NUREG/CR-2601
Vol. 2 of 2

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

Appendices

Prepared by E. S. Murphy, G. M. Holter

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory
Commission

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission,
   Washington, DC 20555
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations,* and *Nuclear Regulatory Commission Issuances.*

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $21.00

NUREG/CR-2601
Vol. 2 of 2

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

Appendices

Manuscript Completed: October 1982
Date Published: November 1982

Prepared by
E. S. Murphy, G. M. Holter

Pacific Northwest Laboratory
Richland, WA 99352

Prepared for
Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN No. B2117

STUDY CONTRIBUTORS

STUDY LEADER

E. S. Murphy

STUDY CONTRIBUTORS

H. K. Elder

G. M. Holter

W. E. Kennedy, Jr.

R. I. Smith

EDITOR

D. R. Payson

ACKNOWLEDGMENT

The study leader is pleased to acknowledge the assistance of Frank P. Cardile of the Nuclear Regulatory Commission for his comments and guidance throughout the project.

REPORT CONTENTS OUTLINE

VOLUME 1 (CHAPTERS)

1 - INTRODUCTION

2 - SUMMARY

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

4 - STUDY APPROACH AND KEY STUDY BASES

5 - REGULATORY CONSIDERATIONS FOR POST-ACCIDENT CLEANUP AND FOR DECOMMISSIONING

6 - FINANCING FOR POST-ACCIDENT CLEANUP AND FOR DECOMMISSIONING

7 - CHARACTERISTICS OF THE REFERENCE PWR POWER STATION

8 - REFERENCE ACCIDENT SCENARIOS AND RESULTANT CONTAMINATION LEVELS AT A REFERENCE PWR

9 - ACCEPTABLE RESIDUAL RADIOACTIVE CONTAMINATION LEVELS FOR A DECOMMISSIONED LWR

10 - ACTIVITIES AND MANPOWER REQUIREMENTS FOR ACCIDENT CLEANUP AT A REFERENCE PWR

11 - COSTS OF ACCIDENT CLEANUP AT A REFERENCE PWR

12 - ACTIVITIES AND MANPOWER REQUIREMENTS FOR DECOMMISSIONING AT A REFERENCE PWR

13 - COSTS OF DECOMMISSIONING AT A REFERENCE PWR

14 - SAFETY IMPACTS OF POST-ACCIDENT CLEANUP AND DECOMMISSIONING AT A REFERENCE PWR

15 - IMPACTS OF TEMPORARY INABILITY TO DISPOSE OF WASTES OFFSITE

16 - POST-ACCIDENT CLEANUP AND DECOMMISSIONING AT A REFERENCE BWR

17 - COMPARISON OF POST-ACCIDENT DECOMMISSIONING WITH NORMAL-SHUTDOWN DECOMMISSIONING

18 - FACILITATION OF POST-ACCIDENT CLEANUP AND DECOMMISSIONING

19 - GLOSSARY

VOLUME 2 (APPENDICES)

A - REFERENCE SITE DESCRIPTION

B - REFERENCE PWR FACILITY DESCRIPTION

C - DETAILS OF REFERENCE ACCIDENT SCENARIOS AND RESULTANT CONTAMINATION LEVELS

D - GENERIC CLEANUP AND DECOMMISSIONING INFORMATION

E - DETAILS OF ACTIVITIES AND MANPOWER REQUIREMENTS FOR ACCIDENT CLEANUP AT A REFERENCE PWR

F - DETAILS OF COSTS OF ACCIDENT CLEANUP AT A REFERENCE PWR

G - DETAILS OF ACTIVITIES AND MANPOWER REQUIREMENTS FOR DECOMMISSIONING AT A REFERENCE PWR

H - DETAILS OF COSTS OF DECOMMISSIONING AT A REFERENCE PWR

I - COST ESTIMATING BASES

J - SAFETY ASSESSMENT DETAILS

K - DETAILS OF POST-ACCIDENT CLEANUP AND DECOMMISSIONING AT A REFERENCE BWR

CONTENTS

STUDY CONTRIBUTORS iii

REPORT CONTENTS OUTLINE v

APPENDIX A - REFERENCE SITE DESCRIPTION A-1

A.1 SITE LOCATION AND SIZE A-1

A.2 DEMOGRAPHY A-1

A.3 LAND USE A-2

A.4 METEOROLOGY A-3

REFERENCES A-7

APPENDIX B - REFERENCE PWR FACILITY DESCRIPTION B-1

B.1 CONTAINMENT BUILDING B-3

B.1.1 Reactor Vessel and Internals B-5

B.1.2 Reactor Coolant System B-10

B.1.3 Engineered Safety Systems B-15

B.1.4 Reactor Defueling Equipment and Procedures . . . B-17

B.1.4.1 Defueling Equipment B-17

B.1.4.2 Defueling Procedures B-18

B.1.5 Radionuclide Containment Systems B-20

B.1.6 Containment Building Major Equipment B-23

B.2 AUXILIARY AND FUEL BUILDINGS B-23

B.2.1 The Auxiliary Building B-23

B.2.2 Fuel Building B-25

B.2.3 Chemical Volume Control System B-29

B.2.4 Liquid Radioactive Waste Systems B-36

B.2.4.1 Dirty Radioactive Waste System B-36

B.2.4.2 Clean Radioactive Waste System B-37
B.2.5 Spent Fuel Handling and Storage B-39
B.2.6 Heating, Ventilating, and Air Conditioning Systems . B-40
B.2.7 Auxiliary and Fuel Buildings Major Equipment . . B-41
B.3 CONTROL BUILDING B-43
REFERENCES B-46
APPENDIX C - DETAILS OF REFERENCE PWR ACCIDENT SCENARIOS AND RESULTANT CONTAMINATION LEVELS C-1
C.1 DETAILS OF FISSION PRODUCT RADIONUCLIDE RELEASES FOR THE REFERENCE ACCIDENTS C-1
C.2 RADIOACTIVE DECAY CHARACTERISTICS OF SELECTED FISSION PRODUCT RADIONUCLIDES C-11
C.3 DETAILS OF EXPOSURE RATE CALCULATIONS C-15
C.3.1 Calculations of Exposure Rates from Plateout on Building and Equipment Surfaces C-16
C.3.2 Calculations of Exposure Rates from Contaminated Sump Water C-18
C.4 ROOM MODEL FOR ESTIMATING RADIATION EXPOSURE RATES . . C-21
C.5 SUMMARY OF TMI-2 CONTAMINATION DATA C-23
REFERENCES C-28
APPENDIX D - GENERIC CLEANUP AND DECOMMISSIONING INFORMATION . . D-1
D.1 DECONTAMINATION OF STRUCTURES AND EQUIPMENT D-1
D.1.1 Surface Film Dissolution D-1
D.1.2 Physical Cleanup of Surfaces and Equipment . . . D-7
D.1.3 Removal of Structural Material D-14
D.2 CONTAMINATION CONTROL D-19
D.2.1 Isolation of Contaminated Areas D-20
D.2.2 Local Mitigation of Contamination Sources . . . D-21
D.2.3 Collection of Contamination D-22

D.2.3.1 Vacuum Collection D-22

D.2.3.2 Portable Ventilation Systems D-23

D.3 SPECIAL TOOL AND EQUIPMENT REQUIREMENTS D-25

D.4 LIQUID WASTE TREATMENT D-26

D.4.1 Liquid Waste Sources D-28

D.4.2 Liquid Waste Treatment Alternatives D-30

D.4.2.1 Interim Onsite Storage D-30

D.4.2.2 Direct Immobilization D-31

D.4.2.3 Waste Processing D-31

D.4.3 Disposal of Processed Water D-33

D.5 PACKAGING, TRANSPORTATION, AND DISPOSAL OF WASTES D-35

D.5.1 Waste Characterization D-35

D.5.2 Alternatives for Waste Management D-40

D.5.3 Packaging of Radioactive Wastes D-45

D.5.4 Shipment of Radioactive Wastes D-46

D.6 ESSENTIAL SYSTEMS AND SERVICES D-48

D.7 QUALITY ASSURANCE D-50

D.8 ENVIRONMENTAL SURVEILLANCE D-51

REFERENCES D-55

APPENDIX E - DETAILS OF ACTIVITIES AND MANPOWER REQUIREMENTS FOR ACCIDENT CLEANUP AT A REFERENCE PWR E-1

E.1 RATIONALE FOR ACCIDENT CLEANUP E-4

E.2 DETAILS OF PREPARATIONS FOR ACCIDENT CLEANUP E-8

E.2.1 Planning and Preparation Activities E-9

E.2.1.1 Containment Entry and Data Acquisition . . E-9

E.2.1.2 Venting of Radioactive Gases E-10

E.2.1.3 Documentation for Regulatory Agencies . . E-13

E.2.1.4 Design, Fabrication, and Installation of Special Equipment E-16

E.2.1.5 Development of Work Plans and Procedures . . E-20

E.2.1.6 Selection and Training of Accident Cleanup Staff E-20

E.2.1.7 Removal of Accumulated Spent Fuel . . . E-21

E.2.2 Time Requirements for Preparations for Accident Cleanup E-22

E.2.3 Occupational Doses for Preparations for Accident Cleanup E-23

E.2.4 Staff Requirements for Preparations for Accident Cleanup E-24

E.3 DETAILS OF ACCIDENT CLEANUP IN THE AUXILIARY AND FUEL BUILDINGS E-29

E.3.1 Procedures for Accident Cleanup in the Auxiliary Fuel Buildings E-30

E.3.2 Schedules and Cleanup Worker Requirements for Accident Cleanup in the Auxiliary and Fuel Buildings . . E-32

E.3.3 Occupational Doses for Accident Cleanup in the Auxiliary and Fuel Buildings E-36

E.3.4 Staff Requirements for Accident Cleanup in the Auxiliary and Fuel Buildings E-38

E.4 DETAILS OF ACCIDENT CLEANUP IN THE CONTAINMENT BUILDING . E-39

E.4.1 Procedures for Accident Cleanup in the Containment Building E-39

E.4.1.1 Processing of Contaminated Liquids . . . E-41

E.4.1.2 Initial Decontamination of the Containment Building E-45

E.4.1.3 Defueling the Reactor E-51

E.4.1.4 Cleanup of the Primary Coolant System . . E-60

E.4.1.5 Waste Treatment and Disposal E-61

E.4.2 Schedules and Cleanup Worker Requirements for Accident Cleanup in the Containment Building . . E-65

E.4.3 Occupational Doses for Accident Cleanup in the Containment Building E-76

E.4.4 Staff Requirements for Accident Cleanup in the Containment Building E-77

REFERENCES E-88

APPENDIX F - DETAILS OF COSTS OF ACCIDENT CLEANUP AT A REFERENCE PWR F-1

F.1 DETAILS OF COSTS OF PREPARATIONS FOR ACCIDENT CLEANUP . . F-1

F.1.1 Costs of Utility Staff Labor F-4

F.1.2 Cost of Waste Management F-4

F.1.3 Cost of Energy F-6

F.1.4 Costs of Special Equipment and Facilities . . . F-7

F.1.5 Costs of Miscellaneous Supplies F-8

F.1.6 Costs of Specialty Contractors F-8

F.1.7 Costs of Nuclear Insurance and License Fees . . F-8

F.1.8 Costs of Disposal of Accumulated Spent Fuel . . F-9

F.2 DETAILS OF COSTS OF ACCIDENT CLEANUP IN THE AUXILIARY AND FUEL BUILDINGS F-10

F.2.1 Costs of Cleanup Worker Labor F-11

F.2.2 Costs of Waste Management F-12

F.2.3 Costs of Special Tools and Equipment F-12

F.2.4 Costs of Miscellaneous Supplies F-12

F.2.5 Costs of Specialty Contractors F-12

F.3 DETAILS OF COSTS OF ACCIDENT CLEANUP IN THE CONTAINMENT BUILDING F-15

F.3.1 Costs of Utility Staff Labor F-17

F.3.2 Costs of Waste Management F-19

F.3.3 Cost of Energy F-23

F.3.4 Costs of Special Tools and Equipment F-23

F.3.5 Costs of Miscellaneous Supplies F-25

F.3.6 Costs of Specialty Contractors F-25

F.3.7 Costs of Nuclear Insurance and License Fees . . F-25

F.4 SUMMARY OF THE COSTS OF ACCIDENT CLEANUP AT THE REFERENCE PWR F-25

REFERENCES F-28

APPENDIX G - DETAILS OF ACTIVITIES AND MANPOWER REQUIREMENTS FOR DECOMMISSIONING AT A REFERENCE PWR G-1

G.1 COMPARISON OF NORMAL VERSUS POST-ACCIDENT DECOMMISSIONING OF THE REFERENCE PWR G-2

G.2 DECON AT THE REFERENCE PWR G-5

G.2.1 Disassembly Methods G-9

G.2.1.1 Containment Building G-10

G.2.1.2 Fuel Building G-12

G.2.1.3 Auxiliary Building G-14

G.2.1.4 Ancillaries G-14

G.2.2 Schedules and Decommissioning Worker Requirements . G-16

G.2.2.1 Containment Building G-17

G.2.2.2 Fuel Building G-19

G.2.2.3 Auxiliary Building G-19

G.2.2.4 Overall Schedule and Decommissioning Worker Requirements G-19

G.2.3 External Occupational Radiation Doses G-23

G.2.4 Overall Staff Labor Requirements G-25

G.3 SAFSTOR AT THE REFERENCE PWR G-30

G.3.1 Methods for Preparations for Safe Storage . . . G-32

G.3.2 Schedule G-34

G.3.3 Occupational Radiation Doses and Decommissioning Worker Requirements G-36

G.3.4 Overall Staff Labor Requirements G-37

G.3.5 Continuing Care and Deferred Decontamination . . G-41

G.4 ENTOMB AT THE REFERENCE PWR G-42

G.4.1 Entombment Methods G-44

G.4.1.1 Containment Building G-45

G.4.1.2 Fuel and Auxiliary Buildings G-48

G.4.1.3 Ancillaries G-49

G.4.2 Schedule G-50

G.4.3 Occupational Radiation Doses and Decommissioning Worker Requirements G-52

G.4.4 Overall Staff Labor Requirements G-55

G.4.5 Continuing Care and Possible Deferred Decontamination of the Entombed Plant G-57

REFERENCES G-59

APPENDIX H - DETAILS OF COSTS OF DECOMMISSIONING AT A REFERENCE PWR H-1

H.1 DECON COSTS H-2

H.1.1 Costs of Staff Labor H-2

H.1.2 Costs of Waste Management H-5

H.1.2.1 Neutron-Activated Materials H-5

H.1.2.2 Contaminated Materials H-9

H.1.2.3 Radioactive Wastes H-11

H.1.3 Costs of Energy H-13

H.1.4 Costs of Special Tools and Equipment H-13

H.1.5 Costs of Miscellaneous Supplies H-13

H.1.6 Costs of Specialty Contractors H-15

H.1.7 Costs of Nuclear Insurance and License Fees . . H-16

H.2 SAFSTOR COSTS H-16

H.2.1 Costs of Preparations for Safe Storage . . . H-18

H.2.1.1 Costs of Staff Labor H-19

H.2.1.2 Costs of Waste Management H-21

H.2.1.3 Costs of Energy H-21

H.2.1.4 Costs of Special Tools and Equipment . . H-25

H.2.1.5 Costs of Miscellaneous Supplies . . . H-25

H.2.1.6 Costs of Specialty Contractors H-25

H.2.1.7 Costs of Nuclear Insurance and License Fees . H-27

H.2.2 Cost of Continuing Care During Safe Storage . . H-27

H.2.3 Cost of Deferred Decontamination to Terminate SAFSTOR H-27

H.2.4 Total SAFSTOR Costs H-30

H.3 ENTOMB COSTS H-30

H.3.1 Costs of Staff Labor H-32

H.3.2 Costs of Waste Management H-32

H.3.2.1 Neutron-Activated Materials H-34

H.3.2.2 Contaminated Materials H-37

H.3.2.3 Radioactive Wastes H-38

H.3.3 Costs of Energy H-38

H.3.4 Costs of Special Tools and Equipment H-39

H.3.5 Costs of Miscellaneous Supplies H-39

H.3.6 Costs of Specialty Contractors H-39

H.3.7 Costs of Nuclear Insurance and License Fees . . H-39

H.3.8 Costs of Continuing Care and Possible Deferred Decontamination H-42

REFERENCES H-44

APPENDIX I - COST ESTIMATING BASES I-1

I.1 LABOR COSTS I-1

I.2 WASTE PACKAGING COSTS I-1

I.2.1 Shipping Container Costs I-3

I.2.2 Overpack and Cask Charges I-4

I.2.3 Additional Shielding Costs I-4

I.2.4 Solidifying Agent Costs I-5

I.3 TRANSPORTATION COSTS I-5

I.3.1 Shipment by Exclusive-Use Truck I-5

I.3.2 Shipment by Rail I-9

I.4 WASTE DISPOSAL COSTS I-9

I.4.1 Shallow-Land Burial I-9

I.4.2 Disposal of Wastes at a Federal Repository . . . I-10

I.5 EQUIPMENT COSTS I-12

I.6 SERVICES AND SUPPLIES I-12

I.7 COST UPDATING FROM 1978 TO 1981 COST BASE I-14

REFERENCES I-16

APPENDIX J - SAFETY ASSESSMENT DETAILS J-1

J.1 TECHNICAL APPROACH J-1

J.2 ACCIDENT CLEANUP J-4

J.2.1 Public Safety Aspects of Accident Cleanup . . . J-4

J.2.1.1 Public Radiation Doses from Routine Tasks During Accident Cleanup J-5

J.2.1.2 Public Radiation Doses from Releases Due to Postulated Industrial Accidents During Accident Cleanup J-13

J.2.2 Occupational Safety Aspects of Accident Cleanup . . J-17

J.2.2.1 Occupational Radiation Doses from Accident Cleanup Activities J-17

J.2.2.2 Industrial Safety Aspects of Accident Cleanup . J-19

J.2.3 Transportation Safety Aspects of Accident Cleanup . J-20

J.2.3.1 Radiation Doses from Routine Transportation Activities During Accident Cleanup . . . J-20

J.2.3.2 Radiation Doses from Postulated Transportation Accidents During Accident Cleanup . . . J-23

J.2.3.3 Nonradiological Safety Aspects of Transportation Activities During Accident Cleanup . . . J-27

J.3 DECOMMISSIONING J-27

J.3.1 Public Safety Aspects of Post-Accident Decommissioning Activities J-27

J.3.1.1 Public Radiation Doses from Routine Tasks During Post-Accident Decommissioning J-29

J.3.1.2 Public Radiation Doses from Releases Due to Postulated Industrial Accidents During Post-Accident Decommissioning J-35

J.3.2 Occupational Safety Aspects of Post-Accident Decommissioning J-38

J.3.2.1 Occupational Radiation Doses from Decommissioning Activities J-38

J.3.2.2 Industrial Safety Aspects of Post-Accident Decommissioning J-40

J.3.3 Transportation Safety Aspects of Post-Accident Decommissioning J-42

J.3.3.1 Radiation Doses from Routine Transportation Activities During Decommissioning . . . J-42

J.3.3.2 Radiation Doses from Postulated Transportation Accidents During Decommissioning . . . J-44

J.3.3.3 Nonradiological Safety Aspects of Transportation Activities During Decommissioning . . J-46

REFERENCES J-48

APPENDIX K - DETAILS OF POST-ACCIDENT CLEANUP AND DECOMMISSIONING AT A REFERENCE BWR K-1

K.1 REFERENCE BWR FACILITY DESCRIPTION K-1

K.1.1 Plant Structures K-1

K.1.1.1 Reactor Building K-2

K.1.1.2 Turbine Generator Building K-5

K.1.1.3 Radwaste and Control Building K-7

K.1.1.4 Other Structures K-7

K.1.2 Primary Containment Vessel K-7

K.1.2.1 Reactor Vessel and Internals K-9

K.1.2.2 Reactor Water Recirculation System . . . K-10

K.1.3 Spent Fuel Handling and Storage K-11

K.1.3.1 Defueling Equipment K-13

K.1.3.2 Defueling Procedures K-13

K.2 DETAILS OF REFERENCE BWR ACCIDENT SCENARIOS AND RESULTANT CONTAMINATION LEVELS K-15

K.2.1 Fission Product Contamination K-15

K.2.2 Exposure Rate Calculations K-19

K.3 DETAILS OF ACTIVITIES AND MANPOWER REQUIREMENTS FOR ACCIDENT CLEANUP K-20

K.3.1 Details of Preparations for Accident Cleanup . . K-23

K.3.2 Details of Accident Cleanup in the Radwaste Building . K-27

K.3.3 Details of Accident Cleanup in the Reactor Building and the Containment Vessel K-32

K.3.3.1 Procedures for Accident Cleanup in the Reactor Building and the Containment Vessel . . . K-32

K.3.3.2 Schedules and Cleanup Worker Requirements for Accident Cleanup in the Reactor Building and the Containment Vessel K-35

K.3.3.3 Occupational Doses for Accident Cleanup in the Reactor Building and the Containment Vessel . K-41

K.3.3.4 Staff Requirements for Accident Cleanup in the Reactor Building and the Containment Vessel . K-45

K.4 DETAILS OF COSTS OF ACCIDENT CLEANUP K-49

K.4.1 Details of Costs of Preparations for Accident Cleanup K-49

K.4.1.1 Cost of Labor for Preparations for Accident Cleanup K-53

K.4.1.2 Cost of Waste Management K-53

K.4.1.3 Cost of Energy K-56

K.4.1.4 Cost of Special Equipment and Facilities . . K-56

K.4.1.5 Cost of Miscellaneous Supplies K-56

K.4.1.6 Cost of Nuclear Insurance and License Fees . K-56

K.4.2 Details of Costs of Accident Cleanup in the Radwaste Building K-57

K.4.2.1 Cost of Labor for Accident Cleanup in the Radwaste Building K-58

K.4.2.2 Cost of Waste Management K-59

K.4.2.3 Cost of Special Tools and Equipment K-59

K.4.2.4 Cost of Miscellaneous Supplies K-59

K.4.3 Details of Costs of Accident Cleanup in the Reactor Building and Containment Vessel K-59

K.4.3.1 Cost of Labor for Accident Cleanup in the Reactor Building and Containment Vessel . . K-64

K.4.3.2 Cost of Waste Management K-64

K.4.3.3 Cost of Energy K-73

K.4.3.4 Cost of Special Tools and Equipment . . . K-73

K.4.3.5 Cost of Miscellaneous Supplies K-74

K.4.3.6 Cost of Nuclear Insurance and License Fees . K-74

K.5 DETAILS OF ACTIVITIES AND MANPOWER REQUIREMENTS FOR DECOMMISSIONING K-74

K.5.1 Details of Planning and Preparation Activities . . K-84

K.5.2 Details of DECON at the Reference BWR K-85

K.5.2.1 DECON in the Reactor Building and the Containment Vessel K-86

K.5.2.2 DECON in the Other Buildings K-91

K.5.2.3 Schedule and Decommissioning Worker Requirements for DECON K-91

K.5.2.4 Total Staff Labor Requirements for DECON . . K-94

K.5.3 Details of SAFSTOR at the Reference BWR K-94

K.5.3.1 Preparations for Safe Storage in the Reactor Building and the Containment Vessel K-96

K.5.3.2 Preparations for Safe Storage in the Other Buildings K-101

K.5.3.3 Schedule and Decommissioning Worker Requirements for Preparations for Safe Storage K-102

K.5.3.4 Total Staff Labor Requirements for Preparations for Safe Storage K-103

K.5.3.5 Continuing Care and Deferred Decontamination . K-106

K.5.4 Details of ENTOMB at the Reference BWR K-107

K.5.4.1 ENTOMB in the Reactor Building and the Containment Vessel K-108

K.5.4.2 ENTOMB in the Other Buildings K-109

K.5.4.3 Schedule and Decommissioning Worker Requirements for ENTOMB K-112

K.5.4.4 Total Staff Labor Requirements for ENTOMB . K-114

K.5.4.5 Continuing Care and Possible Deferred Decontamination of the Entombed Plant . . K-114

K.6 DETAILS OF COSTS OF DECOMMISSIONING K-115

K.6.1 Details of DECON Costs K-117

K.6.1.1 Cost of Staff Labor K-117

K.6.1.2 Cost of Waste Management K-117

K.6.1.3 Cost of Energy K-123

K.6.1.4 Cost of Special Tools and Equipment . . . K-124

K.6.1.5 Cost of Miscellaneous Supplies K-124

K.6.1.6 Cost of Specialty Contractors K-124

K.6.1.7 Cost of Nuclear Insurance and License Fees . K-125

K.6.2 Details of SAFSTOR Costs K-126

K.6.2.1 Cost of Staff Labor K-128

K.6.2.2 Cost of Waste Management K-130

K.6.2.3 Cost of Energy K-130

K.6.2.4 Cost of Special Tools and Equipment . . . K-130

K.6.2.5 Cost of Miscellaneous Supplies K-133

K.6.2.6 Cost of Specialty Contractors K-133

K.6.2.7 Cost of Nuclear Insurance and License Fees . K-133

K.6.2.8 Cost of Continuing Care During Safe Storage . K-134

K.6.2.9 Cost of Deferred Decontamination to Terminate SAFSTOR K-135

K.6.3 Details of ENTOMB Costs K-137

K.6.3.1 Cost of Staff Labor K-137

K.6.3.2 Cost of Waste Management K-140

K.6.3.3 Cost of Energy K-140

K.6.3.4 Cost of Special Tools and Equipment K-143

K.6.3.5 Cost of Miscellaneous Supplies K-143

K.6.3.6 Cost of Specialty Contractors K-144

K.6.3.7 Cost of Nuclear Insurance and License Fees . K-144

K.6.3.8 Cost of Continuing Care During Safe Storage . K-144

K.7 SAFETY ASSESSMENT DETAILS K-146

K.7.1 Accident Cleanup Safety K-146

K.7.1.1 Public Safety Impacts of Accident Cleanup . K-146

K.7.1.2 Occupational Radiation Doses from Accident Cleanup K-148

K.7.1.3 Industrial Safety Impacts of Accident Cleanup K-150

K.7.1.4 Transportation Safety Impacts of Accident Cleanup K-152

K.7.2 Decommissioning Safety K-157

K.7.2.1 Public Safety Impacts of Post-Accident Decommissioning K-157

K.7.2.2 Occupational Radiation Doses from Post-Accident Decommissioning K-158

K.7.2.3 Industrial Safety Impacts of Post-Accident Decommissioning K-160

K.7.2.4 Transportation Safety Impacts of Post-Accident Decommissioning K-162

REFERENCES K-167

FIGURES

A.4-1	Average $(\overline{X}/Q')$ Values Versus Distance in a Sector from the Reference Site	A-6
B.0-1	Reference PWR Plant Layout	B-2
B.1-1	Plan View and Vertical Sections of the PWR Containment Building	B-4
B.1-2	Vertical Section of the PWR Containment Building, Looking East	B-6
B.1-3	PWR Reactor Pressure Vessel	B-7
B.1-4	PWR Reactor Vessel and Internals	B-9
B.1-5	PWR Steam Generator	B-11
B.1-6	PWR Pressurizer	B-13
B.1-7	PWR Emergency Core Cooling System Schematic	B-16
B.1-8	Air Flow Diagram for PWR Containment Building	B-21
B.2-1	Plan Views--PWR Fuel and Auxiliary Buildings, Grade Level	B-24
B.2-2	Plan Views--PWR Fuel and Auxiliary Buildings, Elevation 18.6 m	B-25
B.2-3	Plan Views--PWR Fuel and Auxiliary Buildings, Elevation 23.5 m	B-26
B.2-4	Plan Views--PWR Fuel and Auxiliary Buildings, Operating Floor	B-27
B.2-5	Vertical Section of PWR Auxiliary Building, Looking East	B-28
B.2-6	Vertical Section of PWR Auxiliary Building, Looking North	B-29
B.2-7	Vertical Section of PWR Fuel Building, Looking East	B-30
B.2-8	Vertical Section of PWR Fuel Building, Looking North	B-31
B.2-9	PWR Chemical Volume Control System Schematic	B-33
B.2-10	PWR Dirty Radioactive Waste System	B-37
B.2-11	PWR Clean Radioactive Waste System	B-38
B.3-1	Isometric View of the PWR Control Building	B-44
C.2-1	Radioactive Decay Scheme for $^{90}Sr \xrightarrow{\beta^-} {}^{90}Y$	C-12

