reactor vessel designed to contain or be in contact with the water coming from or going to the reactor core.

Overpack: Secondary (or additional) external containment or cushioning for packaged nuclear waste that exceeds certain limits imposed by regulation.

Plateout: The deposition of a substance from a suspension or solution onto the internal surfaces of the vessels (e.g., pipes) containing the fluid. In this report, plateout refers specifically to thin layers of radionuclides deposited on all exposed building and equipment surfaces inside the reactor building and on the insides of pipes and tanks during and after an accident.

Post-Accident Cleanup: See Accident Cleanup.

Primary System: See Reactor Coolant System.

Primary Water: Water in (or from) the reactor coolant system.

Process Solids: Wet solids in the forms of sludge, high-solids-content slurries, or granular materials generated during an accident or during subsequent treatment of accident water or decontamination liquids.

Purge: To remove undesirable materials. In this study, purging is the venting of radioactive gases from the reactor containment.

Rad: The unit of absorbed dose. The energy imparted by ionizing radiation to a unit mass of irradiated material at the place of interest. One rad equals 0.01 joules/kilogram.

Radiation: 1) The emission and propagation of radiant energy: for instance, the emission and propagation of electromagnetic waves or protons. 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 a dose in excess of 5 millirem in any 1 hour, or a dose in excess of 100 millirem in any 5 consecutive days. (See 10 CFR 20.202.)

Radiation Survey: An evaluation of radiation and associated 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 protective survey.

Radioactive Material: Any material or combination of materials that spontaneously emits ionizing radiation and has a specific activity in excess of 0.002 microcuries per gram of material. [See 49 CFR 173.389(e).]

Radioactivity: The property of certain nuclides of spontaneously transforming to other nuclides by emitting particles and/or gamma radiation. Also used to describe the number of nuclear transformations occurring in a given quantity of material per unit time. Often shortened to "activity."

Radiological Protection: Protection against the effects of internal and external exposure to ionizing radiation and radioactive materials.

Reactor Building Sump: The lowest part of the reactor building, designed to receive and hold, on a temporary basis, drainage and overflow.

Reactor Coolant System: This system, also known as the primary system, consists of the closed loop of components that routinely come in direct contact with the reactor coolant water that recirculates through the reactor.

Regulatory Guides: Documents that describe and make publicly available methods acceptable to the NRC staff for implementing specific parts of the 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 may be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the NRC. (Government agencies other than the NRC have regulatory guides pertaining to non-nuclear matters.)

Release Fraction: In this study, the fraction of the total fuel core inventory of a particular fission product released from the fuel as a result of cladding failure and/or fuel melting.

Rem: A unit of radiation 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 by a shielding wall from the item being maintained. Used in the nuclear industry to reduce the occupational radiation doses to maintenance personnel.

Residual Radioactivity Levels: The amount of radioactively contaminated material remaining in a nuclear facility after decommissioning has been completed and the facility license terminated. To be acceptable, this level must be low enough to permit the facility to be released for unrestricted use.

Restricted Area: Any area to which access is controlled for protection of individuals from exposure to ionizing radiation and radioactive materials.

Roentgen (R): The 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.

Roughing Filter: A prefilter with high efficiency for large particles and fibers but low efficiency for small particles. Usually used to protect a subsequent HEPA filter from high dust concentration.

SAFSTOR: A decommissioning alternative that involves fixing and maintaining property so that the risk to safety is acceptable for the period of storage, followed by decontamination and/or decay to an unrestricted level.

Shield: A body of material used to reduce the passage of ionizing 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). A shield may be required to protect personnel or to reduce radiation enough to allow use of counting instruments.

Short-Lived Radionuclides: For this study, those radioactive isotopes with half-lives less than about 10 years.

Shutdown: The time during which a facility is not in productive operation.

Site: The geographic area upon which the facility is located, subject to controlled public access by the facility licensee (includes the restricted area as designated in the NRC license).

Sludge: A mixture of fine solid materials which includes particles of cement dust, dirt, resin beads, etc., that have settled out from a suspension in the water.

Solid Radioactive Waste: 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.

Sump Water: Water that collects in the lowest part of the reactor building. In this study, it refers to such water that occurs specifically as the result of a reactor accident.

Surface Contamination: The deposition and attachment of radioactive materials to a surface. Also, the resulting deposits.

Surveillance:	Those activities necessary to ensure that the site remains in a safe condition (includes periodic inspection and monitoring of the site, maintenance of barriers preventing access to radioactive materials remaining on the site, and prevention of activities that might impair these barriers).
Technical Specifications:	Requirements and limits encompassing environmental and nuclear safety that 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 into the operating and/or possession-only license issued by the NRC.
Thermoluminescent Detector:	A solid-state device used to measure radiation doses. See Dosimeter.
Transuranic (TRU) Elements:	Elements with an atomic number greater than that of uranium (i.e., 92).
Transuranic (TRU) Waste:	Waste that contains or is contaminated by greater than 10 nanocuries per gram of transuranic elements. TRU wastes require special handling and storage considerations.
Tritiated Water:	See HTO.
Tritium:	A heavy radioactive isotope of hydrogen having mass number 3. It decays by emitting a low-energy beta particle.
Unrestricted Release:	Release of property from regulatory control such that subsequent use is no longer restricted in any way.
Vinyl Ester Styrene (VES):	An organic polymer used to solidify radioactive wastes. After the vinyl ester is mixed homogeneously with the waste, a promoter and catalyst are added to solidify the mixture by polymerization and to trap all free liquids.
Waste Management:	The planning and execution of essential functions relating to radioactive wastes, including treatment, packaging, interim storage, transportation, and disposal.
Waste, Radioactive:	Equipment and materials (from nuclear operations) that are radioactive and have no further use. Also called radwaste.

Zeolites:	Any of various natural or synthetic silicates used to purify water.
Zircaloy:	A zirconium-base alloy used as the cladding for fuel rods and for other reactor core hardware.

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This report presents a conceptual study of post accident decommissioning of light-water-reactors and the accident cleanup that precedes the decommissioning. The study provides information on the available technology, safety considerations, and probable costs of decommissioning, and the accident cleanup, of a reference PWR and BWR following a postulated accident. Three postulated accident scenarios are used in the report to illustrate a range of technological requirements, costs (in 1981 dollars), occupational radiation doses, potential radiation dose to the public, andother safety impacts. The decommissioning alternatives considered are DECON (immediate decontamination), SAFSTOR (safe storage followed by deferred decontamination), and ENTOMB (entombment). The study evaluates the sensitivity of the costs of accident cleanup to various factors which can influence them.

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