C.2-2 Radioactive Decay Scheme for $^{106}Ru \xrightarrow{\beta^-} {}^{106}Rh$ . . . C-12

C.2-3 Radioactive Decay Scheme for ^{134}Cs C-14

C.2-4 Radioactive Decay Scheme for ^{137}Cs C-14

C.2-5 Radioactive Decay Scheme for $^{144}Ce \xrightarrow{\beta^-} {}^{144}Pr$ . . . C-15

C.3-1 Plane Disk Source C-17

C.3-2 Disk Source of Finite Thickness C-19

C.4-1 Exposure as a Function of Room Volume for a ^{60}Co Deposition of 1 Ci/m^2 . . . . C-22

D.1-1 Schematic Diagram of Electropolishing Cell D-12

D.1-2 Vibratory Finishing System D-13

D.1-3 Ultrasonic Cleaning System D-14

D.1-4 Cross Section of a Vacuum Blaster Nozzle D-16

D.1-5 Schematic of Concrete Spaller D-18

D.2-1 Vacuum Collection System Schematic D-23

D.2-2 Portable Filtered Ventilation Enclosure D-24

D.2-3 Portable Filtered Fume Exhauster D-25

D.5-1 Characterization of Waste Forms from Accident Cleanup and Decommissioning D-36

D.5-2 Radiation Dose Limits for Closed Exclusive-Use Vehicles . . D-48

E.0-1 Sequence of Accident Cleanup Tasks E-3

E.2-1 Conceptualized Shielded Facility for Onsite Storage of High-Activity Wastes E-19

E.2-2 Postulated Staff Organization for Preparations for Accident Cleanup E-25

E.3-1 Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Auxiliary and Fuel Buildings . . . E-35

E.3-2 Postulated Cleanup Operations Staff for Accident Cleanup in the Auxiliary and Fuel Buildings E-36

E.4-1	Process Diagram of Filter/Demineralizer System for Contaminated Water Treatment	E-43
E.4-2	Special Tool for Bottom Removal of Damaged Fuel . . .	E-57
E.4-3	Handling Shroud Concept for Removal of Damaged Fuel . . .	E-58
E.4-4	Pivoting Fixture for Packaging Damaged Fuel Assemblies . .	E-59
E.4-5	Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Containment Building Following the Scenario 1 Accident	E-67
E.4-6	Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Containment Building Following the Scenario 2 Accident	E-69
E.4-7	Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Containment Building Following the Scenario 3 Accident	E-71
E.4-8	Postulated Staff Organization for Accident Cleanup in the Containment Building	E-85
G.2-1	DECON Task Schedule and Decommissioning Worker Requirements in the PWR Containment Building Following a Scenario 2 Accident .	G-18
G.2-2	DECON Task Schedule and Sequence and Decommissioning Worker Requirements in the PWR Fuel Building Following a Scenario 2 Accident	G-20
G.2-3	Duration of PWR DECON Activities Following a Scenario 2 Accident	G-22
G.2-4	Postulated Staff Organization for Post-Accident PWR Decommissioning by DECON	G-27
G.3-1	Duration of PWR Preparations for Safe Storage Following a Scenario 2 Accident	G-34
G.4-1	Duration of PWR ENTOMB Activities Following a Scenario 2 Accident	G-51
K.1-1	Reference BWR Plant Layout	K-2
K.1-2	Vertical Section of the BWR Reactor Building	K-3
K.1-3	Major Equipment Locations in the BWR Reactor Building . .	K-4
K.1-4	BWR Power Conversion System	K-6
K.1-5	BWR Reactor Vessel and Internals	K-10

K.1-6 BWR Reactor Water Recirculation System K-12

K.2-1 Simplified Model of BWR Containment K-21

K.3-1 Task schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident K-28

K.3-2 Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 1 Accident K-36

K.3-3 Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 2 Accident K-37

K.3-4 Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 3 Accident K-39

K.5-1 Overall Schedule and Sequence for DECON at the Reference BWR Following a Scenario 2 Accident K-92

K.5-2 Overall Schedule and Sequence for Preparations for Safe Storage at the Reference BWR Following a Scenario 2 Accident K-102

K.5-3 Overall Schedule and Sequence for ENTOMB at the Reference BWR Following a Scenario 2 Accident K-113

TABLES

A.2-1	Population Distribution Around the Reference Site for the Year 2000	A-2
A.4-1	Monthly Meteorological Statistics at the Reference Site	A-3
B.1-1	PWR Reactor Vessel Design Parameters	B-8
B.1-2	PWR Reactor Coolant Pump Data	B-12
B.1-3	PWR Reactor Coolant Piping Design Parameters	B-14
B.1-4	PWR Major System Piping Data	B-14
B.1-5	Equipment List: PWR Containment Building	B-23
B.2-1	Equipment List: PWR Auxiliary and Fuel Buildings	B-41
C.1-1	Estimated Inventory of Radioactivity Released from Damaged Reactor Fuel During Scenario 1 Accident	C-2
C.1-2	Estimated Inventory of Radioactivity Released from Damaged Reactor Fuel During Scenario 2 Accident	C-4
C.1-3	Estimated Inventory of Radioactivity Released from Damaged Reactor Fuel During Scenario 3 Accident	C-6
C.1-4	Estimated Gap and Rod-Melt Release Fractions	C-9
C.3-1	Dose Buildup Factors for Concrete for 0.662-MeV Gamma Rays	C-21
C.5-1	Contamination Condition of TMI-2 Reactor Building Approximately 1 Year After the Accident	C-24
C.5-2	Comparison of TMI-2 Accident Parameters with Reference PWR Accident Parameters	C-27
D.1-1	Comparison of Selected Chemical Decontamination Methods	D-2
D.1-2	Comparison of Major Concrete Surface Removal Techniques	D-15
D.3-1	Special Tools and Equipment for Post-Accident Cleanup and Decommissioning of the Reference PWR	D-27
D.5-1	Waste Management Alternatives for Solid Wastes from Accident Cleanup and Decommissioning	D-41
D.5-2	Waste Management Alternatives for Liquid Wastes from Accident Cleanup and Decommissioning	D-42

D.5-3	Waste Management Alternatives for Process Solids from Accident Cleanup and Decommissioning	D-43
D.6-1	Essential Systems and Services for Accident Cleanup and Decommissioning	D-49
D.8-1	Basic Environmental Monitoring Program for Post-Accident Decommissioning	D-53
E.2-1	Estimated Occupational Doses to Workers Entering Containment During Preparations for Accident Cleanup	E-23
E.2-2	Estimated Utility Staff Labor Requirements for Preparations for Accident Cleanup	E-28
E.3-1	Packaging and Disposal Requirements for Radioactive Wastes from Accident Cleanup in the Auxiliary and Fuel Buildings	E-33
E.3-2	Estimated Occupational Radiation Doses for Accident Cleanup in the Auxiliary and Fuel Buildings	E-37
E.3-3	Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Auxiliary and Fuel Buildings	E-39
E.4-1	Average General Area Exposure Rates in the PWR Containment Building at the Completion of Accident Cleanup Operations	E-40
E.4-2	Packaging and Disposal Requirements for Radioactive Wastes from PWR Accident Cleanup	E-63
E.4-3	Minimum Time Requirements for Processing Contaminated Liquids from Accident Cleanup in the Containment Building	E-75
E.4-4	Estimated Occupational Radiation Doses for Accident Cleanup in the Containment Building Following the Scenario 1 Accident	E-78
E.4-5	Estimated Occupational Radiation Doses for Accident Cleanup in the Containment Building Following the Scenario 2 Accident	E-79
E.4-6	Estimated Occupational Radiation Doses for Accident Cleanup in the Containment Building Following the Scenario 3 Accident	E-80
E.4-7	Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Containment Building Following the Scenario 1 Accident	E-82
E.4-8	Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Containment Building Following the Scenario 2 Accident	E-82

H.2-9	Total Costs of SAFSTOR at the Reference PWR Following a Scenario 2 Accident	H-30
H.3-1	Summary of Estimated Costs of ENTOMB at the Reference PWR Following a Scenario 2 Accident	H-31
H.3-2	Estimated Costs of Staff Labor During ENTOMB at the Reference PWR Following a Scenario 2 Accident	H-33
H.3-3.	Estimated Costs of Radioactive Waste Management During PWR ENTOMB Following a Scenario 2 Accident	H-35
H.3-4	Costs of Special Tools and Equipment for ENTOMB at the Reference PWR Following a Scenario 2 Accident	H-40
H.3-5	Estimated Costs of Miscellaneous Supplies During ENTOMB at the Reference PWR Following a Scenario 2 Accident	H-41
H.3-6	Estimated Costs of Nuclear Liability Insurance and License Fees During ENTOMB at the Reference PWR Following a Scenario 2 Accident	H-41
I.1-1	Decommissioning Labor Cost Data	I-2
I.2-1	Unit Costs of Shipping Containers for Radioactive Materials	I-3
I.2-2	Rental Charges for Reusable Shielded Casks	I-4
I.2-3	Solidifying Agent Costs	I-5
I.3-1	Transportation Rates for Legal-Size and -Weight Shipments	I-7
I.3-2	Additional Charges When Gross Vehicle Weight Exceeds 21.32 Mg	I-8
I.3-3	Requirements for Oversize Truck Shipments	I-8
I.3-4	Example Shipping Costs of Truck Shipments	I-9
I.4-1	Commercial Shallow-Land Burial Charges	I-11
I.5-1	Special Tools and Equipment	I-13
I.7-1	Decommissioning Cost Updating Factors: 1978 to 1981	I-15
J.2-1	Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases Resulting from Routine Tasks During Post-Accident Cleanup After the PWR Scenario 1 Accident	J-7
J.2-2	Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases Resulting from Routine Tasks During Post-Accident Cleanup After the PWR Scenario 2 Accident	J-8

G.4-2 Decommissioning Worker Requirements for ENTOMB at the Reference PWR Following a Scenario 2 Accident G-54

G.4-3 Overall Staff Labor Requirements for ENTOMB at the Reference PWR Following a Scenario 2 Accident G-56

H.1-1 Summary of Estimated Costs of DECON at the Reference PWR Following a Scenario 2 Accident H-3

H.1-2 Estimated Costs of Staff Labor During DECON at the Reference PWR Following a Scenario 2 Accident H-4

H.1-3 Estimated Costs of Radioactive Waste Management During PWR DECON Following a Scenario 2 Accident H-7

H.1-4 Costs of Special Tools and Equipment for DECON at the Reference PWR Following a Scenario 2 Accident H-14

H.1-5 Estimated Costs for Miscellaneous Supplies During DECON at the Reference PWR Following a Scenario 2 Accident H-15

H.1-6 Estimated Costs of Nuclear Liability Insurance and License Fees During DECON at the Reference PWR Following a Scenario 2 Accident H-16

H.2-1 Summary of Estimated Costs of SAFSTOR at the Reference PWR Following a Scenario 2 Accident H-17

H.2-2 Estimated Costs of Staff Labor During Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident . H-20

H.2-3 Estimated Costs of Radioactive Waste Management During PWR Preparations for Safe Storage Following a Scenario 2 Accident H-23

H.2-4 Costs of Special Tools and Equipment for Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident . H-26

H.2-5 Estimated Costs of Miscellaneous Supplies During Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident H-27

H.2-6 Estimated Costs of Nuclear Liability Insurance and License Fees During Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident H-28

H.2-7 Estimated Annual Costs During Safe Storage of the Reference PWR H-29

H.2-8 Comparison of DECON and Deferred Decontamination Costs for the Reference PWR H-29

E.4-9 Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Containment Building Following the Scenario 3 Accident . . . E-83

E.4-10 Estimated Utility Staff Labor Requirements for Accident Cleanup in the Containment Building Following the Postulated PWR Accidents E-87

F.1-1 Summary of Estimated Costs of Preparations for Accident Cleanup at the Reference PWR F-2

F.1-2 Estimated Annual Costs of Utility Staff Labor for Preparations for Accident Cleanup at the Reference PWR F-5

F-1-3 Waste Management Costs for the Disposal of Spent Fuel Racks . F-6

F-1-4 Estimated Costs of Nuclear Liability Insurance and License Fees During Preparations for Accident Cleanup F-9

F.1-5 Estimated Cost of Disposal of Accumulated Spent Fuel During Preparations for Accident Cleanup F-9

F.2-1 Summary of Estimated Costs of Accident Cleanup in the Auxiliary and Fuel Buildings at the Reference PWR . . . F-10

F.2-2 Estimated Cleanup Worker Labor Costs for Accident Cleanup in the Auxiliary and Fuel Buildings F-11

F.2-3 Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Auxiliary and Fuel Buildings at the Reference PWR F-13

F.2-4 Estimated Costs of Special Tools and Equipment for Accident Cleanup in the Auxiliary and Fuel Buildings F-14

F.2-5 Estimated Costs of Miscellaneous Supplies for Accident Cleanup in the Auxiliary and Fuel Buildings F-15

F.3-1 Summary of Estimated Costs of Accident Cleanup in the Containment Building at the Reference PWR F-16

F.3-2 Estimated Costs of Utility Staff Labor for Accident Cleanup in the Containment Building at the Reference PWR F-18

F.3-3 Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Containment Building at the Reference PWR Following the Scenario 1 Accident . . . F-20

F.3-4 Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Containment Building at the Reference PWR Following the Scenario 2 Accident . . . F-21

F.3-5 Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Containment Building at the Reference PWR Following the Scenario 3 Accident . . . F-22

F.3-6 Estimated Costs of Special Tools and Equipment for Accident Cleanup in the Containment Building F-24

F.3-7 Estimated Costs of Miscellaneous Supplies for Accident Cleanup in the Containment Building F-26

F.3-8 Estimated Costs of Nuclear Liability Insurance and License Fees During Accident Cleanup in the Containment Building . . F-27

F.4-1 Summary of Estimated Total Costs of Accident Cleanup Following the Reference Accidents F-27

G.1-1 Comparison of Activities for Normal Versus Post-Accident DECON at the Reference PWR G-6

G.1-2 Comparison of Activities for Normal Versus Post-Accident SAFSTOR at the Reference PWR G-7

G.1-3 Comparison of Activities for Normal Versus Post-Accident ENTOMB at the Reference PWR G-8

G.2-1 Summary of Direct Decommissioning Worker Requirements for DECON at the Reference PWR Following a Scenario 2 Accident . . G-21

G.2-2 Estimated Occupational Radiation Doses for DECON at the Reference PWR Following a Scenario 2 Accident G-24

G.2-3 Adjustments to Decommissioning Worker Requirements to Comply with Occupational Radiation Dose Limitations for DECON at the Reference PWR Following a Scenario 2 Accident . . . G-26

G.2-4 Overall Staff Labor Requirements for DECON at the Reference PWR Following a Scenario 2 Accident G-29

G.3-1 Estimated Occupational Radiation Doses for Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident . G-38

G.3-2 Decommissioning Worker Requirements for Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident . G-39

G.3-3 Overall Staff Labor Requirements for Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident . . G-40

G.4-1 Estimated Occupational Radiation Doses for ENTOMB at the Reference PWR Following a Scenario 2 Accident G-53

J.2-3 Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases Resulting from Routine Tasks During Post-Accident Cleanup After the PWR Scenario 3 Accident . . J-9

J.2-4 Radiation Doses to the Population from Atmospheric Releases Resulting from Routine Tasks During Post-Accident Cleanup After the PWR Scenario 1 Accident J-10

J.2-5 Radiation Doses to the Population from Atmospheric Releases Resulting from Routine Tasks During Post-Accident Cleanup After the PWR Scenario 2 Accident J-11

J.2-6 Radiation Doses to the Population from Atmospheric Releases Resulting from Routine Tasks During Post-Accident Cleanup After the PWR Scenario 3 Accident J-12

J.2-7 Radiation Doses to the Maximum-Exposed Individual from Releases Due to Industrial Accidents During Post-Accident Cleanup - Scenario 1 . . . . J-14

J.2-8 Radiation Doses to the Maximum-Exposed Individual from Releases Due to Industrial Accidents During Post-Accident Cleanup - Scenario 2 . . . . J-15

J.2-9 Radiation Doses to the Maximum-Exposed Individual from Releases Due to Industrial Accidents During Post-Accident Cleanup - Scenario 3 . . . . J-16

J.2-10 Summary of Estimated Occupational Radiation Doses from Accident Cleanup of the Reference PWR Following a Scenario 1 Accident J-18

J.2-11 Summary of Estimated Occupational Radiation Doses from Accident Cleanup of the Reference PWR Following a Scenario 2 Accident J-18

J.2-12 Summary of Estimated Occupational Radiation Doses from Accident Cleanup of the Reference PWR Following a Scenario 3 Accident J-19

J.2-13 Estimated Occupational Lost-Time Injuries and Fatalities Resulting from Accident Cleanup Activities J-21

J.2-14 Estimated Radiation Doses from Truck Shipments of Radioactive Materials During Accident Cleanup J-24

J.2-15 Estimated Radiation Dose from Rail Transport of Spent Fuel and Fuel Debris During Accident Cleanup J-25

J.2-16 Estimated Casualties from Transportation Accidents During Accident Cleanup at the Reference PWR J-28

J.3-1 Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine DECON Tasks . . . J-31

J.3-2 Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine Preparations for Safe Storage Tasks J-32

J.3-3 Radiation Doses to the Population from Atmospheric Releases During Routine DECON Tasks J-33

J.3-4 Radiation Doses to the Population from Atmospheric Releases During Routine Preparations for Safe Storage Tasks . . . J-34

J.3-5 Radiation Doses to the Maximum-Exposed Individual from Releases Due to Industrial Accidents During Post-Accident DECON . . J-36

J.3-6 Radiation Doses to the Maximum-Exposed Individual from Releases Due to Industrial Accidents During Post-Accident Preparations for Safe Storage J-37

J.3-7 Summary of Estimated Occupational Radiation Doses from DECON at the Reference PWR Following a Scenario 2 Accident . . J-39

J.3-8 Summary of Estimated Occupational Radiation Doses from Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident J-39

J.3-9 Summary of Estimated Occupational Radiation Doses from ENTOMB at the Reference PWR Following a Scenario 2 Accident . . J-39

J.3-10 Summary of Estimated Occupational Radiation Doses During All Phases of SAFSTOR Following a Scenario 2 Accident . . J-40

J.3-11 Estimated Occupational Lost-Time Injuries and Fatalities During Post-Accident Decommissioning J-41

J.3-12 Estimated Radiation Doses from Truck Shipments of Radioactive Materials During Post-Accident Decommissioning of the Reference PWR J-43

J.3-13 Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Truck Transportation Accidents . . J-45

J.3-14 Estimated Casualties from Transportation Accidents During Post-Accident Decommissioning of the Reference PWR J-47

K.1-1 BWR Reactor Vessel Design Parameters K-11

K.2-1	Reference BWR Accident Parameters	K-16
K.3-1	Estimated Utility Staff Labor Requirements for Preparations for Accident Cleanup at the Reference BWR	K-25
K.3-2	Estimated Occupational Radiation Doses to Workers Entering the Reactor Building During Preparations for Accident Cleanup at the Reference BWR	K-26
K.3-3	Estimated Occupational Radiation Doses to Workers During Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident	K-30
K.3-4	Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident	K-31
K.3-5	Estimated Occupational Radiation Doses for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 1 Accident	K-42
K.3-6	Estimated Occupational Radiation Doses for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 2 Accident	K-43
K.3-7	Estimated Occupational Radiation Doses for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 3 Accident	K-44
K.3-8	Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Scenario 1 Accident	K-46
K.3-9	Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Scenario 2 Accident	K-47
K.3-10	Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Scenario 3 Accident	K-48
K.3-11	Estimated Utility Staff Labor Requirement for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Accidents	K-50

K.4-1	Summary of Estimated Total Costs of Accident Cleanup Following the Reference BWR Accidents	K-51
K.4-2	Summary of Estimated Costs of Preparations for Accident Cleanup Following the Reference BWR Accidents	K-52
K.4-3	Estimated Costs of Utility Staff Labor for Preparations for Accident Cleanup Following the Reference BWR Accidents	K-54
K.4-4	Estimated Waste Management Costs for the Disposal of BWR Spent Fuel Racks	K-55
K.4-5	Summary of Estimated Costs of Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident	K-57
K.4-6	Estimated Cleanup Worker Labor Costs for Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident	K-58
K.4-7	Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident	K-61
K.4-8	Summary of Estimated Costs of Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Accidents	K-63
K.4-9	Estimated Costs of Utility Staff Labor for Accident Cleanup in the Reactor Building Following the Postulated BWR Accidents	K-65
K.4-10	Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Scenario 1 Accident	K-67
K.4-11	Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Scenario 2 Accident	K-69
K.4-12	Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Scenario 3 Accident	K-71
K.4-13	Estimated Costs of Miscellaneous Supplies for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Accidents	K-75
K.5-1	Comparison of Activities for Normal Versus Post-Accident DECON at the Reference BWR	K-77

K.5-2	Comparison of Activities for Normal Versus Post-Accident SAFSTOR at the Reference BWR	K-79
K.5-3	Comparison of Activities for Normal Versus Post-Accident ENTOMB at the Reference BWR	K-81
K.5-4	Estimated Direct Worker Requirements for DECON in the Reactor Building and Containment Vessel Following a Scenario 2 Accident	K-88
K.5-5	Estimated Occupational Radiation Doses for DECON in the Reactor Building and Containment Vessel Following a Scenario 2 Accident	K-89
K.5-6	Adjustments to Decommissioning Worker Requirements to Comply with Occupational Radiation Dose Limitations for DECON in the Reactor Building and Containment Vessel Following a Scenario 2 Accident	K-90
K.5-7	Adjusted Decommissioning Worker Requirements for DECON at the Reference BWR Following a Scenario 2 Accident	K-93
K.5-8	Overall Staff Labor Requirements for DECON at the Reference BWR Following a Scenario 2 Accident	K-95
K.5-9	Estimated Direct Worker Requirements for Preparations for Safe Storage in the Reactor Building and Containment Vessel Following a Scenario 2 Accident	K-99
K.5-10	Estimated Occupational Radiation Doses for Preparations for Safe Storage in the Reactor Building and Containment Vessel Following a Scenario 2 Accident	K-100
K.5-11	Adjustments to Decommissioning Worker Requirements to Comply with Occupational Radiation Dose Limitations for Preparations for Safe Storage in the Reactor Building and Containment Vessel Following a Scenario 2 Accident	K-101
K.5-12	Adjusted Decommissioning Worker Requirements for Preparations for Safe Storage at the Reference BWR Following a Scenario 2 Accident	K-104
K.5-13	Overall Staff Labor Requirements for Preparations for Safe Storage at the Reference BWR Following a Scenario 2 Accident	K-105
K.5-14	Estimated Direct Worker Requirements for ENTOMB in the Reactor Building and Containment Vessel Following a Scenario 2 Accident	K-110

K.5-15 Estimated Occupational Radiation Doses for ENTOMB in the Reactor Building and Containment Vessel Following a Scenario 2 Accident K-111

K.5-16 Adjustments to Decommissioning Worker Requirements to Comply with Occupational Radiation Dose Limitations for ENTOMB in the Reactor Building and Containment Vessel Following a Scenario 2 Accident K-112

K.5-17 Adjusted Decommissioning Worker Requirements for ENTOMB at the Reference BWR Following a Scenario 2 Accident K-114

K.5-18 Overall Staff Labor Requirements for ENTOMB at the Reference BWR Following a Scenario 2 Accident K-116

K.6-1 Summary of Estimated Costs of DECON at the Reference BWR Following a Scenario 2 Accident K-118

K.6-2 Estimated Costs of Staff Labor During DECON at the Reference BWR Following a Scenario 2 Accident K-119

K.6-3 Estimated Costs of Radioactive Waste Management During BWR DECON Following a Scenario 2 Accident K-121

K.6-4 Estimated Costs of Miscellaneous Supplies During DECON at the Reference BWR Following a Scenario 2 Accident . . . K-125

K.6-5 Estimated Costs of Nuclear Liability Insurance and License Fees During DECON at the Reference BWR Following a Scenario 2 Accident K-126

K.6-6 Summary of Estimated Costs of SAFSTOR at the Reference BWR Following a Scenario 2 Accident K-127

K.6-7 Estimated Costs of Staff Labor During Preparations for Safe Storage at the Reference BWR Following a Scenario 2 Accident . K-129

K.6-8 Estimated Costs of Radioactive Waste Management During BWR Preparations for Safe Storage Following a Scenario 2 Accident . K-131

K.6-9 Estimated Costs of Miscellaneous Supplies During Preparations for Safe Storage at the Reference BWR Following a Scenario 2 Accident K-134

K.6-10 Estimated Costs of Nuclear Liability Insurance and License Fees During Preparations for Safe Storage at the Reference BWR Following a Scenario 2 Accident K-135

K.6-11 Estimated Annual Costs During Safe Storage of the Reference BWR K-136

K.6-12	Comparison of DECON and Deferred Decontamination Costs for the Reference BWR	K-137
K.6-13	Summary of Estimated Costs of ENTOMB at the Reference BWR Following a Scenario 2 Accident	K-138
K.6-14	Estimated Costs of Staff Labor During ENTOMB at the Reference BWR Following a Scenario 2 Accident	K-139
K.6-15	Estimated Costs of Radioactive Waste Management During BWR ENTOMB Following a Scenario 2 Accident	K-141
K.6-16	Estimated Costs of Miscellaneous Supplies During ENTOMB at the Reference BWR Following a Scenario 2 Accident	K-143
K.6-17	Estimated Costs of Nuclear Liability Insurance and License Fees During ENTOMB at the Reference BWR Following a Scenario 2 Accident	K-145
K.7-1	Summary of Estimated Occupational Radiation Doses from Accident Cleanup at the Reference BWR	K-149
K.7-2	Estimated Occupational Lost-Time Injuries and Fatalities During Accident Cleanup at the Reference BWR	K-151
K.7-3	Estimated Radiation Doses from Routine Transportation Activities During Accident Cleanup at the Reference BWR	K-154
K.7-4	Estimated Casualties from Transportation Accidents During Accident Cleanup at the Reference BWR	K-156
K.7-5	Summary of Estimated Occupational Radiation Doses from Decommissioning at the Reference BWR Following a Scenario 2 Accident	K-158
K.7-6	Summary of Estimated Occupational Radiation Doses During All Phases of SAFSTOR	K-159
K.7-7	Estimated Occupational Lost-Time Injuries and Fatalities During Decommissioning at the Reference BWR Following a Scenario 2 Accident	K-161
K.7-8	Estimated Radiation Doses from Routine Truck Shipments of Radioactive Materials During Post-Accident Decommissioning at the Reference BWR	K-163
K.7-9	Estimated Casualties from Truck Accidents During Post-Accident Decommissioning at the Reference BWR	K-166

APPENDIX A

REFERENCE SITE DESCRIPTION

The reference site used to assess the public safety of post-accident decommissioning operations is the same as that used in previous LWR decommissioning studies.[1,2] Only information directly relating to the radiation exposure pathway analysis, required for estimating radiation doses to the public from decommissioning operations, is included here. The meteorological parameters and population distribution are based on information presented in Reference 3. Other necessary site information is based on data reported in the site description of an operating nuclear power station.[4] Information in this appendix is believed to be representative of many existing and potential LWR power station sites in the midwestern and middle-southeastern United States.

Individual features of any actual LWR site may differ from those of the reference site. However, it is believed that the use of a reference site results in a more meaningful overall analysis of the potential impacts associated with decommissioning most LWR facilities. Site-specific assessments would be required for particular LWR power stations.

A.1 SITE LOCATION AND SIZE

The reference site is located in a rural area with characteristics similar to those found in the midwestern or middle-southeastern United States. The site occupies about 4.7 km^2 in a rectangular shape of about 2 km by 2.35 km. A moderate-size river with an average flow rate of 1420 m^3/sec flows through one corner of the site.

A.2 DEMOGRAPHY

The site is located in a rural area with a relatively low population density. The highest population densities occur at distances of 20 to 60 km. Population distribution data are given in Table A.2-1. The total population residing within an 80-km radius of the reference site is about 3.52 million.

TABLE A.2-1. Population Distribution Around the Reference Site for the Year 2000[3]

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

(a) It is assumed that the population in each annulus is uniformly divided into 16 uniform 22.5-degree sectors.
(b) Totals are rounded to three significant figures.
(c) Indicates a population density less than 1.0/km^2.

A.3 LAND USE

Use of any part of the total site area for anything other than reactor operations is assumed to be prohibited during the operational lifetime of the reactor. The major plant facilities are located inside a 0.12-km^2, fenced portion of the site. The minimum distance from the point of airborne release to the outer site boundary is 1 km. The outer site boundary is fenced and marked.

About 80% of the land within 20 km of the site is used for farming. The main crops are soybeans (60%), corn, oats, and other grains (30%) and hay (10%). It is expected that this area will remain largely agricultural, and that the population will not change significantly because of reactor operations.

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

There are large truck gardens in the area. The nearest dwelling (the residence location of the maximum-exposed individual for the public safety analysis in the study) is a farm located about 1.3 km from the site. A milk cow is kept at this farm and is maintained on fresh pasture 7 months of the year. A family garden with a growing season of 5 months is kept for fresh vegetables. River water is used to irrigate the crops on this farm.

A.4 METEOROLOGY

The reference site has a typical continental climate. It is characterized by wide variations in temperature, modest winter precipitation, normally ample spring and summer rainfall, and a general tendency to extremes in all climatic features. January is the coldest month and July is the warmest. Table A.4-1 shows monthly meteorological statistics.

TABLE A.4-1. Monthly Meteorological Statistics at the Reference Site

	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
Air Temperature (°C)												
Maximum	-6.1	-4.4	3.3	12.8	20	25	28.3	26.7	22.2	15	4.4	-3.3
Minimum	-16.1	-14.4	-6.7	1.7	7.8	13.3	16.1	15	10	3.9	-4.4	-12.2
Mean	-11.1	-9.4	-1.7	7.2	13.9	18.9	22.2	21.1	16.1	9.4	0.0	-7.8
Extreme Maximum	15	16.1	27.8	32.8	40.6	39.4	41.7	40	40.6	32.2	23.9	17.2
Extreme Minimum	-38.9	-36.7	-34.4	-15.6	-6.7	0.6	5.6	3.3	-5.6	-13.3	-27.8	-33.9
Mean Relative Humidity (%)												
	74	75	73	66	62	66	68	70	70	66	73	78

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

The average annual rainfall in the area is 610 mm. The months of May through September have the greatest amounts of rainfall, with an average during this period of 432 to 457 mm (70% of the annual total). The maximum

24-hr total rainfall for the period 1894 through 1965 was 127 mm and occurred in May. Thunderstorms, with an average annual frequency of 36, are the chief source of rain from May through September. Snowfall in the area averages 1070 mm annually, with occurrences recorded in all months except June, July, and August. Extremes of record in annual snowfall are 152 mm minimum and 2235 mm maximum.

The annual distribution of winds is predominantly bimodal. This bimodal distribution is characteristic of the seasonal wind distributions as well. The average wind speed for spring is 11 km/hr and for the other seasons is about 16 km/hr. The maximum reported wind speed of 148 km/hr was associated with a tornado. Tornadoes and other severe storms occur occasionally. The probability of a tornado striking a given point in this area is about 5×10^{-4} per year. For design purposes, a wind velocity of 480 km/hr is assumed to be associated with tornadoes.

Natural fog that restricts visibility to 0.4 km or less occurs about 30 hr/yr. Icing caused by freezing rain can occur between October and April, with an average of one to two storms per year.

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

Data from a number of river sites for nuclear power reactors[3] are used to calculate the "typical" annual atmospheric dispersion pattern in an average 22.5-degree sector around the site. This is done by calculating the dispersion factor, $\overline{\chi}/Q'$, for each sector at selected downwind distances and then calculating the average dispersion factor at each distance. In other words, the dispersion factors in those sectors corresponding to overland trajectories are added without regard to direction and divided by the number of sectors involved. Thus, an average dispersion factor is obtained for each selected downwind distance for all 16 sectors.

Standard groups of meteorological data are interpolated from the specific site data. The groupings provide four stability classes based on vertical temperature gradient and five wind speed classes based on the Beaufort wind scale.[5] The stability classes are based on Reference 6 information, with Pasquill Classes A, B, and C classified as B (unstable); Pasquill Class D (neutral); Pasquill Class E (slightly stable); and Pasquill Classes F and G as F (moderately stable).

Where wind-speed data are available for only one height, the measured values are extrapolated to the 10-m level for ground-level release calculations and to the 150-m level for reference-LWR stack release calculations. Where measurements at two heights are available, the highest is extrapolated to 150 m and the lowest to 10 m, using a standard power-law extrapolation procedure.[5]

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

To assess the potential effect of increased stack height, atmospheric dispersion factors for stack heights of 150, 200, and 300 m are estimated from the original joint frequency distributions of the information from Reference 3. These values are graphically presented in Figure A.4-1.

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

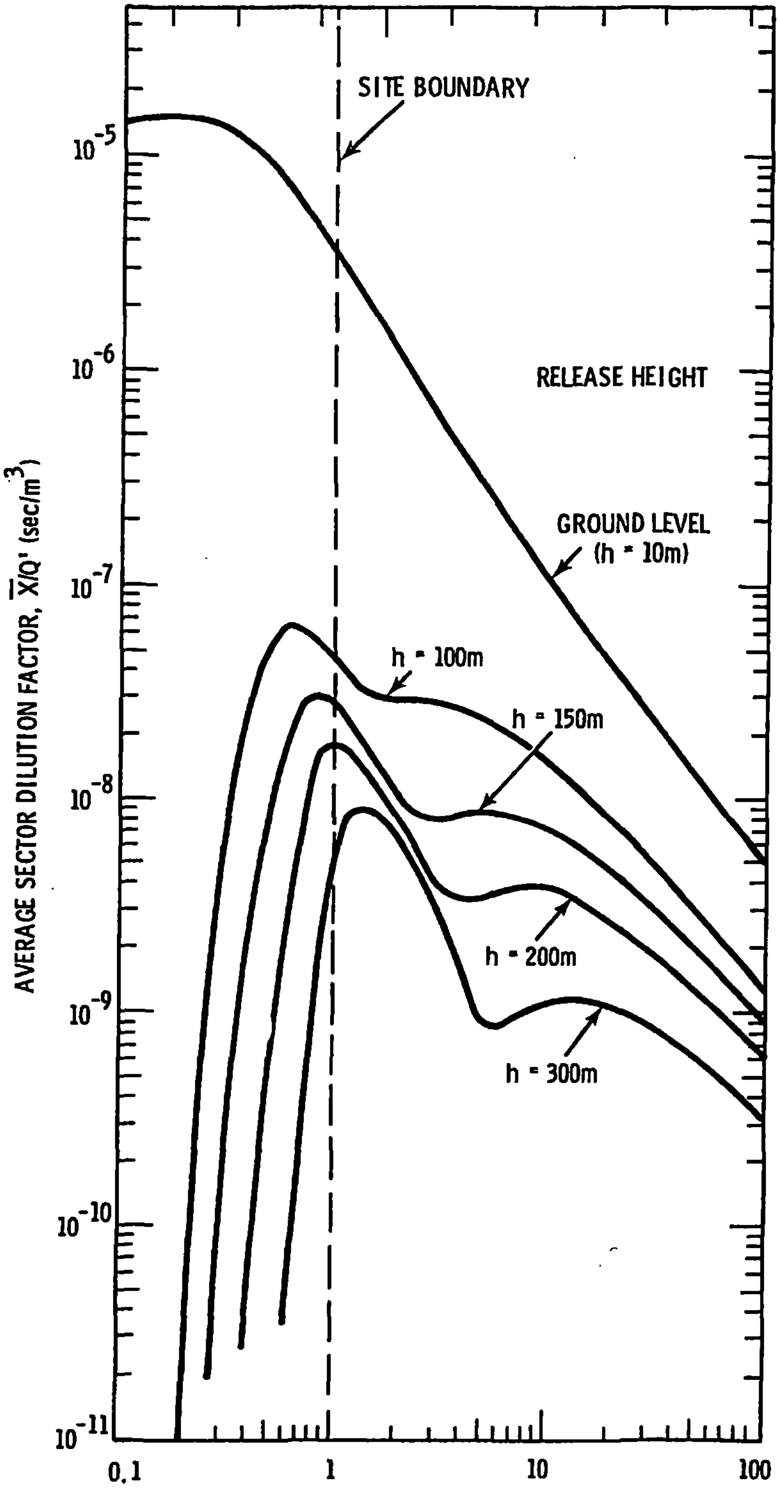


FIGURE A.4-1. Average ($\overline{X}/Q'$) Values Versus Distance in a Sector from the Reference Site

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

3. Final Environmental Statement Concerning Proposed Rule-Making Action: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criteria "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, WASH-1258, Volume 1 of 3, Figure 6B-1, p. 6B-43; Figure 6C-8, p. 6C-12; and Table 6B-6, p. 6B-36, U.S. Atomic Energy Commission, Washington, D.C., July 1973.

4. Final Environmental Statement Related to Operation of Monticello Nuclear Generating Plant, Docket No. 50-263, pp. II-15 to II-26, U.S. Atomic Energy Commission, Washington, D.C., November 1972.

5. D. H. Slade, ed., Meteorology and Atomic Energy 1968, TID-24190, p. 73, U.S. Atomic Energy Commission, Washington, D.C., July 1968.

6. Regulatory Guide 1.23, Onsite Meteorological Programs (Safety Guide 23), U.S. Atomic Energy Commission, Washington, D.C., February 1972.

APPENDIX B

REFERENCE PWR FACILITY DESCRIPTION

The reference pressurized water reactor (PWR) is the 3500-MWt (1175-MWe) Trojan Nuclear Plant at Rainier, Oregon, operated by the Portland General Electric Company. The description of the Trojan reactor presented in this appendix is intended to provide the background for understanding the estimates of time and manpower requirements and waste volumes for post-accident cleanup and decommissioning that are presented in other chapters and appendices of this report. This description is based primarily on the PWR facility description in Appendix A of Reference 1; additional details are presented in Reference 1.

The PWR facility description in Appendix A of Reference 1 is based on the Trojan Final Safety Analysis Report,(2) the RESAR-3 Preliminary Safety Analysis Report,(3) the SNUPPS Preliminary Safety Analysis Report,(4) and drawings and other data supplied by personnel of the Portland General Electric Company.

The site layout for the reference PWR is shown in Figure B.0-1. The principal structures located on the reference plant site are:

- containment building - houses the nuclear steam supply system that includes the nuclear reactor, the steam generators, the pressurizer, and associated shielding and auxiliary fluid systems
- auxiliary building - houses the liquid radwaste systems, the filter and ion exchanger vaults, the waste gas treatment system, and the ventilation equipment for the containment, fuel, and auxiliary buildings
- fuel building - houses new and spent fuel storage and handling facilities, the solid radwaste system, and the makeup water treatment system

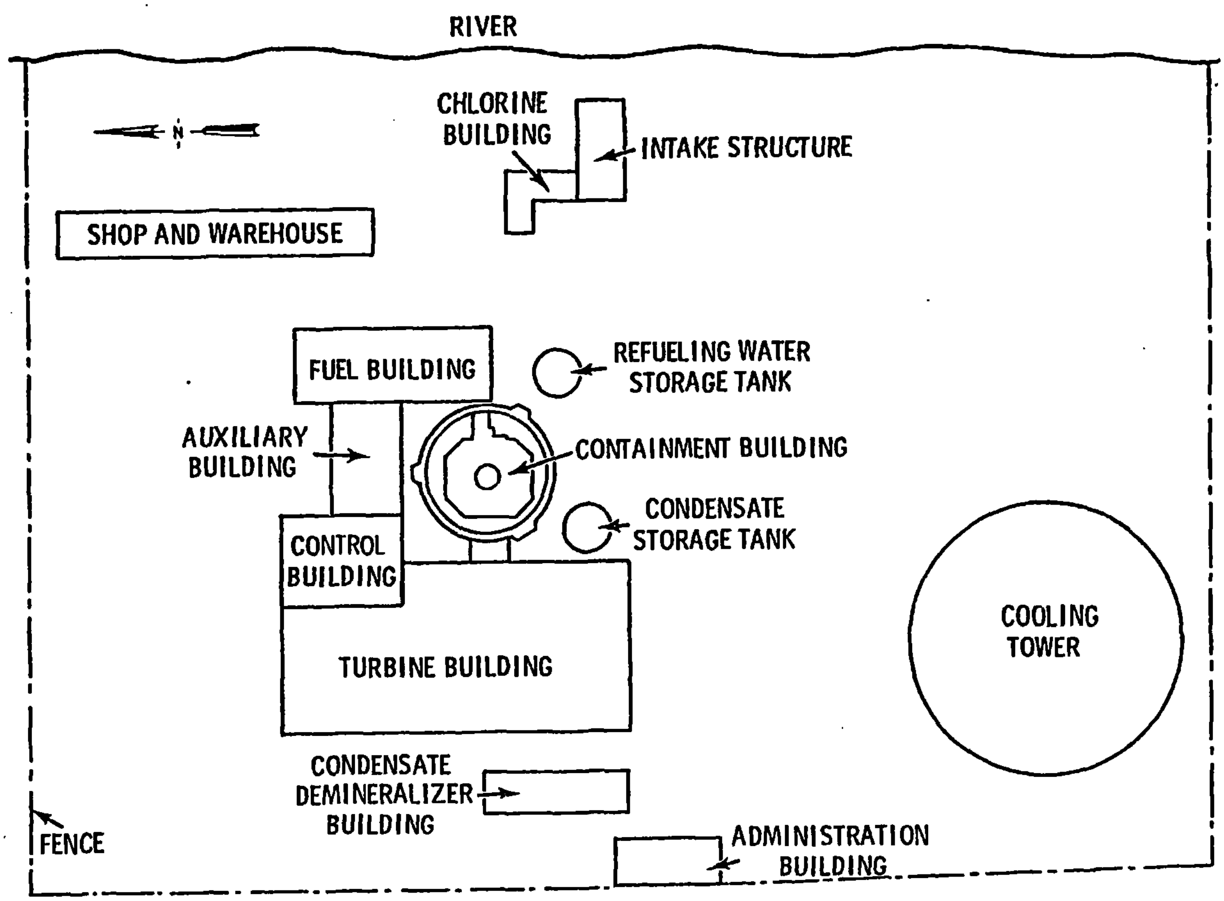


FIGURE B.0-1. Reference PWR Plant Layout

- control building - houses the reactor control room and process control laboratories and counting rooms
- turbine building - houses the power conversion equipment
- cooling tower - a hyperbolic natural-draft cooling tower for dissipation of waste heat from the turbine condenser system
- condensate demineralizer building - houses the equipment for condensate demineralization and expended resin disposal
- shop and warehouse - houses warehouse and shop facilities
- administration building - houses the gatehouse and offices for the plant superintendent and staff.

The containment, auxiliary, fuel, and control buildings are the ones on the site that would require the major decontamination and decommissioning effort. Brief descriptions of these buildings and the equipment they contain are given in the following sections. Descriptions of other buildings on the site are found in Section 7 and Appendix A of Reference 1.

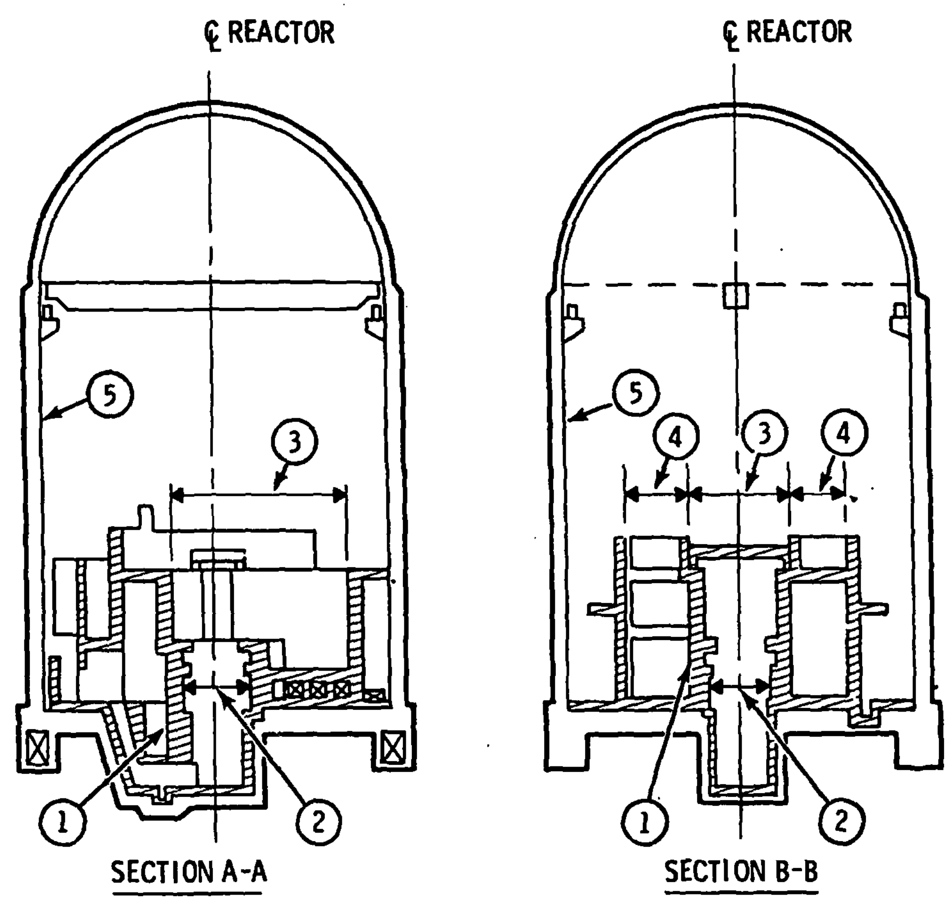
B.1 CONTAINMENT BUILDING

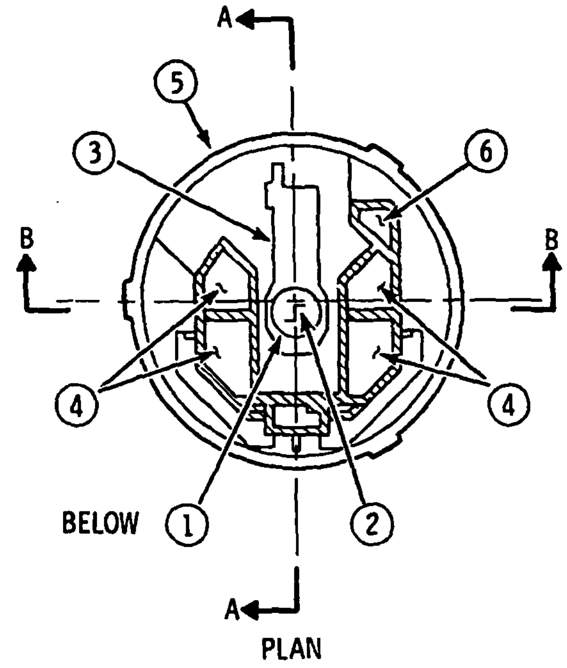
The containment building consists essentially of two structures on a common foundation. One is the containment which provides a leaktight vessel and biological shielding for normal and accident situations. The other is the interior structure, referred to as containment internals, which provides biological shielding around the nuclear steam supply system. A plan view and vertical sections of the containment building are shown in Figure B.1-1.

The containment building (sometimes referred to as the reactor building) is in the shape of a right circular cylinder about 64 m in height and 43 m in diameter. It has a hemispherical dome and a flat base slab with a central cavity and instrumentation tunnel (see Figure B.1-1).

The containment is constructed of reinforced concrete prestressed by post-tensioned tendons in the cylinder walls and dome. The interior is lined with steel plates welded to form a leak-tight barrier. Penetrations provided for personnel, equipment, fuel transfer, piping, and electrical access are attached to the liner plate and are anchored into the concrete structure.

A 5.8-m-diameter equipment hatch provides access to the operating floor of the containment building from the east end of the auxiliary building. A 3.05-m-diameter personnel lock also penetrates the containment building at the operating floor level, providing entry access from the west end of the auxiliary building. A second personnel access lock (also 3.05-m-diameter) on the south side of the containment building allows grade-level entry from outdoors.





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

FIGURE B.1-1. Plan View and Vertical Sections of the PWR Containment Building

The containment internals consist of the reactor cavity, biological shield, steam-generator and pressurizer compartments, as well as the refueling cavity (see Figure B.1-1). Supports for equipment, operating decks, access stairways, and platforms are included in the conventionally reinforced concrete structure. Floor slabs and gratings are supported by structural steel beams.

The containment building is designed to house the nuclear steam supply system. Major components of this system include the reactor vessel and internals, four steam generators, four reactor coolant pumps, the pressurizer, and the reactor coolant piping. Placement of these components in the containment building is shown in Figure B.1-2. The components are described in the following sections.

B.1.1 Reactor Vessel and Internals

The reactor pressure vessel, shown in Figure B.1-3, is cylindrical, with a welded hemispherical bottom head and a removable, bolted, flanged and gasketed, hemispherical upper head. The bottom head contains penetration nozzles for connection and entry of the nuclear in-core instrumentation. Inlet and outlet nozzles for the primary coolant water are spaced evenly around the upper cylindrical portion of the vessel.

Internal surfaces of the vessel in contact with the primary coolant are weld overlaid with 3.96 mm, minimum, of stainless steel or Inconel.® The exterior of the reactor vessel is insulated with canned stainless steel reflective sheets contoured to completely enclose the vessel. All the insulation modules are removable, but access to vessel side insulation is limited by the surrounding concrete. Reactor vessel design parameters are shown in Table B.1-1.

The reactor vessel internals are shown in Figure B.1-4. The vessel internal structures support and constrain the fuel assemblies, direct coolant flow, guide in-core instrumentation, and provide some neutron shielding. The

®Registered trademark of Huntington Alloys, Inc.

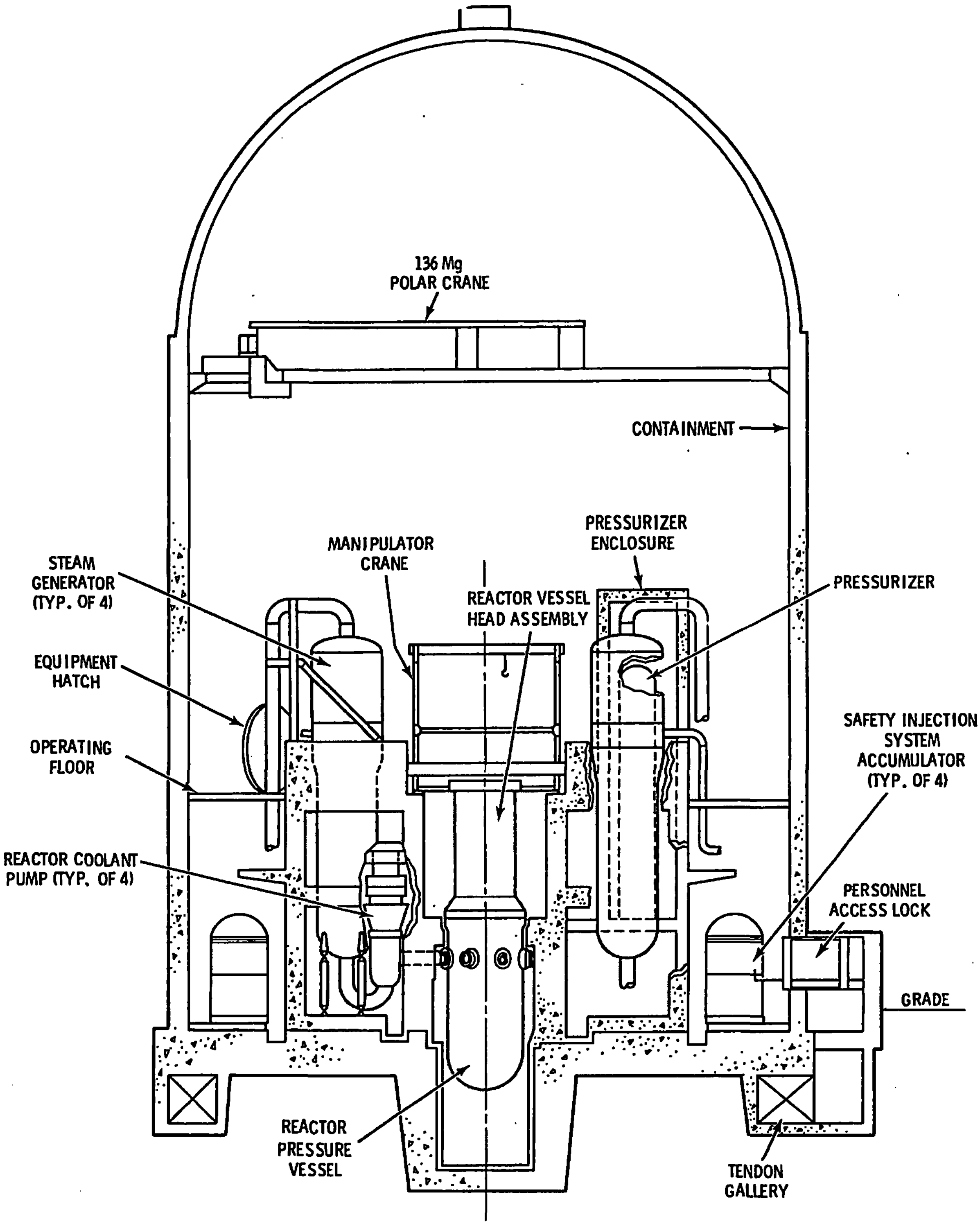


FIGURE B.1-2. Vertical Section of the PWR Containment Building, Looking East

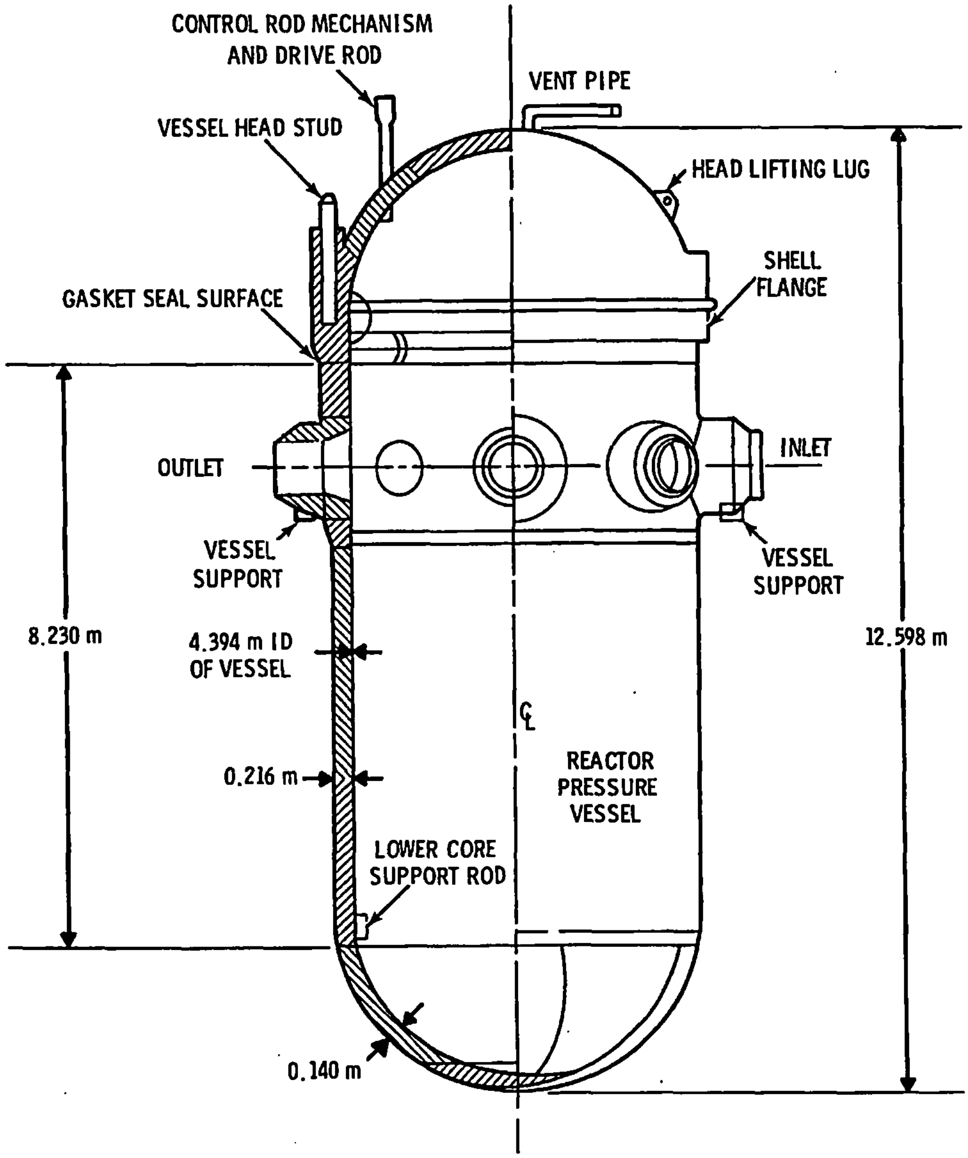


FIGURE B.1-3. PWR Reactor Pressure Vessel

components of the reactor internals are divided into three parts consisting of the lower core support structure, the upper core support structure, and the in-core instrumentation support structure.

The major containment and support member of the reactor internals is the lower core support structure. This assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the neutron shield pods, and the core support that is welded to the core barrel. The lower core

TABLE B.1-1. PWR Reactor Vessel Design Parameters

Parameter	Design Specification
Overall Height of Vessel and Closure Head (bottom head OD to top of control rod mechanism adapter)	13.36 m
Thickness of Insulation, Minimum	76 mm
Number of Reactor Closure Head Studs	54
ID at Shell	4.394 m
Inlet Nozzle ID	0.699 m
Outlet Nozzle ID	0.737 m
Lower Head Thickness, Minimum	0.136 m
Vessel Belt-Line Thickness, Minimum	0.216 m
Closure Head Thickness	0.165 m
Mass Without Head and Insulation	308.4 Mg
Mass of Head Without Insulation Including Stud Nuts and Washers	88.5 Mg

support structure provides passageways and control for the coolant flow. The lower core plate provides support and orientation for the fuel assemblies. The lower core support structure is supported at its upper flange from a ledge .n the reactor vessel flange, and its lower end is restrained from transverse motion by a radial support system attached to the vessel wall.

The upper core support structure consists of the upper core support assembly and the upper core plate between which are contained support columns and guide tube assemblies. The support columns connect and establish the spacing between the upper support assembly and the upper core plate. The guide tube assemblies shield and guide the control rod drive shafts and control rods. The upper core support structure, which can be removed as a unit, is properly positioned with respect to the lower support structure by slots in the upper core plate which engage flat-sided alignment pins welded to the core barrel.

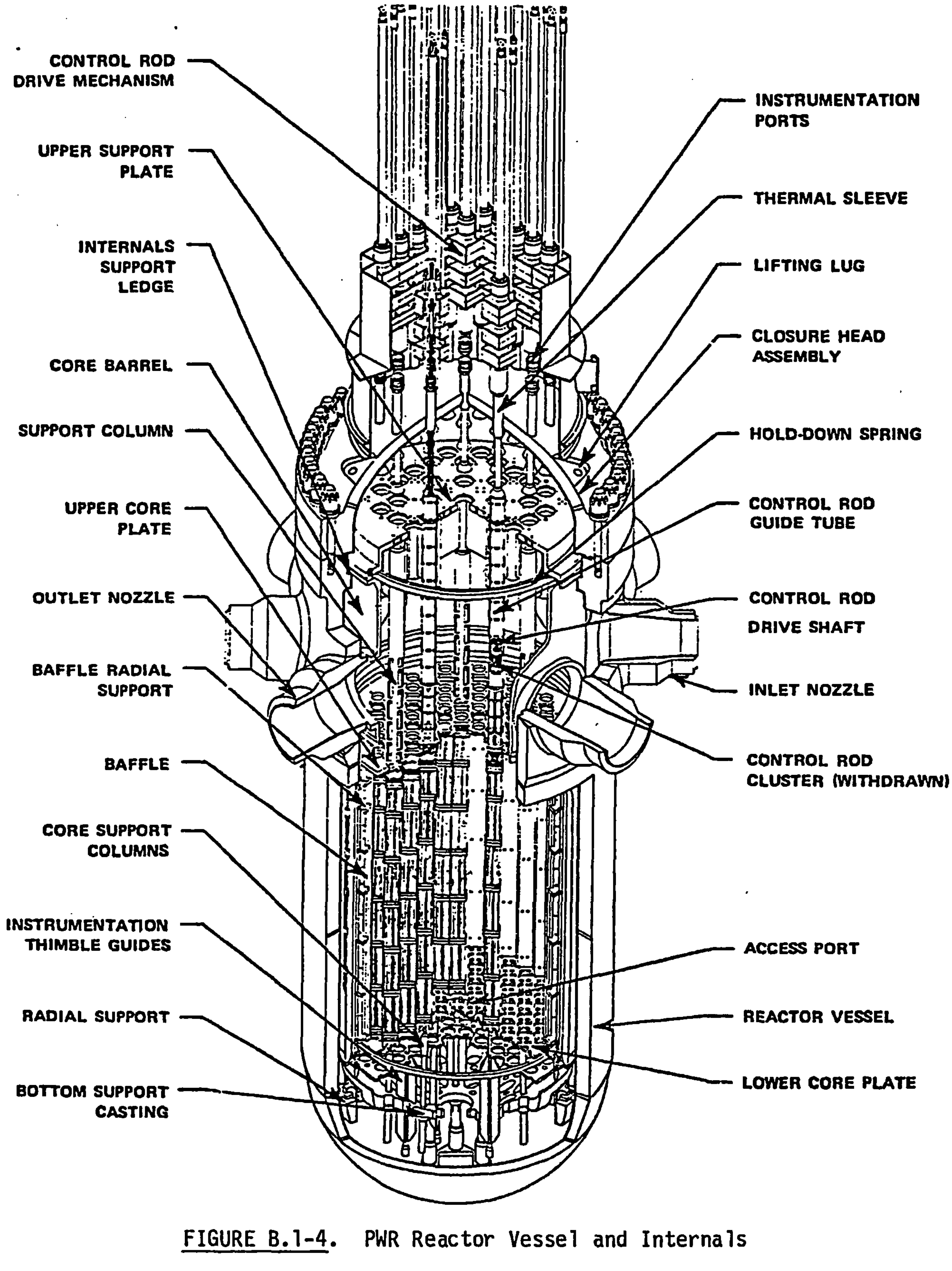


FIGURE B.1-4. PWR Reactor Vessel and Internals

The in-core instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom.

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

B.1.2 Reactor Coolant System

The reactor coolant system (RCS) for the reference PWR consists of four loops for transferring heat from the reactor to the secondary coolant system and a pressurizer for maintaining coolant pressure. Each loop contains a steam generator, a coolant pump, and connecting piping. The RCS contains approximately 380 m^3 of primary coolant water. The components of the RCS are discussed separately in the following paragraphs.

Steam Generators

The steam generators are the largest components of the reactor coolant loops. As illustrated in Figure B.1-5, each 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.

Reactor coolant flows through the inverted U-tubes, entering and leaving through nozzles in the hemispherical bottom head of the steam generator. The head is divided into inlet and outlet chambers by a vertical partition plate; manways are provided for access to both chambers. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel. The steam drum has two bolted and gasketed access openings for inspection and maintenance of the moisture separators, which can be disassembled and removed through the openings.

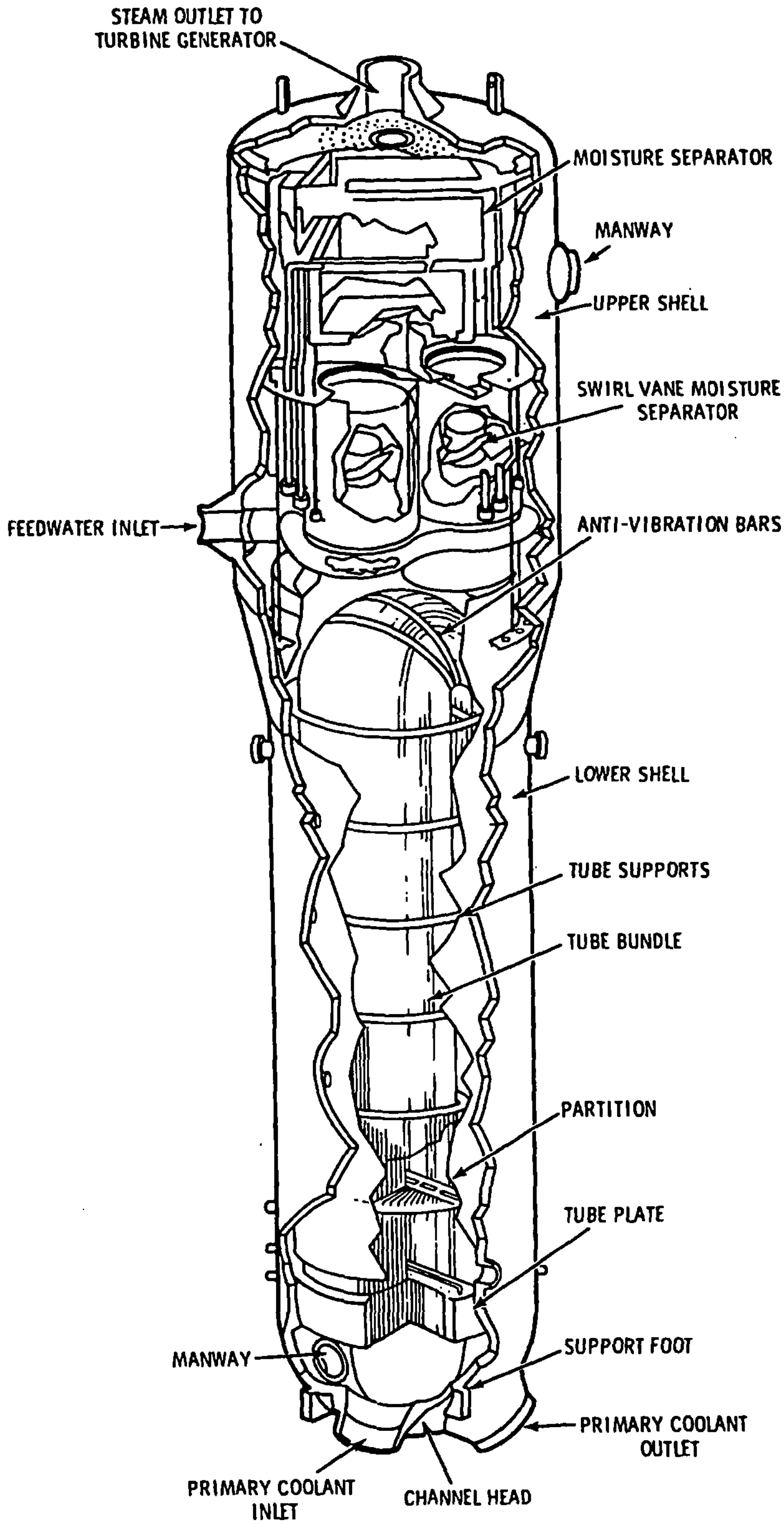


FIGURE B.1-5. PWR Steam Generator

The units are primarily carbon steel. The heat transfer tubes and the divider plate are Inconel and the interior surfaces of the reactor coolant channels and nozzles are clad with austenitic stainless steel. The primary side of the tube sheet is weld clad with Inconel.

Reactor Coolant Pumps

The reactor coolant pumps are vertical, single-stage, centrifugal, shaft-seal units designed to pump large volumes of primary coolant at high temperatures and pressures. Reactor coolant pump data are given in Table B.1-2.

TABLE B.1-2. PWR Reactor Coolant Pump Data

Parameter	Design Specification
Capacity per pump	335 m^3/min
Developed head	84.4 m of water
Overall unit height	8.702 m
Water Volume	1.586 m^3
Mass, dry	85.37 Mg

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

Pressurizer

The pressurizer, a vertical, cylindrical vessel with hemispherical top and bottom heads, is shown in Figure B.1-6. It is constructed of carbon steel with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant.

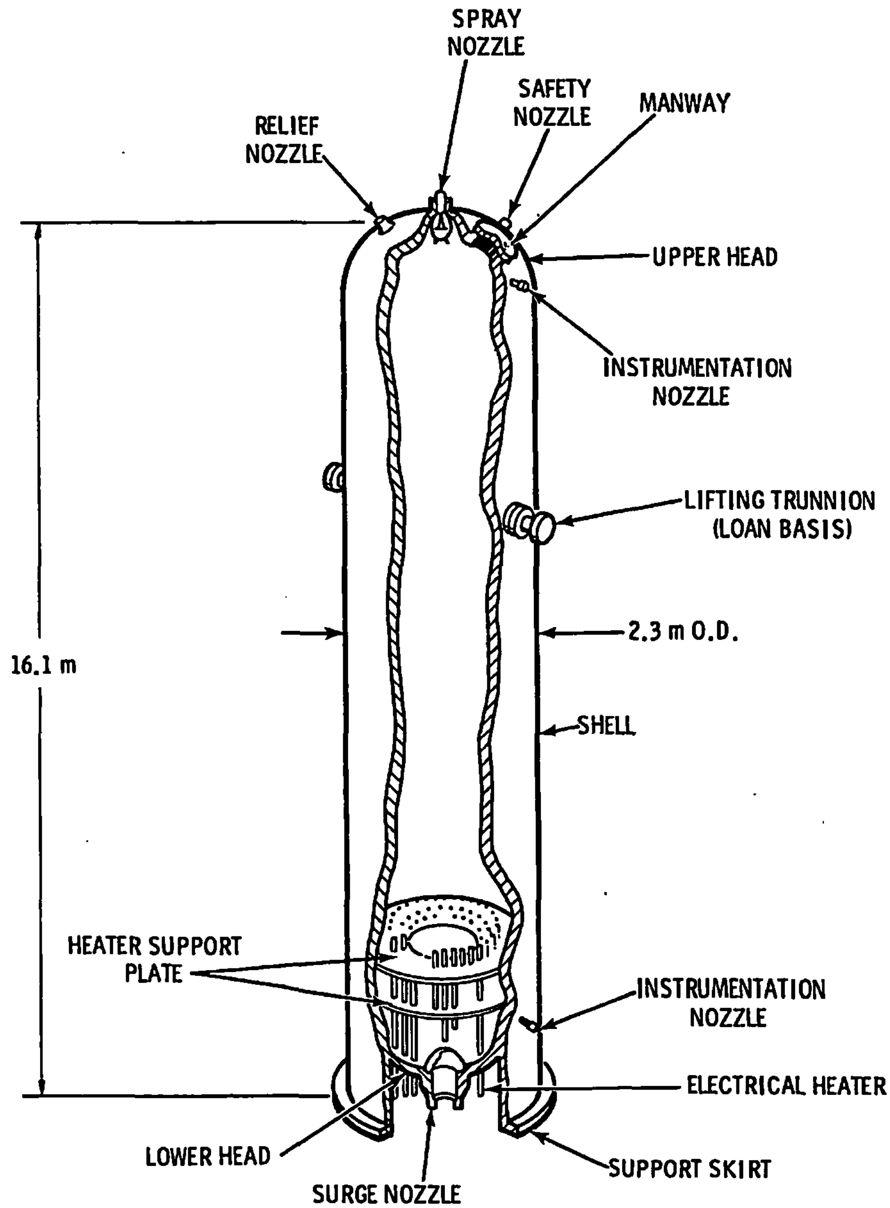


FIGURE B.1-6. PWR Pressurizer

The surge line nozzle and removable electric heaters are installed in the bottom head. The pressurizer surge line connects the pressurizer to one reactor hot leg, allowing continuous RCS pressure adjustments. A thermal sleeve is provided to minimize stresses in the surge line nozzle.

Reactor Coolant Piping

Principal design data for reactor coolant system piping are given in Table B.1-3. Major system piping data are given in Table B.1-4.

TABLE B.1-3. PWR Reactor Coolant Piping Design Parameters

Parameter	Design Specification
Reactor inlet piping, ID	0.699 m
Reactor inlet piping, nominal wall thickness	58.9 mm
Reactor outlet piping, ID	0.737 m
Reactor outlet piping, nominal wall thickness	62.0 mm
Coolant pump suction piping, ID	0.737 m
Coolant pump suction piping, nominal wall thickness	65.8 mm
Pressurizer surge line piping, nominal pipe size	0.356 m
Pressurizer surge line piping, nominal wall thickness	35.7 mm

TABLE B.1-4. PWR Major System Piping Data

System	Size (m, OD)	Length (m)	Mass (kg)
Reactor Coolant Piping	0.745 to 0.846	81	100 698
Reactor Coolant Piping	0.051 to 0.356	677	11 793
Residual Heat Removal	0.051 to 0.356	194	11 929
Chem. & Volume Control	0.051 to 0.203	2 103	17 763
Emergency Core Cooling	0.051 to 0.610	1 923	100 698
Containment Spray	0.051 to 0.356	1 210	66 224
Auxiliary Feedwater	0.051 to 0.254	604	14 361
Spent Fuel Pool Cooling	0.051 to 0.152	594	3 783
Condensate Facility	0.051 to 0.203	518	38 782
Station Service	0.051 to 0.508	1 779	59 874
Service Cooling	0.051 to 0.457	427	25 107
Component Cooling	0.051 to 0.610	4 887	168 736
Makeup Water System	0.051 to 0.152	998	5 625

B.1.3 Engineered Safety Systems

The reference PWR is equipped with engineered safety systems that are activated during certain accident conditions. Two of these systems, the emergency core cooling system and the containment spray system, are of particular importance to this study and are described briefly in the following paragraphs.

Emergency Core Cooling System

The primary function of the emergency core cooling system (ECCS) is to remove the stored and fission-product-decay heat from the reactor core following a loss-of-coolant accident (LOCA) such that fuel rod damage, to the extent that it would impair effective core cooling, is prevented. The ECCS also provides shutdown capability by means of boron injection. The ECCS is shown schematically in Figure B.1-7. Major components of the system include the four safety system injection accumulators, the boron injection tank, the refueling water storage tank,[a] and the residual heat removal heat exchangers.

Containment Spray System

The containment spray system (CSS) is designed to spray cool water into the containment building atmosphere in the event of a LOCA to suppress temperature and pressure transients. In addition, sodium hydroxide solution is introduced with the spray to remove fission product iodine from the building atmosphere and retain it in the recirculation sump water, to prevent its release from the containment building.

The CSS consists of two separate and independent trains of equal capacity ($\sim$10.5 m^3/min each). Each train includes a pump, spray ring header, spray additive eductor, isolation valves, and the necessary piping, instrumentation and controls. The trains are supplied with spray additive from a common

---

(a) In addition to its usual service of supplying borated water during refueling operations, this tank provides borated water to the ECCS and the containment spray system during a LOCA. The 1500-m^3 tank is located outdoors adjacent to the containment building (see Section B.1.4).

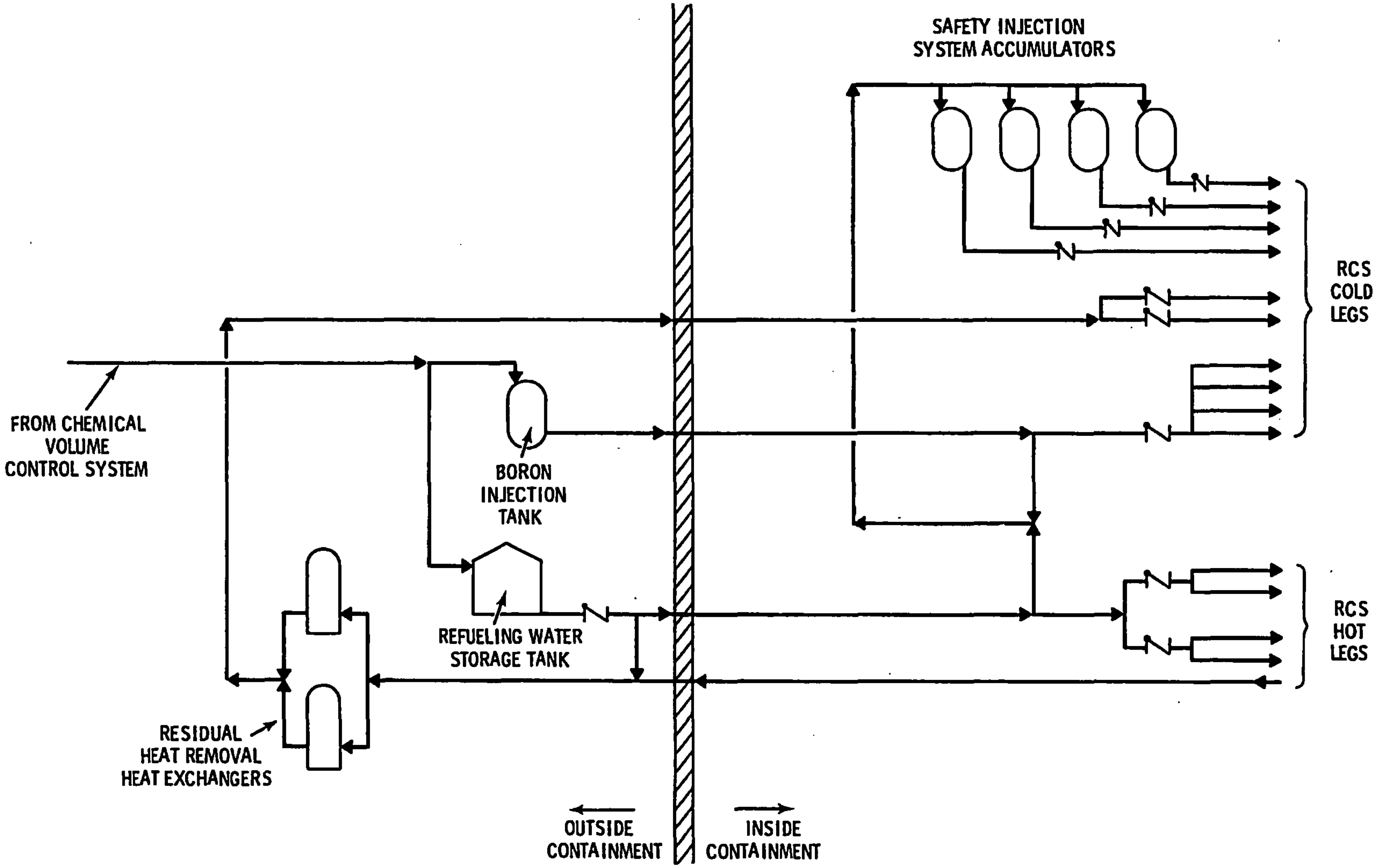


FIGURE B.1-7. PWR Emergency Core Cooling System Schematic sodium hydroxide tank and with water from the refueling water storage tank. When the refueling water storage tank is exhausted, spray pump suction can be manually shifted to the containment recirculation sump.

B.1.4. Reactor Defueling Equipment and Procedures

The equipment and procedures used for normal defueling operations at the reference PWR are described in this subsection. The information presented here pertains only to defueling activities in the containment building; related spent fuel handling and storage activities in the fuel building are discussed in Section B.2.5.

B.1.4.1 Defueling Equipment

The major containment building equipment items used during reactor defueling are described briefly in the following paragraphs.

Manipulator Crane. The manipulator crane is a bridge-and-trolley crane that spans the refueling cavity and runs on rails set into either side of the refueling cavity. A telescoping vertical mast extends down from the crane into the refueling water. A pneumatic gripper on the end of the mast is lowered to grip a fuel assembly. A winch mounted on the trolley raises the fuel assembly up into the mast. The fuel assembly is transported to its new position while inside the mast. All controls for the manipulator crane are mounted on a console on the trolley.

Fuel Transfer System. The fuel transfer system consists of an underwater transfer car that runs on tracks extending from the refueling canal through the transfer tube and into the spent fuel pool in the fuel building, with an upender lifting frame at each end of the transfer tube. The manipulator crane is used to place a fuel assembly vertically in the upender in the refueling canal. The fuel assembly is then lowered to a horizontal position for passage through the transfer tube into the fuel building.

Rod Cluster Control Changing Fixture. Rod cluster control (RCC) elements are removed from spent fuel assemblies in the RCC changing fixture. A frame and track structure supports a movable carriage with three compartments, two for holding individual fuel assemblies and a third for supporting a single RCC element. A guide tube, mounted on the refueling canal wall above the carriage, positions and guides the gripper and RCC element during raising and lowering. Two flexible fingers on the pneumatically operated gripper engage the top of the RCC element for removal from the fuel assembly. A drive mechanism mounted on the operating deck controls the position of the carriage and the elevation of the gripper.

Reactor Vessel Head Lifting Device. This device is a structural steel frame with suitable rigging to enable the reactor vessel head to be removed from the vessel using the polar crane. The lifting device is permanently attached to the head.

Reactor Internals Lifting Device. The reactor internals lifting device is a structural frame suspended from the polar crane. The frame is lowered onto the guide tube support plate of the internals and is then manually bolted to the plate with three bolts. Bushings on the frame engage guide studs in the vessel flange to provide guidance during removal of the internals.

Refueling Water Storage Tank. This 1500-m^3 tank, located outdoors adjacent to the containment building, supplies borated water to fill the refueling cavity during defueling operations. The vertical, cylindrical tank contains sufficient water to provide a level in the refueling cavity adequate to ensure occupational radiation safety and fuel assembly cooling during defueling operations.

B.1.4.2 Defueling Procedures

Defueling procedures are divided into three major phases: preparation, reactor disassembly, and fuel handling. A general description of each of these phases is presented in the following paragraphs.

Preparation. The reactor is shut down and cooled to cold shutdown conditions. Following a radiation survey, the containment is entered. The coolant level in the reactor vessel is lowered to slightly below the vessel flange. The defueling equipment is then checked for proper operation.

Reactor Disassembly. All cables, air ducts, and insulation are removed from the vessel head. The refueling cavity is then prepared for flooding, after which the vessel head is unseated and raised 0.3 m above the vessel flange. Water from the refueling water storage tank is pumped into the reactor coolant system and overflows into the refueling cavity. The vessel head is raised to keep it just above the water level until the water reaches a safe shielding depth, when the vessel head is placed on its storage pedestal. The control rod drive shafts are disconnected and removed from the vessel with the upper internals to free the fuel assemblies and rod cluster control assemblies from obstruction.

Fuel Handling. The defueling sequence is started with the manipulator crane. Spent fuel assemblies are removed from the core in a predetermined order to ensure the safety of the defueling operation. The general fuel handling sequence is as follows:

- The manipulator crane is placed over the fuel assembly to be removed.
- The fuel assembly is lifted to a predetermined height to clear the reactor vessel but still maintain sufficient water depth for radiation shielding.
- If the fuel assembly contains a rod cluster control, the assembly is placed in the changing fixture where the rod cluster control is removed.
- The fuel transfer car is positioned in the refueling cavity and the fuel assembly container is pivoted to the vertical position by the upender.
- The manipulator crane is moved to line up the fuel assembly with the fuel transfer system.
- The fuel assembly is loaded into the fuel assembly container which is then pivoted to the horizontal position by the upender.
- The fuel assembly container is moved by the transfer car through the transfer tunnel to the spent fuel pool in the fuel building.

Spent fuel handling and storage procedures in the fuel building are discussed in Section B.2.5.

B.1.5 Radionuclide Containment Systems

The containment system encloses the reactor and the reactor coolant system, primarily for the protection of public safety. Its ability to provide an effective barrier to confine potential releases of radioactivity depends upon maintaining leaktightness within specific bounds.

The containment system includes the containment building and associated ventilation and exhaust system, shown schematically in Figure B.1-8. Portions of the ventilation and exhaust systems that may be directly concerned with facility decommissioning are described in this section.

The purge supply system (CS-1 in Figure B.1-8) and the purge exhaust and refueling cavity supply and exhaust system (CS-2 in Figure B.1-8) are designed to provide 1416 m^3/min, or one complete containment air change every 40 minutes. The purge exhaust passes through HEPA filters to remove airborne contaminants. The systems are operated before and during personnel occupancy of the containment.

The purge supply system (CS-1) consists of an outside air intake, a roll-type prefilter, a bank of HEPA filters, two 1416-m^3/min fans arranged in parallel, anti-backdraft dampers, a containment penetration, quick-closing inboard and outboard isolation valves, and ductwork. The fans and filters are located outside of containment.

The purge exhaust and refueling cavity supply and exhaust system (CS-2) consists of a roll-type prefilter, a bank of HEPA filters, two 1416-m^3/min fans arranged in parallel, anti-backdraft dampers, a containment penetration, quick-closing inboard and outboard isolation valves, and ductwork. The fans and filters are located outside of the containment. The system exhausts to the containment purge vent at the top of the containment building.

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

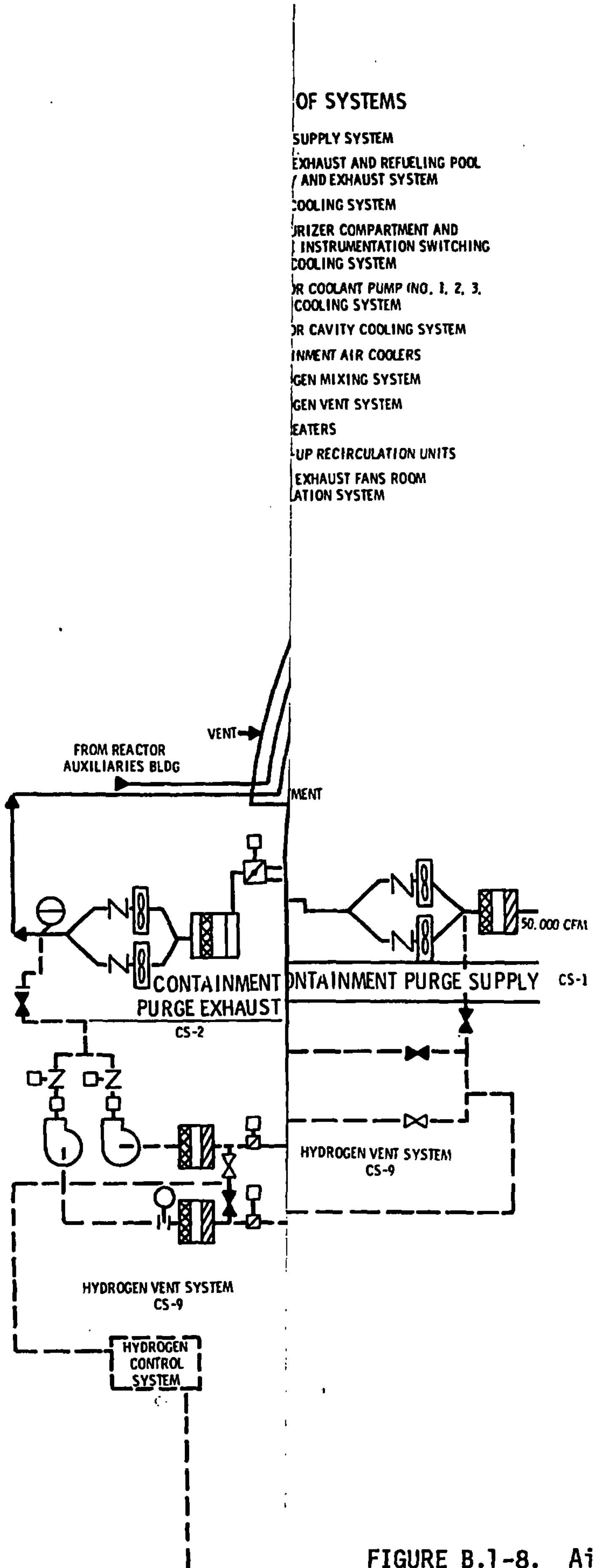


FIGURE B.1-8. Air Flow Diagram for PWR Containment Building

The refueling cavity supply consists of two fans which draw approximately 227 m^3/min of air from the containment atmosphere and discharge it horizontally across the refueling cavity. The refueling cavity exhaust consists of two fans which draw approximately 425 m^3/min of air from inlets at the surface of the refueling cavity and discharge to the purge exhaust system (CS-2).

B.1.6 Containment Building Major Equipment

Table B.1-5 contains a list of major equipment items located in the containment building that are not part of the nuclear steam supply system.

TABLE B.1-5. Equipment List: PWR Containment Building

Equipment Piece (quantity)	Mass (each) kg	Overall Dimensions (each), m
Regenerative Heat Exchanger (1)	2 994	0.36 dia x 5.49
Excess Letdown Heat Exchanger (1)	726	0.28 dia x 3.35
Containment Sump Pumps (2)	635	1.83 long
Reactor Cavity Drain Pump (1)	363	4.57 long
Pressurizer Relief Tank (1)	12 338	3.25 dia x 8.25
Safety Inj. Sys. Accumulator (4)	34 700	3.66 dia x 6.4

B.2 AUXILIARY AND FUEL BUILDINGS

The auxiliary and fuel buildings house the solid and liquid radwaste systems, new and spent fuel handling and storage facilities, off-gas treatment equipment, and ventilating and air conditioning equipment. Plan views and vertical sections of these buildings are shown in Figures B.2-1 through B.2-8.

B.2.1 The Auxiliary Building

The auxiliary building is a steel-frame and reinforced-concrete structure with two floors below grade and four floors above grade. It is approximately

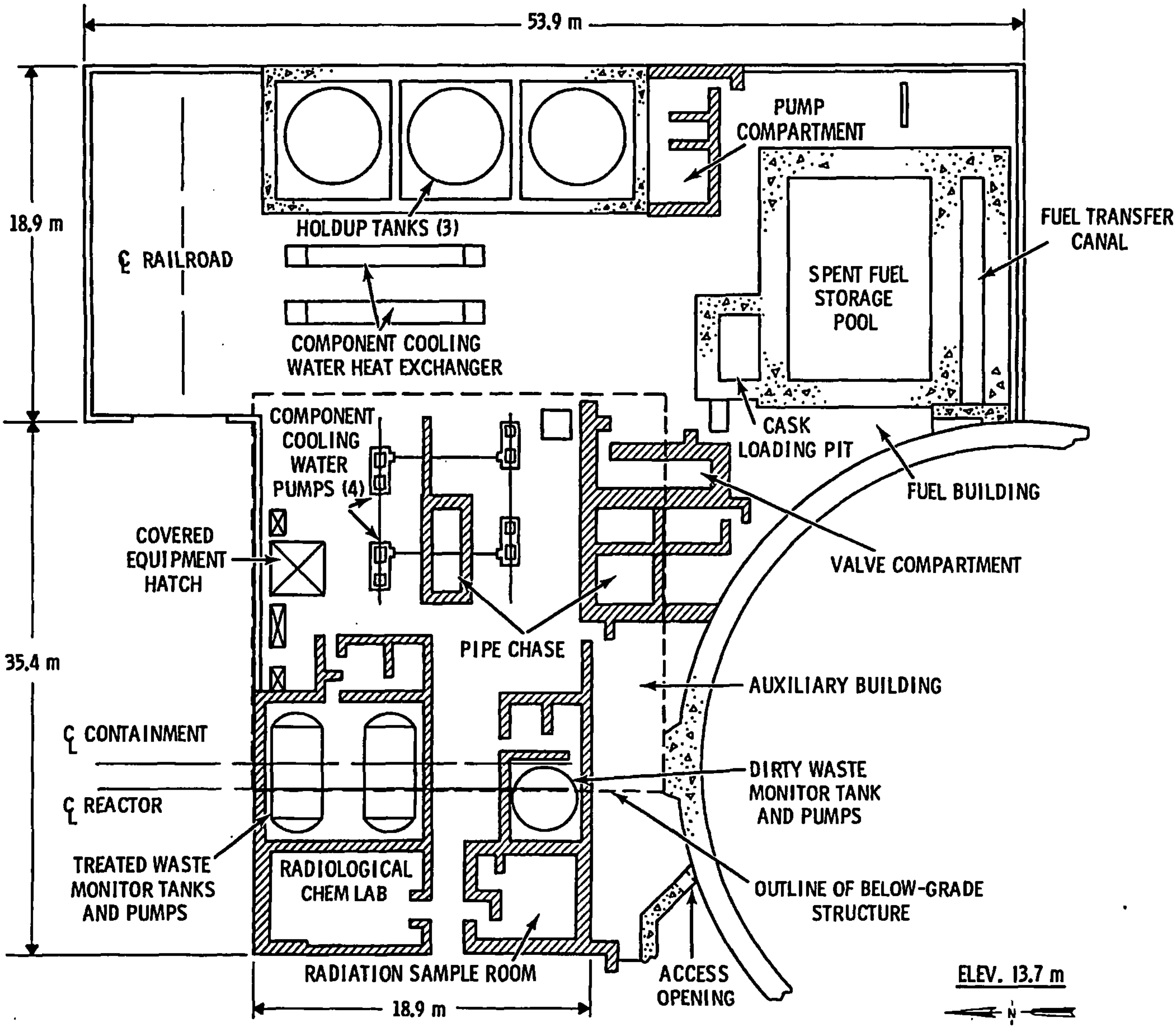


FIGURE B.2-1. Plan Views--PWR Fuel and Auxiliary Buildings, Grade Level

30 m in overall height and has lateral dimensions of about 35 m by 19 m. The exterior walls are concrete block masonry at the first floor and metal siding above, with the below grade portion of reinforced concrete. The interior walls are constructed of concrete block masonry. To facilitate equipment removal, certain interior walls are provided with removable panels. The auxiliary building is separated from the containment building by a 76-mm expansion joint.

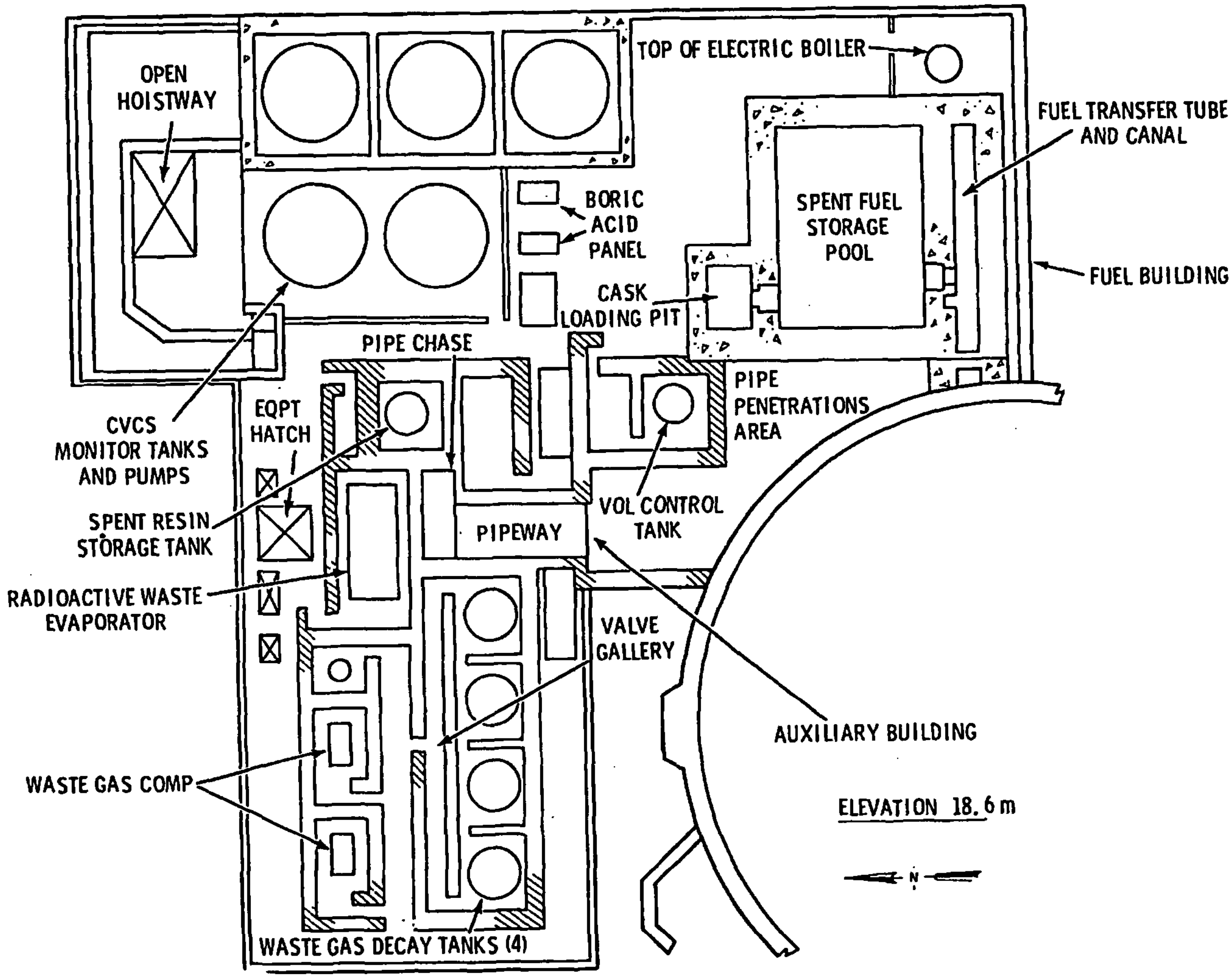


FIGURE B.2-2. Plan Views--PWR Fuel and Auxiliary Buildings, Elevation 18.6 m

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, equipment for the emergency core cooling system and the containment spray system, and the ventilation equipment for the containment, fuel, and auxiliary buildings.

B.2.2 Fuel Building

The fuel building is a steel-frame and reinforced-concrete structure with four floors above grade. It is approximately 27 m in height and has lateral dimensions of about 54 m by 19 m. Exterior walls to an elevation of 19.2 m

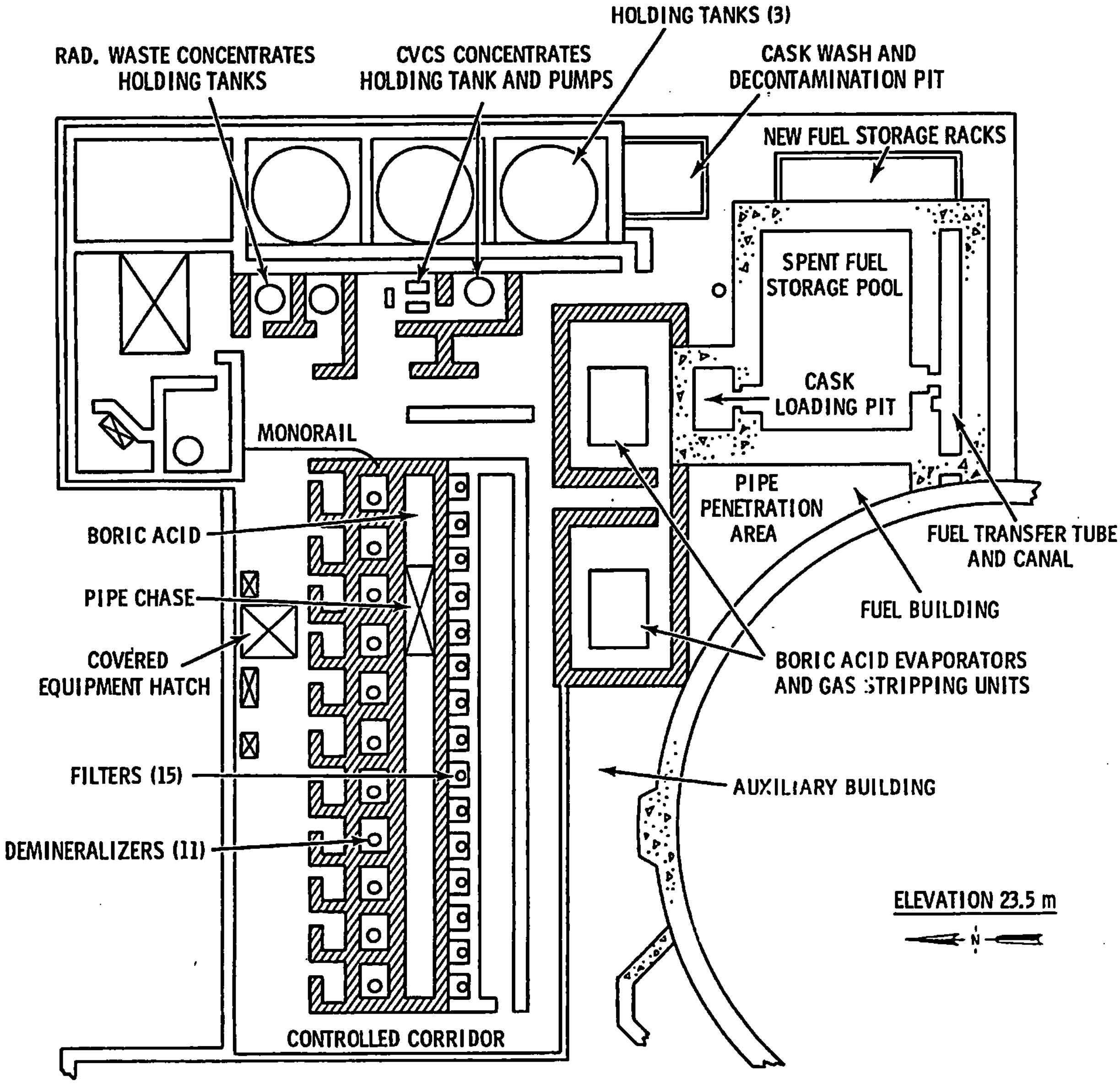


FIGURE B.2-3. Plan Views--PWR Fuel and Auxiliary Buildings, Elevation 23.5 m

are precast concrete panels with metal siding above. Interior walls are concrete block masonry. At the west side, the fuel building is structurally connected to the auxiliary building.

The fuel building houses the spent fuel storage pool. The pool walls and base slab are constructed of thick (1.57 m to 2.0 m) reinforced concrete, and the inside faces are lined with a 6.4-mm-thick stainless steel liner to provide

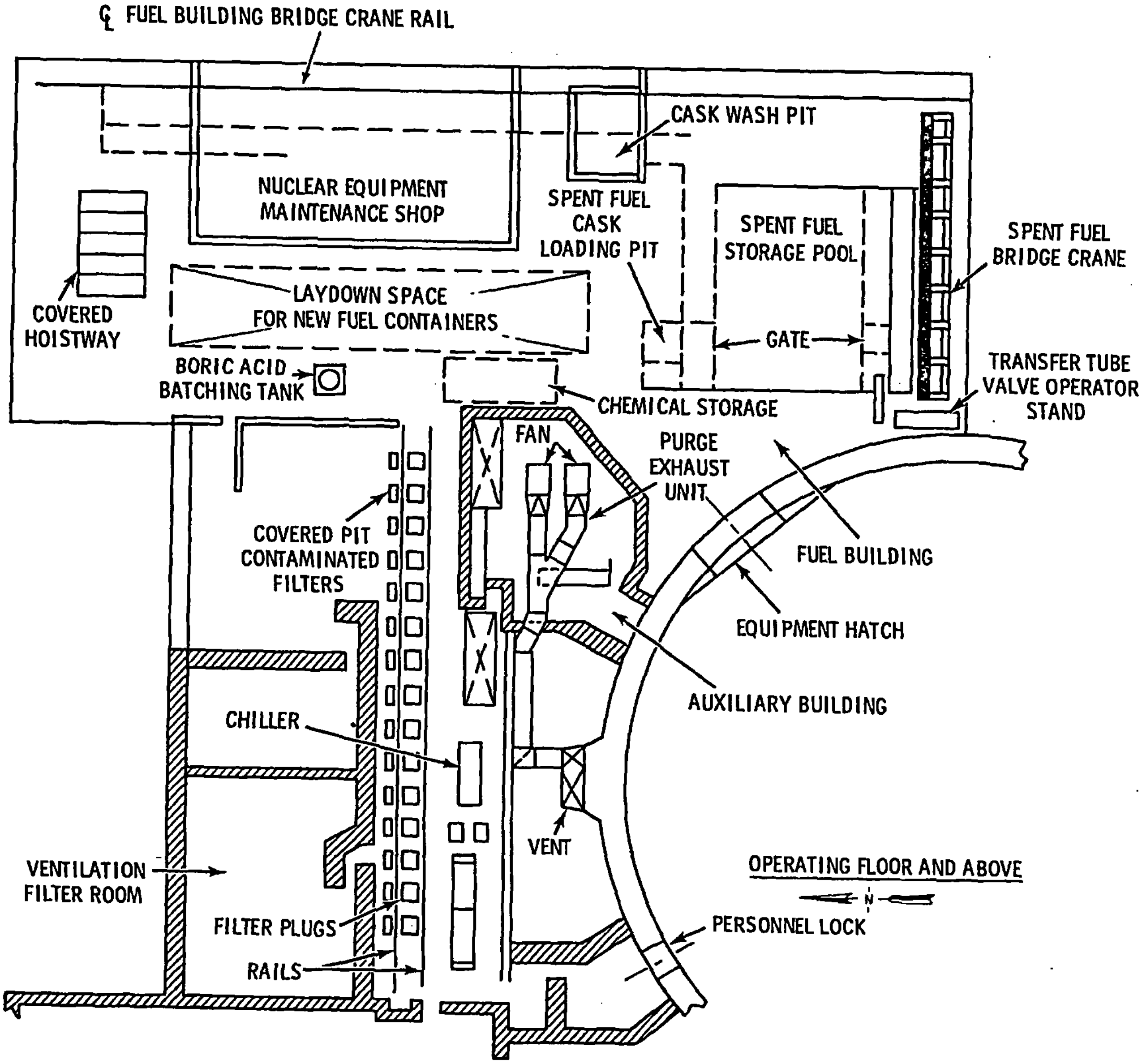


FIGURE B.2-4. Plan Views--PWR Fuel and Auxiliary Buildings, Operating Floor

leaktightness. Expansion joint bellows at the fuel transfer tube provide for the relative movement between containment, containment internals, and the spent fuel pool.

A 113-Mg overhead crane capable of handling the fuel cask runs on rails mounted on the operating floor. The ground floor encloses a rail access at

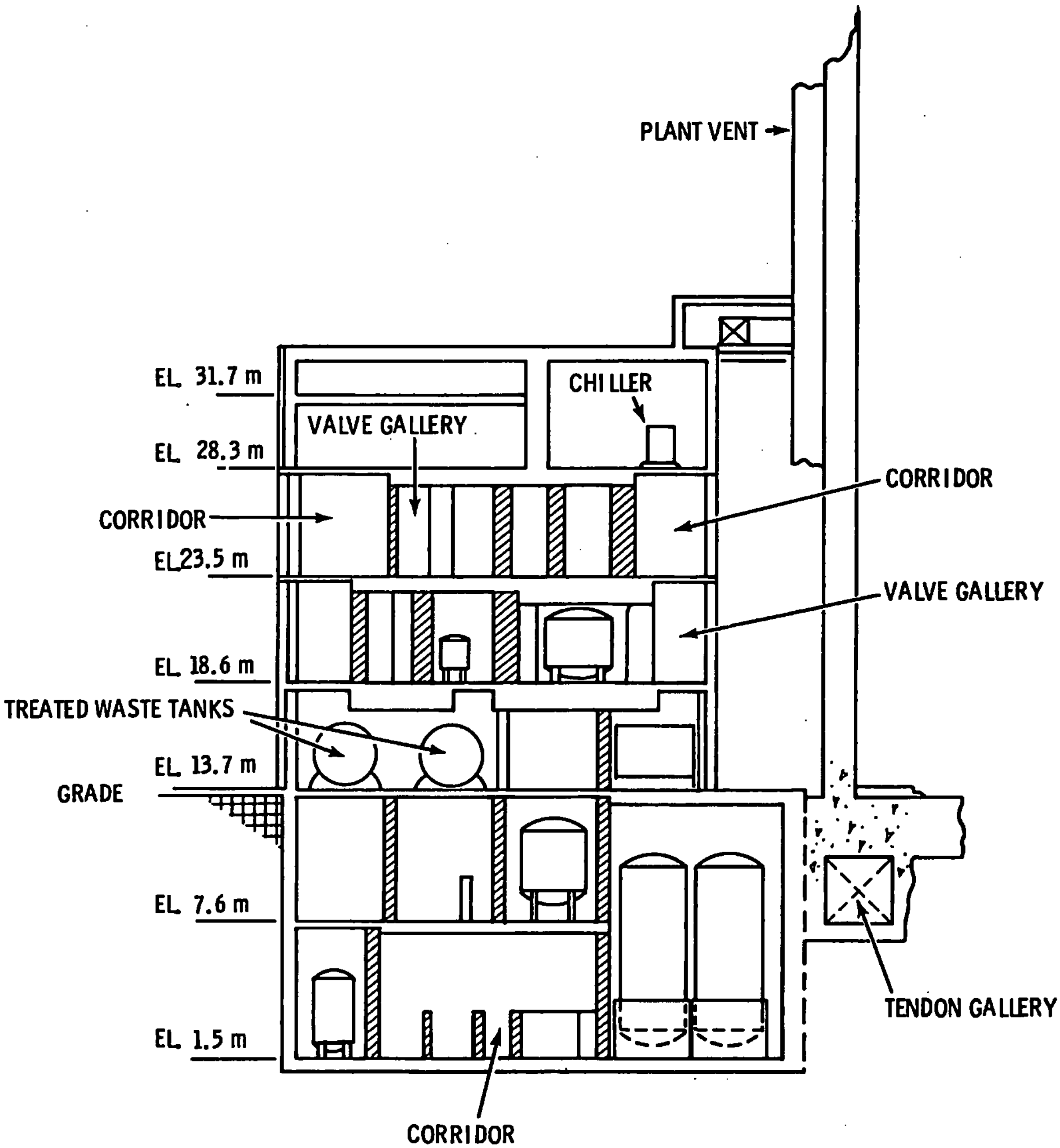


FIGURE B.2-5. Vertical Section of PWR Auxiliary Building, Looking East

one end and a hatchway up through the operating floor. The building also includes the following structural features:

- cask loading pit
- new fuel storage pit
- cask wash pit

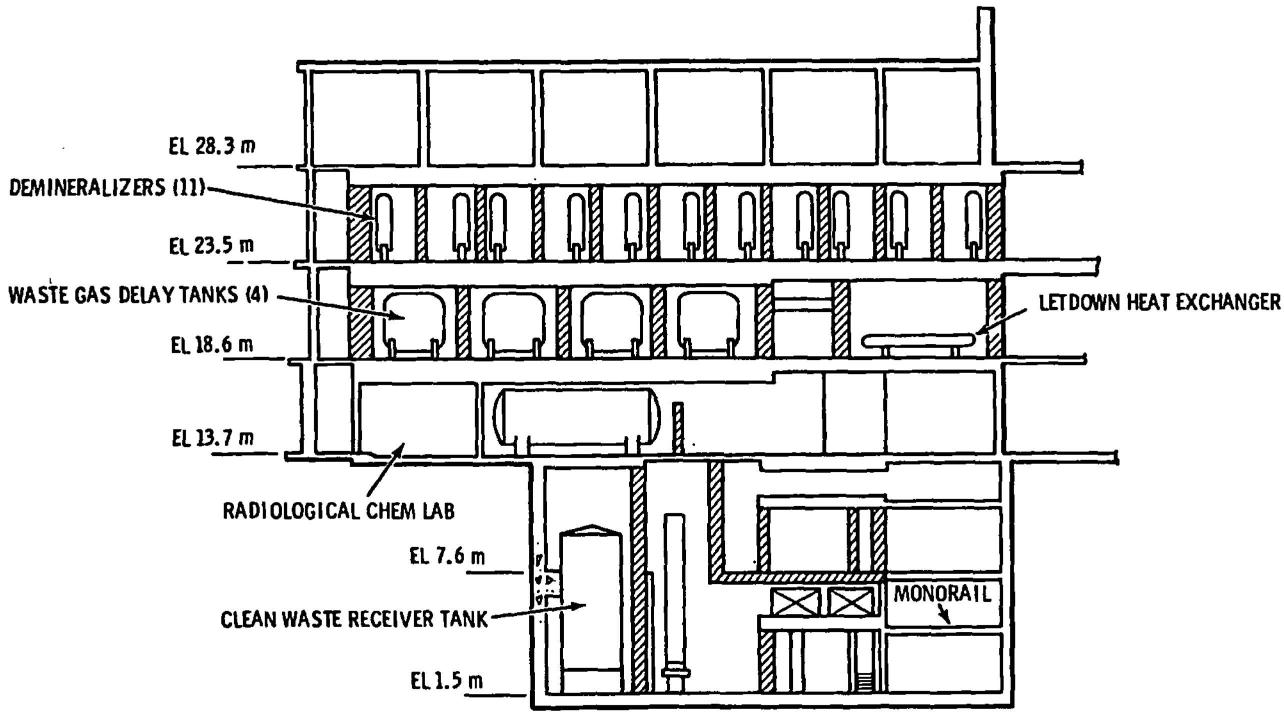


FIGURE B.2-6. Vertical Section of PWR Auxiliary Building, Looking North

- reinforced concrete vaults with 0.76-m-thick walls to enclose the three chemical volume control system (CVCS) holdup tanks.

The fuel building also contains the makeup water treatment system (i.e., the CVCS) and the solid radioactive waste handling equipment.

B.2.3 Chemical Volume Control System

The CVCS is designed to provide the following services to the reactor coolant system (RCS):

- maintain required water inventory in the RCS
- maintain seal-water injection flow to the reactor coolant pumps
- control water chemistry, activity level, soluble chemical neutron absorber concentration and makeup
- process effluent reactor coolant for recovery and reuse of soluble chemical neutron absorber and makeup water.

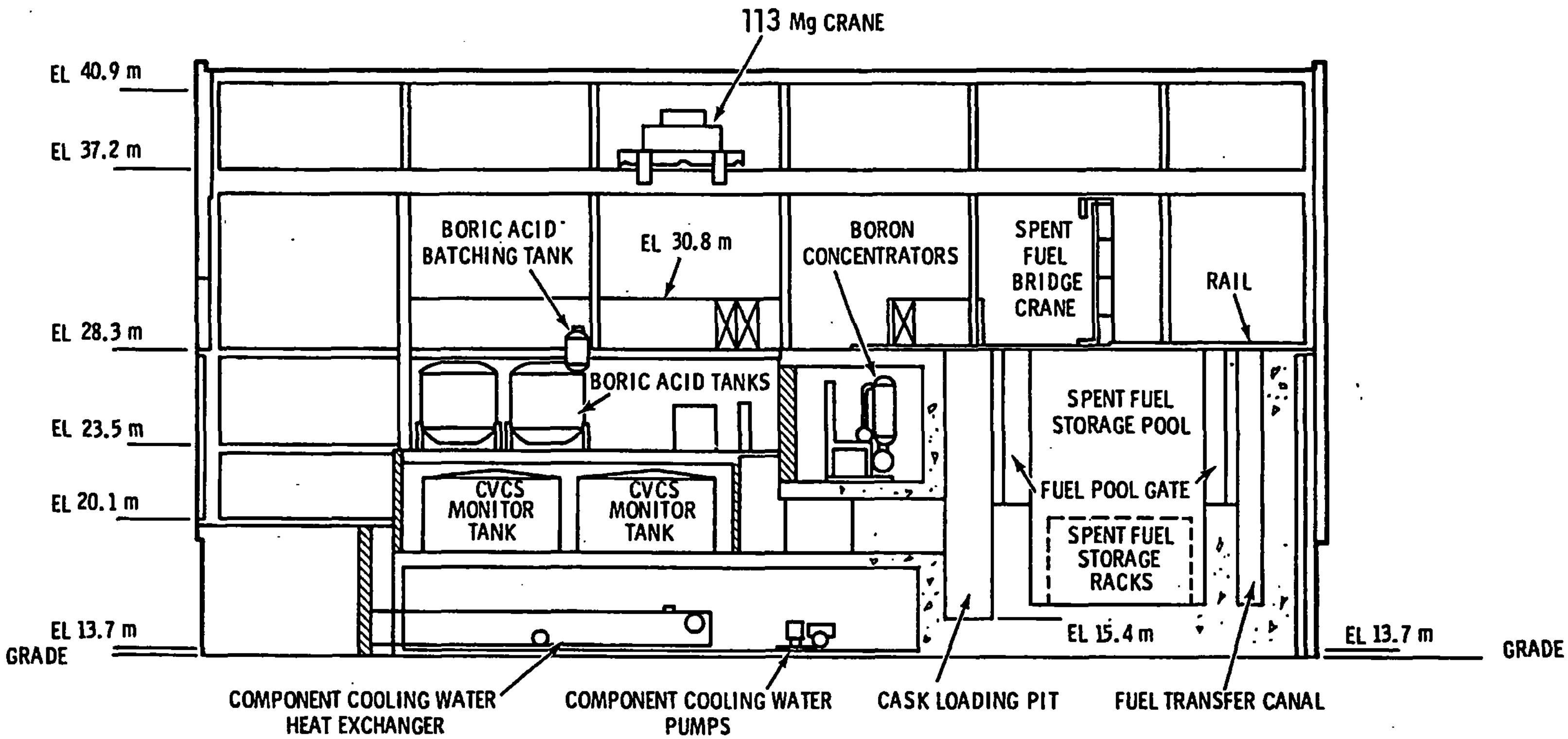


FIGURE B.2-7. Vertical Section of PWR Fuel Building, Looking East

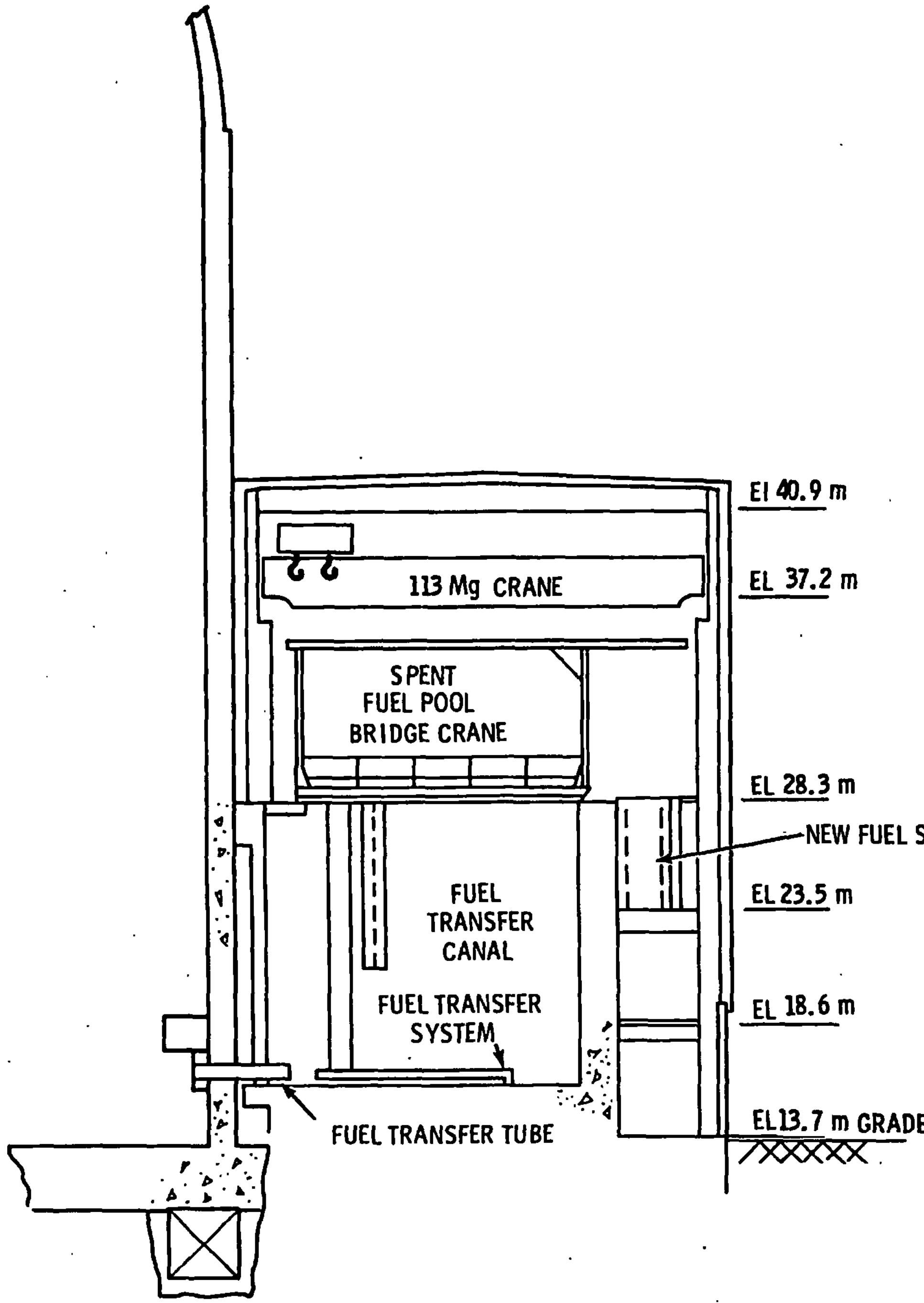


FIGURE B.2-8. Vertical Section of PWR Fuel Building, Looking North

In addition, the CVCS is shared in part with the emergency core cooling system (see Section B.1.3). The CVCS is shown schematically in Figure B.2-9.

The CVCS consists of several subsystems:

- charging, letdown, and seal water
- chemical control, purification, and makeup
- boron recovery.

These subsystems are discussed separately in the following paragraphs.

Charging, Letdown, and Seal Water

The charging and letdown capabilities of the CVCS maintain a programmed water level in the pressurizer by means of a continuous, automatically controlled feed-and-bleed process. Seal water for the reactor coolant pumps is provided by diverting a portion of the charging flow; the seal water system is not shown in Figure B.2-9 for simplicity. Letdown flow is normally 17.0 m^3/hr, with a maximum of 27.3 m^3/hr. Charging flow rates are 12.5 m^3/hr normally and 22.7 m^3/hr maximum. Seal water supply and return flows are normally 7.3 m^3/hr and 2.7 m^3/hr, respectively.

Reactor coolant is bled from upstream of the reactor coolant pump and flows through the regenerative heat exchanger, passing heat to the charging flow returning to the reactor coolant loop. After this cooling, the coolant passes through letdown orifices to reduce pressure, after which it is further cooled in the letdown heat exchanger. The coolant is then purified in one of two mixed-bed demineralizers and, if further purification is required, through the cation-bed demineralizer. The design flow rates through each mixed-bed demineralizer and through the cation-bed demineralizer are 27.3 m^3/hr and 16.4 m^3/hr, respectively. The coolant is then filtered and sprayed into the volume control tank, which contains a regulated hydrogen atmosphere to control hydrogen concentration in the coolant.

The charging pumps normally take suction from the volume control tank to provide reactor coolant feed, which is heated in the regenerative heat exchanger before returning to the RCS. Feed can also be provided from the

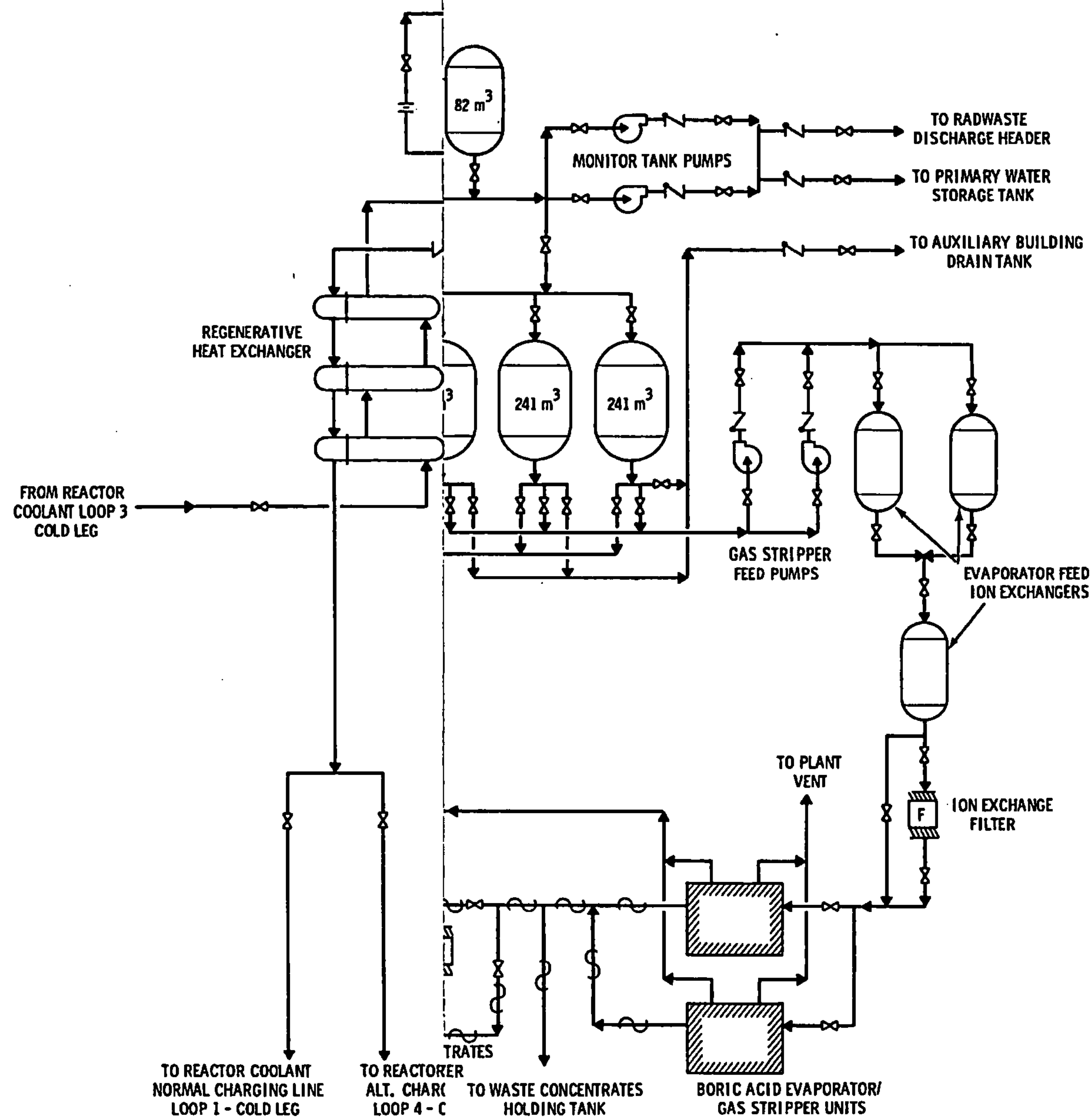


FIGURE B.2-9. PWR Chemical Volume Control System Schematic primary makeup water system and from the boron recovery section of the CVCS. The charging pumps are also used during a LOCA to inject borated water into the RCS from the boron injection tank (see Section B.1.3).

The volume control tank provides surge capacity for reactor coolant expansion not accommodated by the pressurizer. If the water level in the volume control tank exceeds set limits, a three-way valve downstream of the reactor coolant filter automatically diverts some of the letdown to the CVCS holdup tanks.

Chemical Control, Purification, and Makeup

Chemical control of the reactor coolant water consists mainly of pH control and oxygen control. Lithium hydroxide solution is introduced through the chemical mixing facility into the CVCS and subsequently, with the charging flow, into the RCS to control pH. Hydrazine is similarly introduced during startup from cold shutdown to scavenge oxygen. During normal operations, the hydrogen dissolved into the coolant in the volume control tank serves to scavenge the oxygen produced by radiolysis of water in the core region.

The reactor coolant letdown flow is purified in the mixed-bed and cation-bed demineralizers to remove ionic corrosion products and certain fission products. Filters at various locations remove particulates and resin fines. Fission gases are removed from the system by venting the volume control tank to the waste disposal system.

The boric acid (soluble neutron absorber) concentration and the reactor coolant inventory are controlled by the reactor makeup control system. The reactor makeup control system consists of a group of instruments arranged to provide a preselected makeup composition in the required volume to the RCS. The concentrated boric acid is stored in two boric acid tanks equipped with pumps to provide recirculation and feed to the boric acid blender. The primary makeup water pumps take suction from the primary makeup water storage tank and provide feed to the boric acid blender on demand. The flow from the boric acid blender is directed to either the charging pump suction manifold or the volume control tank.

Boron Recovery

Excess borated water is diverted from the letdown line to the holdup tanks when the volume control tank liquid level exceeds a set limit. This collected water is subsequently processed in batches for boron recovery. The liquid is pumped through the evaporator feed ion exchangers and the ion exchange filter to the boric acid evaporator/gas stripper units.

Vapor produced in the units is condensed, purified in the evaporator condensate demineralizers, filtered, and collected in the monitor tanks. The condensate can then be routed to the primary water storage tank, the radwaste discharge header, or the CVCS holdup tanks, depending on sample analysis results. The evaporator bottoms are discharged through a concentrates filter to the concentrates holding tank. Concentrates not meeting specifications are recycled to the holdup tanks for reprocessing or disposal to the auxiliary building drain tank; those meeting specifications are routed to the boric acid tanks to provide feed to the boric acid blender when makeup is required.

The flow rate through the boron recovery system is controlled by the capacity of the evaporator/gas stripper units, which can handle 3.4 m^3/hr of feed each, or 6.8 m^3/hr with both units operating simultaneously. The evaporator feed ion exchangers and the evaporator condensate demineralizer can handle 16.4 m^3/hr each.

B.2.4 Liquid Radioactive Waste Systems

Systems and equipment for processing liquid radwastes at the reference PWR are located in the auxiliary and fuel buildings.

B.2.4.1 Dirty Radioactive Waste System

The dirty radioactive waste (DRW) system collects, processes, and monitors liquids having low radioactivity and high particulate content. The system is shown schematically in Figure B.2-10.

The sources of liquids for this system are floor drains in contaminated areas and the drains from the sample sink and radioactive chemical laboratory. The liquid collected in this system is not reusable and, depending on the quantity and nature of contamination, is either processed or

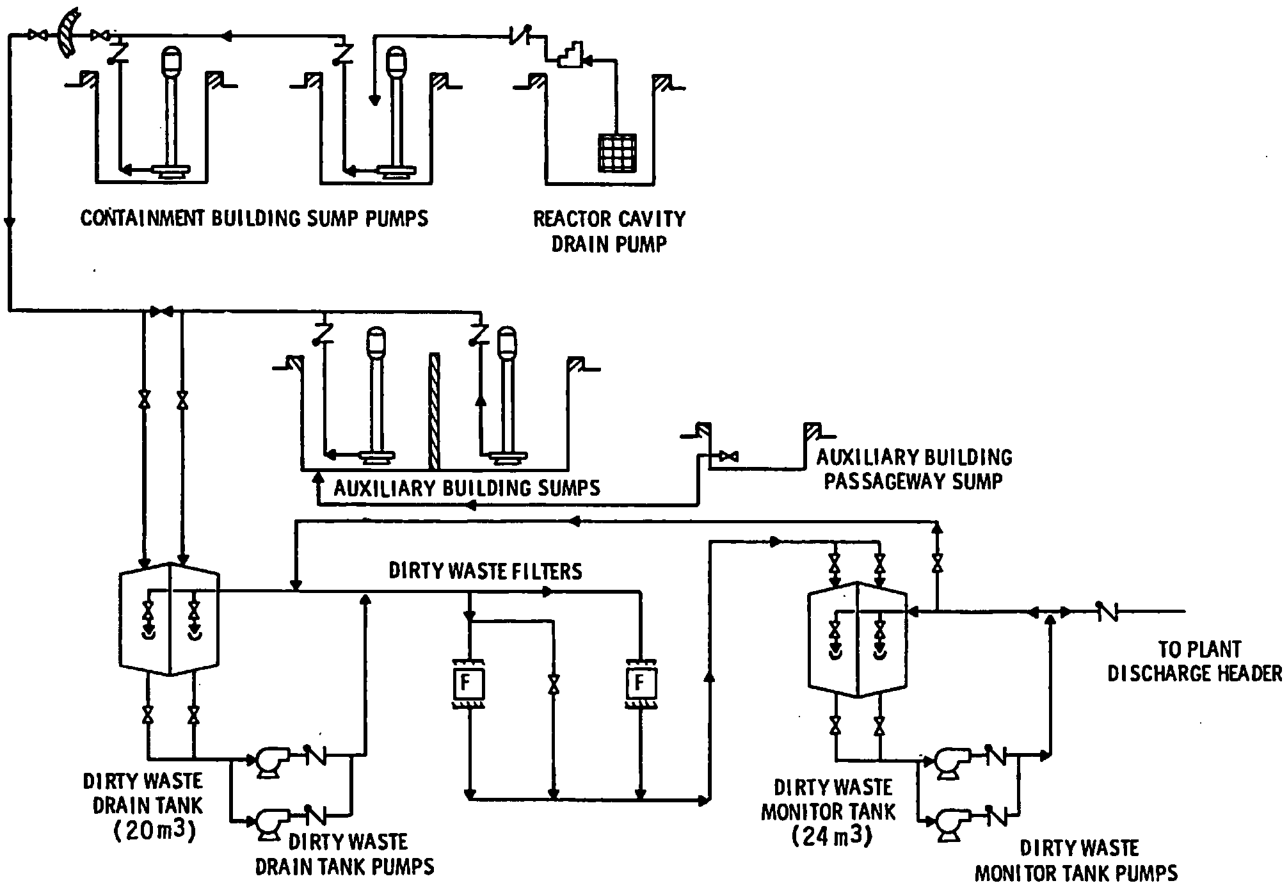


FIGURE B.2-10. PWR Dirty Radioactive Waste System

discharged. The only method of radioactivity reduction provided by the system, other than removal of particulates by filtration, is the decay of radionuclides prior to discharge. Waste volumes found to exceed allowable discharge limits are transferred to the clean radioactive waste system for processing.

B.2.4.2 Clean Radioactive Waste System

The clean radioactive waste (CRW) system is shown schematically in Figure B.2-11.

The auxiliary building drain tank collects drainage from nuclear steam supply system components within the auxiliary building. The chemical waste drain tank collects chemically contaminated drainage from the steam generator

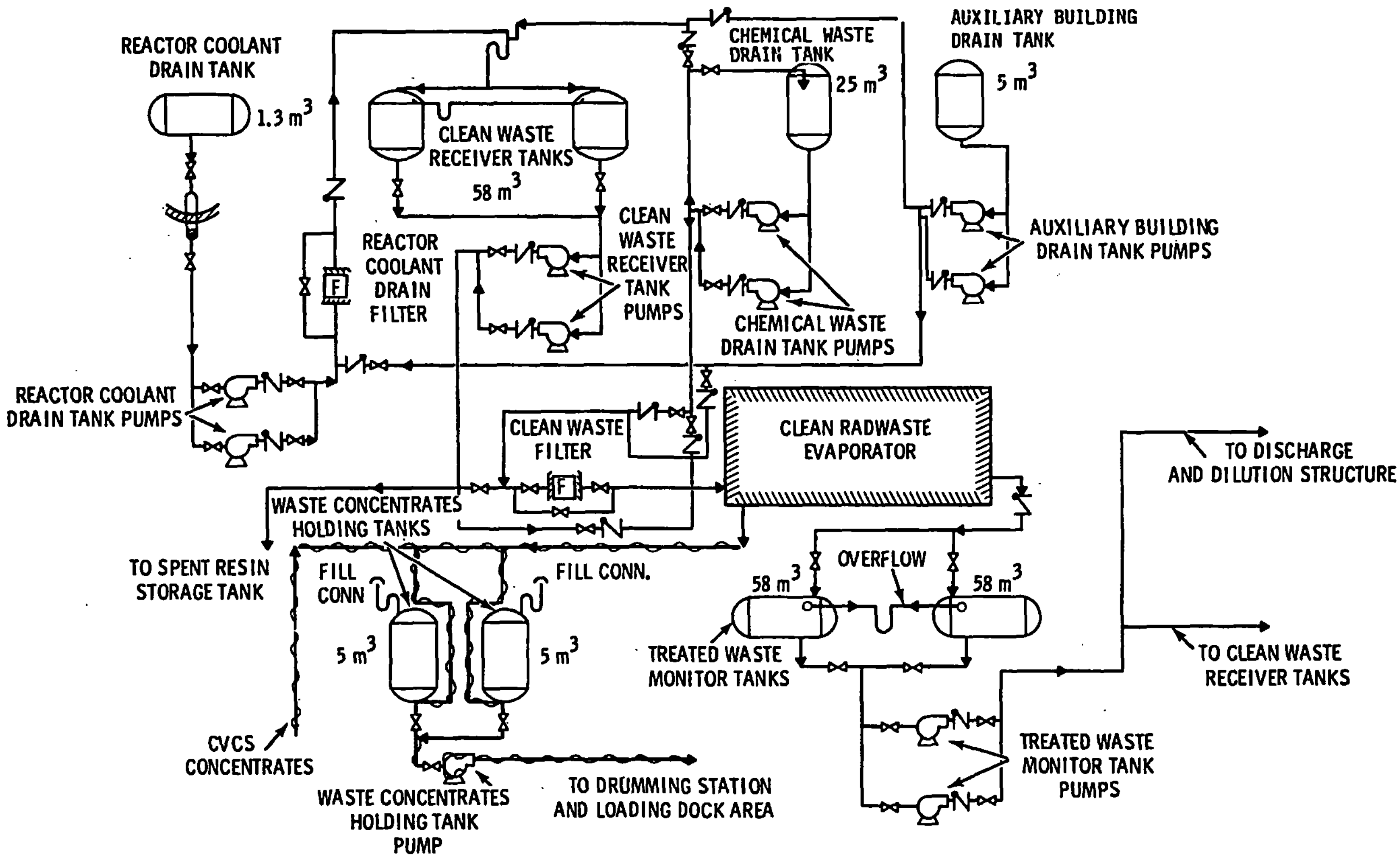


FIGURE B.2-11. PWR Clean Radioactive Waste System blowdown system. The reactor coolant drain tank collects contaminated drainage that originates within the containment. The clean waste receiver tanks collect waste from pump discharges as well as from drains with a high probability of being contaminated with particulate matter or chemicals.

The CRW evaporator concentrates radioactive wastes collected in the tanks, producing distillate with radioactive contamination levels reduced by a factor of at least 10^3. The clean waste filter removes particulate matter that could reduce evaporator efficiency. The CRW evaporator is designed for a continuous feed rate of 3.4 m^3/hr.

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

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

B.2.5 Spent Fuel Handling and Storage

During reactor defueling, the spent fuel removed from the reactor is transferred through the transfer tunnel to the spent fuel storage pool in the fuel building (see Section B.1.4). The spent fuel assembly is unloaded from the fuel assembly container on the transfer car using the spent fuel handling tool attached to the spent fuel pool bridge hoist. The fuel assembly is then placed in a spent fuel storage rack.

The spent fuel handling tool is a manually activated tool used to handle fuel assemblies in the spent fuel pool. An operator on the spent fuel pool bridge guides and operates the tool, which is attached to the end of a long pole suspended from the spent fuel pool bridge hoist.

The spent fuel pool bridge is a wheel-mounted walkway that spans the spent fuel pool and carries an electric monorail hoist on an overhead structure. Hoist travel and tool length are designed to limit the maximum lift of a fuel assembly to a safe shielding depth in the pool water.

B.2.6 Heating, Ventilating, and Air Conditioning Systems

The auxiliary and fuel buildings heating, ventilating, and air conditioning (HVAC) system consists of a single outdoor air supply system and two exhaust systems, one exhausting only the spent fuel pool area and the other exhausting the remaining building areas. Exhaust air is monitored for radioactivity and discharged through a common vent at the top of the containment building.

The common supply system consists of two parallel 50%-capacity vane-axial direct-driven fans that draw outside air through an automatic roll filter and electric heating coil and direct it to all parts of the buildings through a ductwork system. The system is designed to supply approximately 2350 m^3/min with both fans operating.

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

The exhaust system that serves the remaining parts of the auxiliary and fuel buildings consists of exhaust inlets, located throughout the buildings, connected by ductwork to four parallel exhaust air plenums designed to handle approximately 793 m^3/min each. Each plenum contains an automatic roll prefilter, a bank of HEPA filters, motorized isolation dampers at each end of the filter trains, and backdraft dampers. The system is designed to exhaust approximately 2379 m^3/min with three exhausts in service and the fourth in standby.

A portion of the total exhaust flow originates from spaces of the auxiliary building that contain CVCS equipment. Because this equipment processes reactor coolant, it presents an increased airborne radioactivity potential. Exhaust from these spaces is drawn by a separate booster fan through a filtration unit consisting of a prefilter, a HEPA filter, a deep-bed charcoal filter, and a final HEPA filter before discharge to normal building exhaust ducts.

B.2.7 Auxiliary and Fuel Buildings Major Equipment

Major equipment contained in the auxiliary and fuel buildings is listed in Table B.2-1.

TABLE B.2-1. Equipment List: PWR Auxiliary and Fuel Buildings

Equipment Piece (Quantity)	Mass (each) kg	Overall Dimensions (each) L x W x H, m
Waste Gas Compressor (2)	3 629	3 x 1.2 x 1.5
Pool Purification Filter (1)	163	279 mm dia x 1.17
Pool Skimmer Filter (1)	68	229 mm dia x 1.14
Pool Demineralizer Filter (1)	163	279 mm dia x 1.17
Dirty Waste Filter (2)	34	178 mm dia x 0.91
Clean Waste Filter (1)	30	178 mm dia x 0.66
Reactor Coolant Drain Filter (1)	159	406 mm dia x 1.42
Reactor Coolant Filter (1)	91	381 mm dia x 1.30
Boric Acid Filter (1)	91	381 mm dia x 1.30
Ion Exchange Filter (1)	68	305 mm dia x 1.02
Condensate Filter (2)	68	305 mm dia x 1.02
Seal Water Filter (1)	181	406 mm dia x 1.68
Seal Water Injection Filter (2)	748	254 mm dia x 1.91
Component Cooling Water Hx. (2)	31 751	1 499 mm dia x 9.75
Spent Fuel Pool Hx. (2)	2 767	508 mm dia x 5.79
Seal Water Hx. (1)	771	356 mm dia x 4.27
Letdown Hx. (1)	862	457 mm dia x 5.49
Residual Heat Removal Hx. (2)	10 478	914 mm dia x 91.44
Auxiliary Building Sump Pump (2)	590	4.57 long
Residual Heat Removal Pumps (2)	3 084	2.74 long
Containment Spray Pumps (2)	3 084	2.74 long

(contd on next page)

TABLE B.2-1. (contd)

Equipment Piece (Quantity)	Mass (each) kg	Overall Dimensions (each) L x W x H, m
Service Water Booster Pumps (4)		
Spent Fuel Pool Cooling Pumps (2)	408	1.52 x 0.46 x 0.61
Spent Fuel Pool Purification Pump (1)	454	1.52 x 0.46 x 0.61
Spent Fuel Pool Skimmer Pump (1)	318	1.22 x 0.46 x 0.61
Component Cooling Water Pumps (3)	6 804	3.15 x 1.42 x 1.60
Component Cooling Makeup Pumps (2)	363	1.22 x 0.30 x 0.61
Primary Water Makeup Pumps (2)	363	1.22 x 0.30 x 0.61
Reactor Coolant Drain Tank Pumps (2)	227	1.22 x 0.30 x 0.60
Auxiliary Building Drain Tank Pumps (2)	91	0.91 x 0.30 x 0.30
Dirty Waste Drain Tank Pumps (2)	227	1.22 x 0.30 x 0.61
Dirty Waste Monitor Tank Pumps (2)	91	0.91 x 0.30 x 0.30
Clean Waste Rec. Tank Pumps (2)	227	1.22 x 0.30 x 0.61
Chemical Waste Drain Tank Pumps (2)	91	0.91 x 0.30 x 0.30
Treated Water Monitor Tank Pumps (2)	104	0.91 x 0.30 x 0.30
Waste Concentrates Holding Tank Pumps (1)	104	0.91 x 0.30 x 0.30
Safety Injection Pumps (2)	3 901	4.37 x 4.02 x 1.07
Posit. Displace. Charging Pump (1)	8 029	4.27 x 1.73 x 1.32
Centrifugal Charging Pumps (2)	7 752	5.44 x 1.27 x 1.40
Boric Acid Transfer Pumps (2)	280	1.32 x 0.38 x 0.53
Holdup Tank Recirc. Pump (1)	288	1.32 x 0.38 x 0.86
Gas Stripper Feed Pumps (2)	227	0.66 x 0.56 x 0.36
Chem. Vol. Cont. Sys. Mont. Tank Pumps (2)	272	1.32 x 0.38 x 0.53
Concent. Holding Tank Transfer Pumps (2)	91	0.46 x 0.48 x 0.25
Steam Generator Blowdown Tank (1)	907	1.83 dia x 3.05
Component Cooling Water Surge Tanks (2)	907	2.13 dia x 2.44
Contain. Spray Additive Tanks	1 134	2.74 dia x 3.05
Component Cooling Water Chem. Add. Tank (1)	454	0.61 dia x 1.52
Fuel Pool Demineralizer	998	1.22 dia x 3.05
Reactor Coolant Drain Tank (1)	771	0.91 dia x 2.44
Auxiliary Building Drain Tank (1)	953	1.83 dia x 2.74
Clean Waste Rec. Tanks (2)	4 970	3.05 dia x 9.14
Chem. Waste Drain Tank (1)	2 449	3.05 dia x 1.52
Spent Resin Storage Tank (1)	3 084	2.74 dia x 3.35

(contd on next page)

TABLE B.2-1. (contd)

Equipment Piece (Quantity)	Mass (each) kg	Overall Dimensions (each) L x W x H, m
Treated Waste Monitor Tank (2)	5 080	3.05 dia x 7.92
Dirty Waste Drain Tank (1)	2 966	3.05 dia x 3.96
Dirty Waste Monitor Tank (1)	2 631	3.05 dia x 3.66
Waste Concen. Holding Tank (2)	953	1.82 dia x 3.05
Waste Gas Surge Tank	408	0.91 dia x 1.83
Waste Gas Decay Tank	4 899	3.05 dia x 4.88
Boron Inj. Tank (1)	12 927	1.67 dia x 3.81
Chem. Vol. Cont. Sys. Demin. (3)	476	0.66 dia x 1.65
Resin Fill Tank (1)	118	1.63 dia x 1.88
Volume Control Tank (1)	2 223	2.29 dia x 3.18
Boric Acid Tanks (2)	9 072	3.66 dia x 10.36
Boric Acid Batching Tank (1)	658	1.22 dia x 1.78
Evap. Ion Exch. (5)	476	0.66 dia x 1.65
Concentrates Holding Tank (1)	1 588	1.68 dia x 2.36
CVCS Holdup Tanks (3)	13 608	5.49 dia x 10.36
CVCS Monitor Tanks (2)	9 072	6.10 dia x 3.05
Electric Steam Boiler (1)	5 897	3.05 x 2.13 x 6.71
Clean Radioactive Waste Evap. (1)	18 144	5.79 x 2.74 x 3.66
Boric Acid Evap. and Gas Stripper (2) (Skid Mounted)	9 525	4.57 x 3.35 x 2.79

B.3 CONTROL BUILDING

The control building consists of four floors above grade, as shown in Figure B.3-1. It is structurally connected to the auxiliary building and is approximately 18 m in height with lateral dimensions of about 31 m by 24 m. The framing members of the building are structural steel. Floor slabs are reinforced concrete and precast prestressed concrete panels. Concrete block masonry is used for walls.

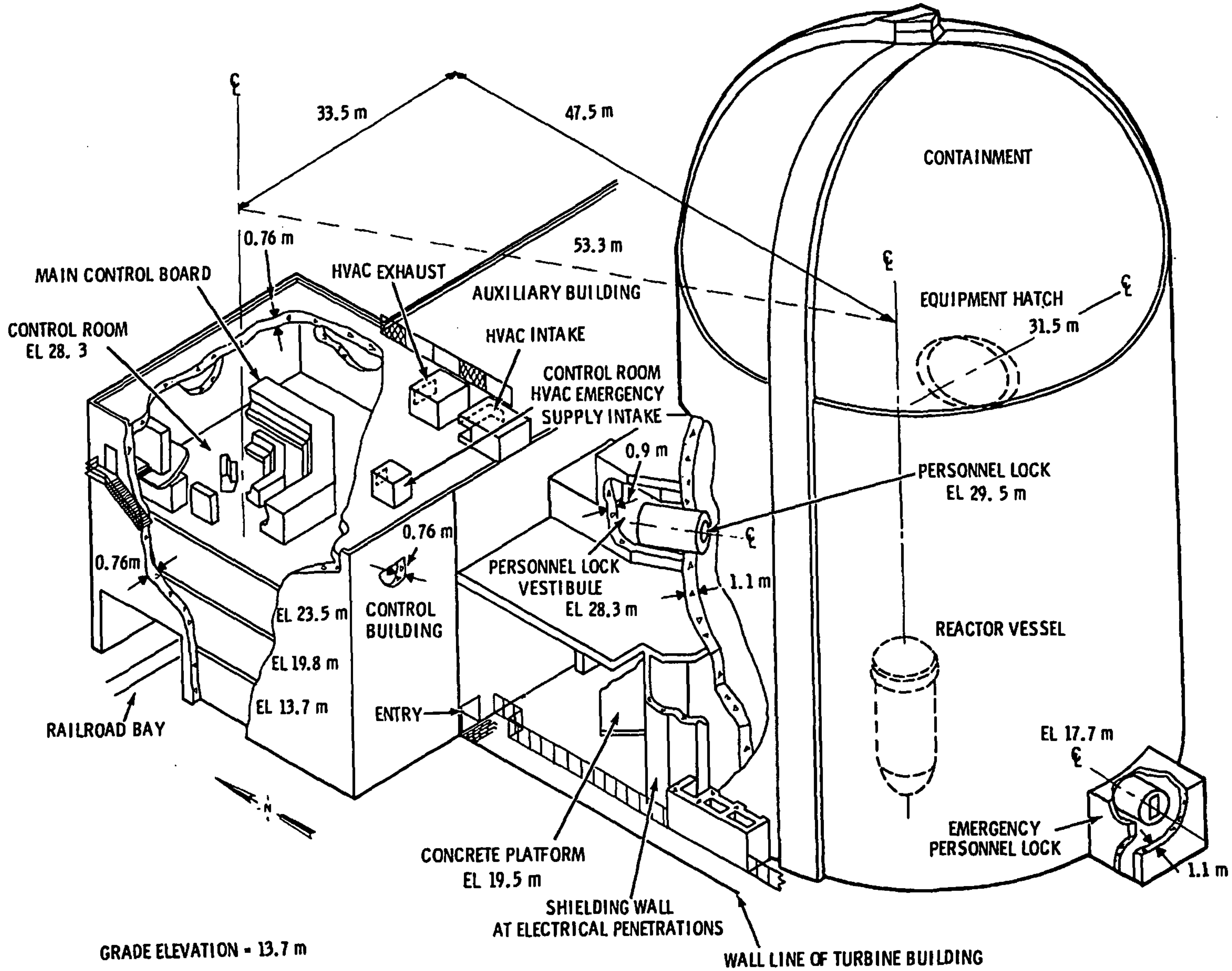


FIGURE B.3-1. Isometric View of the PWR Control Building

The principal contents of the control building are the reactor control room, the cable spreading room, process control laboratories and counting rooms, and personnel facilities. Major components and equipment items are as follows:

<u>Elevation 13.7 m</u>

- radiological chemical laboratory
- radiation sample room
- counting room
- decontamination room
- showers and locker rooms.

<u>Elevation 18.6 m and 19.8 m</u>

- battery rooms
- electrical auxiliaries
- telephone equipment
- mechanical room.

<u>Elevation 23.5 m</u>

- cable spreading room
- computer room.

<u>Elevation 28.3 m</u>

- control room and annex
- supervisor's room
- chart room
- water sampling laboratory.

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. Trojan Final Safety Analysis Report, Portland General Electric Co., Portland, Oregon, September 1973.

3. RESAR-3 Preliminary Safety Analysis Report, Westinghouse Electric Co., Pittsburg, Pennsylvania, November 1973.

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

APPENDIX C

DETAILS OF REFERENCE PWR ACCIDENT SCENARIOS AND RESULTANT CONTAMINATION LEVELS

Details to support the descriptions of the reference PWR accident scenarios and resultant contamination levels presented in Chapter 8 are given in this appendix. The details include estimated inventories of fission products released from damaged reactor fuel for the three reference accident scenarios and an explanation of the methods used to estimate radiation exposure rates from fission product contamination of the containment building. A summary of measured radioactive contamination levels and radiation exposure rates in the TMI-2 reactor building is also given in this appendix for comparison with the contamination levels and exposure rates postulated for the reference accident scenarios of Chapter 8.

C.1 DETAILS OF FISSION PRODUCT RADIONUCLIDE RELEASES FOR THE REFERENCE ACCIDENTS

This section provides details of estimated fission product releases from damaged reactor fuel for the three reference accidents. Inventory details are shown in Tables C.1-1, C.1-2, and C.1-3. The bases and assumptions used to estimate these fission product inventories are given in the following paragraphs.

The fission product source activity inventory (i.e., the amount of fission product radioactivity in the fuel) is taken from the Reactor Safety Study (RSS).[1] The RSS source 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 RSS inventory was calculated by means of the ORIGEN[2] program for a 1000-MWe (3200-MWt) three-region PWR core with a composition that is typical of four-loop Westinghouse PWRs. It was assumed in this calculation that the three regions of the core operated at a constant specific power density of 40 kW/kg of uranium charged. Inventories were calculated for

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

Radionuclide	Source Activity (Ci)[b]	Cladding Failure Release Fraction	Fuel Melt Release Fraction[c]	Total Release Fraction	Activity Released (Ci)	Half-Life (days)	Activity After 1 Year (Ci)
^{58}Co	7.8×10^{5}	--	--	--	--	7.10×10^{1}	--[d]
^{60}Co	2.9×10^{5}	--	--	--	--	1.92×10^{3}	--
^{85}Kr	5.6×10^{5}	0.003	--	0.003	1.7×10^{3}	3.95×10^{3}	1.6×10^{3}[e]
^{85m}Kr	2.4×10^{7}	0.003	--	0.003	7.2×10^{4}	1.83×10^{-1}	--
^{87}Kr	4.7×10^{7}	0.003	--	0.003	1.4×10^{5}	5.28×10^{-2}	--
^{88}Kr	6.8×10^{7}	0.003	--	0.003	2.0×10^{5}	1.17×10^{-1}	--
^{86}Rb	2.5×10^{4}	0.005	--	0.005	1.3×10^{2}	1.87×10^{1}	--
^{89}Sr	9.4×10^{7}	0.0000001	--	0.0000001	9.4×10^{0}	5.21×10^{1}	7.3×10^{-2}
^{90}Sr	3.7×10^{6}	0.0000001	--	0.0000001	3.7×10^{-1}	1.10×10^{4}	3.6×10^{-1}
^{91}Sr	1.1×10^{8}	0.0000001	--	0.0000001	1.1×10^{1}	4.03×10^{-1}	--
^{90}Y	3.9×10^{6}	--	--	--	--	2.67×10^{0}	3.6×10^{-1}[f]
^{91}Y	1.2×10^{8}	--	--	--	--	5.90×10^{1}	--
^{95}Zr	1.5×10^{8}	--	--	--	--	6.52×10^{1}	--
^{97}Zr	1.5×10^{8}	--	--	--	--	7.10×10^{-1}	--
^{95}Nb	1.5×10^{8}	--	--	--	--	3.50×10^{1}	--
^{99}Mo	1.6×10^{8}	--	--	--	--	2.80×10^{0}	--
^{99m}Tc	1.4×10^{8}	--	--	--	--	2.50×10^{-1}	--
^{103}Ru	1.1×10^{8}	--	--	--	--	3.95×10^{1}	--
^{105}Ru	7.2×10^{7}	--	--	--	--	1.85×10^{-1}	--
^{106}Ru	2.5×10^{7}	--	--	--	--	3.66×10^{2}	--
^{105}Rh	4.9×10^{7}	--	--	--	--	1.50×10^{0}	--
^{127}Te	5.9×10^{6}	0.00001	--	0.00001	5.9×10^{1}	3.91×10^{-1}	--
^{127m}Te	1.1×10^{6}	0.00001	--	0.00001	1.1×10^{1}	1.09×10^{2}	1.1×10^{0}
^{129}Te	3.1×10^{7}	0.00001	--	0.00001	3.1×10^{2}	4.80×10^{-2}	--
^{129m}Te	5.3×10^{6}	0.00001	--	0.00001	5.3×10^{1}	3.40×10^{1}	3.1×10^{-2}
^{131m}Te	1.3×10^{7}	0.00001	--	0.00001	1.3×10^{2}	1.25×10^{0}	--
^{132}Te	1.2×10^{8}	0.00001	--	0.00001	1.2×10^{3}	3.25×10^{0}	--
^{127}Sb	6.1×10^{6}	0.00001	--	0.00001	6.1×10^{1}	3.88×10^{0}	--
^{129}Sb	3.3×10^{7}	0.00001	--	0.00001	3.3×10^{2}	1.79×10^{-1}	--
^{131}I	8.5×10^{7}	0.0017	--	0.0017	1.4×10^{5}	8.05×10^{0}	--

(contd on next page)

TABLE C.1-1. (contd)

Radionuclide	Source Activity (Ci)(b)	Cladding Failure Release Fraction	Fuel Melt Release Fraction(c)	Total Release Fraction	Activity Released (Ci)	Half-Life (days)	Activity After 1 Year (Ci)
^{132}I	1.2×10^{8}	0.0017	--	0.0017	2.0×10^{5}	9.58×10^{-2}	--
^{133}I	1.7×10^{8}	0.0017	--	0.0017	2.9×10^{5}	8.75×10^{-1}	--
^{134}I	1.9×10^{8}	0.0017	--	0.0017	3.2×10^{5}	3.66×10^{-2}	--
^{135}I	1.5×10^{8}	0.0017	--	0.0017	2.6×10^{5}	2.80×10^{-1}	--
^{133}Xe	1.7×10^{8}	0.003	--	0.003	5.1×10^{5}	5.28×10^{0}	--
^{135}Xe	3.4×10^{7}	0.003	--	0.003	1.0×10^{5}	3.84×10^{-1}	--
^{134}Cs	7.5×10^{6}	0.005	--	0.005	3.8×10^{4}	7.50×10^{2}	2.7×10^{4}
^{136}Cs	3.0×10^{6}	0.005	--	0.005	1.5×10^{4}	1.30×10^{1}	--
^{137}Cs	4.7×10^{6}	0.005	--	0.005	2.4×10^{4}	1.10×10^{4}	2.3×10^{4}
^{140}Ba	1.6×10^{8}	0.0000001	--	0.0000001	1.6×10^{1}	1.28×10^{1}	--
^{140}La	1.6×10^{8}	--	--	--	--	1.67×10^{0}	--
^{141}Ce	1.5×10^{8}	--	--	--	--	3.23×10^{1}	--
^{143}Ce	1.3×10^{8}	--	--	--	--	1.38×10^{0}	--
^{144}Ce	8.5×10^{7}	--	--	--	--	2.84×10^{2}	--
^{143}Pr	1.3×10^{8}	--	--	--	--	1.37×10^{1}	--
^{147}Nd	6.0×10^{7}	--	--	--	--	1.11×10^{1}	--
^{239}Np	1.6×10^{9}	--	--	--	--	2.35×10^{0}	--
^{238}Pu	5.7×10^{4}	--	--	--	--	3.25×10^{4}	--
^{239}Pu	2.1×10^{4}	--	--	--	--	8.9×10^{6}	--
^{240}Pu	2.1×10^{4}	--	--	--	--	2.4×10^{6}	--
^{241}Pu	3.4×10^{6}	--	--	--	--	5.35×10^{3}	--
^{241}Am	1.7×10^{3}	--	--	--	--	1.5×10^{5}	--
^{242}Cm	5.0×10^{5}	--	--	--	--	1.63×10^{2}	--
^{244}Cm	2.3×10^{4}	--	--	--	--	6.63×10^{3}	--
Totals					2.3×10^{6}		5.2×10^{4}

(a) Scenario 1 assumes 10% fuel cladding failure and no fuel melting.
(b) From NUREG 75/014, Table VI 3-1.
(c) The fuel melt release fraction is zero for a scenario 1 accident.
(d) Less than 1×10^{-3} Ci.
(e) Released to the atmosphere by controlled venting of the containment building prior to the start of initial cleanup.
(f) Daughter of ^{90}Sr.

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

Radionuclide	Source Activity (Ci)[b]	Cladding Failure Release Fraction	Fuel Melt Release Fraction	Total Release Fraction	Activity Released (Ci)	Half-Life (days)	Activity After 1 Year (Ci)
^{58}Co	7.8×10^5	--	0.0015	0.0015	1.2×10^3	7.10×10^1	3.4×10^1
^{60}Co	2.9×10^5	--	0.0015	0.0015	4.4×10^2	1.92×10^3	3.8×10^2
^{85}Kr	5.6×10^5	0.015	0.044	0.059	3.3×10^4	3.95×10^3	3.1×10^4[c]
^{85m}Kr	2.4×10^7	0.015	0.044	0.059	1.4×10^6	1.83×10^{-1}	--[d]
^{87}Kr	4.7×10^7	0.015	0.044	0.059	2.8×10^6	5.28×10^{-2}	--
^{88}Kr	6.8×10^7	0.015	0.044	0.059	4.0×10^6	1.17×10^{-1}	--
^{86}Rb	2.6×10^4	0.025	0.038	0.063	1.6×10^3	1.87×10^1	2.1×10^{-3}
^{89}Sr	9.4×10^7	0.0000005	0.005	0.005	4.7×10^5	5.21×10^1	3.7×10^3
^{90}Sr	3.7×10^6	0.0000005	0.005	0.005	1.8×10^4	1.10×10^4	1.8×10^4
^{91}Sr	1.1×10^8	0.0000005	0.005	0.005	5.5×10^5	4.03×10^{-1}	--
^{90}Y	3.9×10^6	--	0.00015	0.00015	5.8×10^2	2.67×10^0	1.8×10^4[e]
^{91}Y	1.2×10^8	--	0.00015	0.00015	1.8×10^4	5.90×10^1	2.5×10^2
^{95}Zr	1.5×10^8	--	0.00015	0.00015	2.2×10^4	6.52×10^1	4.6×10^2
^{97}Zr	1.5×10^8	--	0.00015	0.00015	2.2×10^4	7.10×10^{-1}	--
^{95}Nb	1.5×10^8	--	0.00015	0.00015	2.2×10^4	3.50×10^1	4.6×10^2[f]
^{99}Mo	1.6×10^8	--	0.0015	0.0015	2.4×10^5	2.80×10^0	--
^{99m}Tc	1.4×10^8	--	0.0015	0.0015	2.1×10^5	2.50×10^{-1}	--
^{103}Ru	1.1×10^8	--	0.0015	0.0015	1.6×10^5	3.95×10^1	2.7×10^2
^{105}Ru	7.2×10^7	--	0.0015	0.0015	1.1×10^5	1.85×10^{-1}	--
^{106}Ru	2.5×10^7	--	0.0015	0.0015	3.8×10^4	3.66×10^2	1.9×10^4
^{105}Rh	4.9×10^7	--	0.0015	0.0015	7.4×10^4	1.50×10^0	--
^{127}Te	5.9×10^6	0.00005	0.0075	0.0076	4.5×10^4	3.91×10^{-1}	--
^{127m}Te	1.1×10^6	0.00005	0.0075	0.0076	8.4×10^3	1.09×10^2	8.2×10^2
^{129}Te	3.1×10^7	0.00005	0.0075	0.0076	2.4×10^5	4.80×10^{-2}	--
^{129m}Te	5.3×10^6	0.00005	0.0075	0.0076	4.0×10^4	3.40×10^1	2.4×10^1
^{131m}Te	1.3×10^7	0.00005	0.0075	0.0076	9.9×10^4	1.25×10^0	--
^{132}Te	1.2×10^6	0.00005	0.0075	0.0076	9.1×10^5	3.25×10^0	--
^{127}Sb	6.1×10^6	0.00005	0.0075	0.0076	4.6×10^4	3.88×10^0	--

(contd on next page)

TABLE C.1-2. (contd)

Radionuclide	Source Activity (Ci)(b)	Cladding Failure Release Fraction	Fuel Melt Release Fraction	Total Release Fraction	Activity Released (Ci)	Half-Life (days)	Activity After 1 Year (Ci)
^{129}Sb	3.3×10^{7}	0.00005	0.0075	0.0076	2.5×10^{5}	1.79×10^{-1}	--
^{131}I	8.5×10^{7}	0.0085	0.044	0.052	4.4×10^{6}	8.05×10^{0}	--
^{132}I	1.2×10^{8}	0.0085	0.044	0.052	6.2×10^{6}	9.58×10^{-2}	--
^{133}I	1.7×10^{8}	0.0085	0.044	0.052	8.8×10^{6}	8.75×10^{-1}	--
^{134}I	1.9×10^{8}	0.0085	0.044	0.052	9.9×10^{6}	3.66×10^{-2}	--
^{135}I	1.5×10^{8}	0.0085	0.044	0.052	7.8×10^{6}	2.80×10^{-1}	--
^{133}Xe	1.7×10^{8}	0.015	0.044	0.059	1.0×10^{7}	5.28×10^{0}	--
^{135}Xe	3.4×10^{7}	0.015	0.044	0.059	2.0×10^{6}	3.84×10^{-1}	--
^{134}Cs	7.5×10^{6}	0.025	0.038	0.063	4.7×10^{5}	7.50×10^{2}	3.4×10^{5}
^{136}Cs	3.0×10^{6}	0.025	0.038	0.063	1.9×10^{5}	1.30×10^{1}	--
^{137}Cs	4.7×10^{6}	0.025	0.038	0.063	3.0×10^{5}	1.10×10^{4}	2.9×10^{5}
^{140}Ba	1.6×10^{8}	0.0000005	0.005	0.005	8.0×10^{5}	1.28×10^{1}	2.1×10^{-3}
^{140}La	1.6×10^{8}	--	0.00015	0.00015	2.4×10^{4}	1.67×10^{0}	2.1×10^{-3}(g)
^{141}Ce	1.5×10^{8}	--	0.00015	0.00015	2.2×10^{4}	3.23×10^{1}	8.8×10^{0}
^{143}Ce	1.3×10^{8}	--	0.00015	0.00015	2.0×10^{4}	1.38×10^{0}	--
^{144}Ce	8.5×10^{7}	--	0.00015	0.00015	1.3×10^{4}	2.84×10^{2}	5.3×10^{3}
^{143}Pr	1.3×10^{8}	--	0.00015	0.00015	2.0×10^{4}	1.37×10^{1}	--
^{147}Nd	6.0×10^{7}	--	0.00015	0.00015	9.0×10^{3}	1.11×10^{1}	--
^{239}Np	1.6×10^{9}	--	0.00015	0.00015	2.4×10^{5}	2.35×10^{0}	--
^{238}Pu	5.7×10^{4}	--	0.00015	0.00015	8.6×10^{0}	3.25×10^{4}	8.6×10^{0}
^{239}Pu	2.1×10^{4}	--	0.00015	0.00015	3.2×10^{0}	8.9×10^{6}	3.2×10^{0}
^{240}Pu	2.1×10^{4}	--	0.00015	0.00015	3.2×10^{0}	2.4×10^{6}	3.2×10^{0}
^{241}Pu	3.4×10^{6}	--	0.00015	0.00015	5.1×10^{2}	5.35×10^{3}	5.1×10^{2}
^{241}Am	1.7×10^{3}	--	0.00015	0.00015	2.6×10^{-1}	1.5×10^{5}	2.6×10^{-1}
^{242}Cm	5.0×10^{5}	--	0.00015	0.00015	7.5×10^{1}	1.63×10^{2}	1.6×10^{1}
^{244}Cm	2.3×10^{4}	--	0.00015	0.00015	3.4×10^{0}	6.63×10^{3}	3.4×10^{0}
Totals					6.3×10^{7}		7.3×10^{5}

(a) Scenario 2 assumes 50% fuel cladding failure and 10% fuel melting.
(b) From NUREG 75/014, Table VI 3-1.
(c) Released to the outside atmosphere by controlled venting of the containment building prior to the start of decommissioning.
(d) Less than 1×10^{-3} Ci.
(e) Daughter of ^{90}Sr.
(f) Daughter of ^{95}Zr.
(g) Daughter of ^{140}Ba.

TABLE C.1-3. Estimated Inventory of Radioactivity Released from Damaged Reactor Fuel During Scenario 3 Accident[a]

Radionuclide	Source Activity (Ci)[b]	Cladding Failure Release Fraction	Fuel Melt Release Fraction	Total Release Fraction	Activity Released (Ci)	Half-Life (days)	Activity After 1 Year (Ci)
^{58}Co	7.8×10^5	--	0.015	0.015	1.2×10^4	7.10×10^1	3.4×10^2
^{60}Co	2.9×10^5	--	0.015	0.015	4.4×10^3	1.92×10^3	3.8×10^3
^{85}Kr	5.6×10^5	0.030	0.435	0.465	2.6×10^5	3.95×10^3	2.4×10^5[c]
^{85m}Kr	2.4×10^7	0.030	0.435	0.465	1.1×10^7	1.83×10^{-1}	--[d]
^{87}Kr	4.7×10^7	0.030	0.435	0.465	2.2×10^7	5.28×10^{-2}	--
^{38}Kr	6.8×10^7	0.030	0.435	0.465	3.2×10^7	1.17×10^{-1}	--
^{86}Rb	2.6×10^4	0.050	0.380	0.430	1.1×10^4	1.87×10^1	1.5×10^{-2}
^{89}Sr	9.4×10^7	0.000001	0.050	0.050	4.7×10^6	5.21×10^1	3.7×10^4
^{90}Sr	3.7×10^6	0.000001	0.050	0.050	1.9×10^5	1.10×10^4	1.8×10^5
^{91}Sr	1.1×10^8	0.000001	0.050	0.050	5.5×10^6	4.03×10^{-1}	--
^{90}Y	3.9×10^6	--	0.0015	0.0015	5.8×10^3	2.67×10^0	1.8×10^5[e]
^{91}Y	1.2×10^8	--	0.0015	0.0015	1.8×10^5	5.90×10^1	2.5×10^3
^{95}Zr	1.5×10^8	--	0.0015	0.0015	2.3×10^5	6.52×10^1	4.8×10^3
^{97}Zr	1.5×10^8	--	0.0015	0.0015	2.3×10^5	7.10×10^{-1}	--
^{95}Nb	1.5×10^8	--	0.0015	0.0015	2.3×10^5	3.50×10^1	4.8×10^3[f]
^{99}Mo	1.6×10^8	--	0.015	0.015	2.4×10^6	2.80×10^0	--
^{99m}Tc	1.4×10^8	--	0.015	0.015	2.1×10^6	2.50×10^{-1}	--
^{103}Ru	1.1×10^8	--	0.015	0.015	1.7×10^6	3.95×10^1	2.9×10^3
^{105}Ru	7.2×10^7	--	0.015	0.015	1.1×10^6	1.85×10^{-1}	--
^{106}Ru	2.5×10^7	--	0.015	0.015	3.8×10^5	3.66×10^2	1.9×10^5
^{105}Rh	4.9×10^7	--	0.015	0.015	7.4×10^5	1.50×10^0	--
^{127}Te	5.9×10^6	0.0001	0.075	0.075	4.4×10^5	3.91×10^{-1}	--
^{127m}Te	1.1×10^6	0.0001	0.075	0.075	8.2×10^4	1.09×10^2	8.0×10^3
^{129}Te	3.1×10^7	0.0001	0.075	0.075	2.3×10^6	4.80×10^{-2}	--
^{129m}Te	5.3×10^6	0.0001	0.075	0.075	4.0×10^5	3.40×10^1	2.4×10^2
^{131m}Te	1.3×10^7	0.0001	0.075	0.075	9.8×10^5	1.25×10^0	--
^{132}Te	1.2×10^8	0.0001	0.075	0.075	9.0×10^6	3.25×10^0	--
^{127}Sb	6.1×10^6	0.0001	0.075	0.075	4.6×10^5	3.88×10^0	--
^{129}Sb	3.3×10^7	0.0001	0.075	0.075	2.5×10^6	1.79×10^{-1}	--
^{131}I	8.5×10^7	0.017	0.442	0.459	3.9×10^7	8.05×10^0	--
^{132}I	1.2×10^8	0.017	0.442	0.459	5.5×10^7	9.58×10^{-2}	--
^{133}I	1.7×10^8	0.017	0.442	0.459	7.8×10^7	8.75×10^{-1}	--
^{134}I	1.9×10^8	0.017	0.442	0.459	8.7×10^7	3.66×10^{-2}	--

(contd on next page)

TABLE C.1-3. (contd)

Radionuclide	Source Activity (Ci)[b]	Cladding Failure Release Fraction	Fuel Melt Release Fraction	Total Release Fraction	Activity Released (Ci)	Half-Life (days)	Activity After 1 Year (Ci)
^{135}I	1.5×10^8	0.017	0.442	0.459	6.9×10^7	2.80×10^{-1}	--
^{133}Xe	1.7×10^8	0.030	0.435	0.465	7.9×10^7	5.28×10^0	--
^{135}Xe	3.4×10^7	0.030	0.435	0.465	1.6×10^7	3.84×10^{-1}	--
^{134}Cs	7.5×10^6	0.050	0.380	0.430	3.2×10^6	7.50×10^2	2.3×10^6
^{136}Cs	3.0×10^6	0.050	0.380	0.430	1.3×10^5	1.30×10^1	4.6×10^{-3}
^{137}Cs	4.7×10^6	0.050	0.380	0.430	2.0×10^6	1.10×10^4	2.0×10^6
^{140}Ba	1.6×10^8	0.000001	0.050	0.050	8.0×10^6	1.28×10^1	2.1×10^{-2}
^{140}La	1.6×10^8	--	0.0015	0.0015	2.4×10^5	1.67×10^0	2.1×10^{-2}(g)
^{141}Ce	1.5×10^8	--	0.0015	0.0015	2.2×10^5	3.23×10^1	8.7×10^1
^{143}Ce	1.3×10^8	--	0.0015	0.0015	2.0×10^5	1.38×10^0	--
^{144}Ce	8.5×10^7	--	0.0015	0.0015	1.3×10^5	2.84×10^2	5.3×10^4
^{143}Pr	1.3×10^8	--	0.0015	0.0015	2.0×10^5	1.37×10^1	--
^{147}Nd	6.0×10^7	--	0.0015	0.0015	9.0×10^4	1.11×10^1	--
^{239}Np	1.6×10^9	--	0.0015	0.0015	2.4×10^6	2.35×10^0	--
^{238}Pu	5.7×10^4	--	0.0015	0.0015	8.6×10^1	3.25×10^4	8.6×10^1
^{239}Pu	2.1×10^4	--	0.0015	0.0015	3.2×10^1	8.9×10^6	3.2×10^1
^{240}Pu	2.1×10^4	--	0.0015	0.0015	3.2×10^1	2.4×10^6	3.2×10^1
^{241}Pu	3.4×10^6	--	0.0015	0.0015	5.1×10^3	5.35×10^3	5.1×10^3
^{241}Am	1.7×10^3	--	0.0015	0.0015	2.6×10^0	1.5×10^5	2.6×10^0
^{242}Cm	5.0×10^5	--	0.0015	0.0015	7.5×10^2	1.63×10^2	1.6×10^2
^{244}Cm	2.3×10^4	--	0.0015	0.0015	3.5×10^1	6.63×10^3	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) From NUREG 75/014, Table VI 3-1.
(c) Released to the atmosphere by controlled venting of the containment building prior to the start of initial cleanup.
(d) Less than 1×10^{-3} Ci.
(e) Daughter of ^{90}Sr.
(f) Daughter of ^{95}Zr.
(g) Daughter of ^{140}Ba.

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 26,400 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. Use of the RSS inventory is believed to provide radionuclide data of sufficient accuracy to allow reasonable estimates to be made of fission product contamination following the postulated reference accidents.

Accident scenario radionuclide inventories are obtained from the fission product source inventory by multiplying each individual isotopic contribution in the source inventory by an appropriate release fraction. The fraction of a particular fission product radionuclide that escapes from the fuel core during an accident depends on fuel temperatures and on other specific accident conditions. Detailed models for estimating fission product release during an accident are currently being tested[3] and are not yet in general use. In this study, fission product release during an accident is treated as being proportional to the fraction of fuel rods that experience cladding failure or fuel melting. This approach to estimating accident scenario radionuclide inventories is consistent with that used in the Reactor Safety Study.[1] The approach is believed to give radionuclide data of sufficient accuracy to allow reasonable estimates of radiation exposure to be made for the reference accidents.

The release of fission product radioactivity from damaged fuel during an accident is expressed in terms of a total release fraction, RF, defined by:

$$RF_i = (G_i \times F_c) + (M_i \times F_m) \tag{C.1}$$

where:

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

G_i = the 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 = the fraction of fuel rods experiencing cladding failure

M_i = the 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 = the fraction of fuel rods experiencing fuel melting.

Fission products can be divided into typical release groups, based on the ease with which they are volatilized. A grouping of eight categories proposed by the Reactor Safety Study,[1] listed in order of decreasing volatility, is shown in Table C.1-4. Gap release fractions and rod-melt release fractions for these categories are also shown in the table.

For the reference accidents postulated for this study, the assumed fractions of fuel rods experiencing cladding failure or fuel melting are as follows:

- scenario 1 accidents assume 10% cladding failure and no fuel melting
- scenario 2 accidents assume 50% cladding failure and 5% fuel melting
- scenario 3 accidents assume 100% cladding failure and 50% fuel melting.

The scenario 1 accident is intentionally chosen to have minor fuel damage and to result in a small release of radioactivity from the fuel core. Values of cladding failure and fuel melting for the scenario 2

TABLE C.1-4. Estimated Gap and Rod-Melt 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 rod-melt release fractions, 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.

accident are chosen to be intermediate between those for the scenario 1 and scenario 3 accidents. A value of 50% fuel melting is postulated for the scenario 3 accident. This is consistent with the value used for the severe core damage accident with delayed ECCS operation analyzed in an NRC study of fission product behavior during an LWR accident.[3,4]

The results of calculations of fission product releases from damaged fuel for the three reference accidents are shown in Tables C.1-1, C.1-2, and C.1-3. The cladding failure release fractions shown in the tables are the products of a gap release fraction multiplied by the fraction of fuel rods experiencing cladding failure ($G_i \times F_c$). The fuel melt release fractions are the products of a rod melt release fraction multiplied by the fraction of fuel rods experiencing fuel melting ($M_i \times F_m$). The total release fractions are the sums of a cladding release fraction plus a fuel melt release fraction.

Both the total estimated curies of fission product radioactivity released from damaged fuel during the reference accidents and the resultant radioactivity 1 year after the accidents are shown in Tables C.1-1 through C.1-3. The 1-year time period is chosen as being representative of the time delay that would occur between the accident and the start of cleanup operations inside the containment building. This time delay allows for planning and preparation activities, described in Section E.2 of Appendix E. During this planning and preparation period, the short-lived fission products decay to insignificant levels. The remaining fission products that contribute more than 0.1% to the total inventory all have half-lives that are significant compared to 1 year. The principal fission products remaining after 1 year include ^{85}Kr (assumed to be vented to the outside atmosphere prior to the start of cleanup operations), ^{90}Sr, ^{106}Ru, ^{134}Cs, ^{137}Cs, and ^{144}Ce.

As explained in Section 8.3 of Chapter 8, for the PWR accident scenarios it is assumed that 50% of the non-gaseous fission product radioactivity released from the damaged reactor fuel is plated out on internal surfaces of the reactor pressure vessel, on vessel internals, and on primary system piping, or is retained in the primary coolant. The remaining 50% of the released non-gaseous radioactivity is carried into the containment building by the escaping steam and water vapor during the course of a postulated accident. This radioactivity plates out on building and equipment surfaces or is retained in the sump water. The resultant fission product radioactivity in the PWR reactor building 1 year after the postulated accidents is 25,000 Ci for the scenario 1 accident, 350,000 Ci for the scenario 2 accident, and 2.5 million Ci for the scenario 3 accident.

C.2 RADIOACTIVE DECAY CHARACTERISTICS OF SELECTED FISSION PRODUCT RADIONUCLIDES

The principal fission product radionuclides present in surface and sump-water contamination 1 year after the reference LWR accidents are ^{90}Sr, ^{106}Ru, ^{134}Cs, ^{137}Cs, and ^{144}Ce. The decay of these radionuclides is described in the following paragraphs. The decay schemes are taken from the Table of Isotopes.[5]

^{90}Sr

The radioactive decay scheme of $^{90}Sr \xrightarrow{\beta^-} {}^{90}Y$ is shown in Figure C.2-1. ^{90}Sr has a radioactive half-life of 28.1 years and decays by β^- emission to ^{90}Y, which is also radioactive and decays with a half-life of 64 hours. The maximum energy of the β^- particles from the decay of ^{90}Sr to ^{90}Y is 0.545 MeV. There is no gamma emission associated with this decay The maximum energy of the β^- emission from the decay of ^{90}Y to the ^{90}Zr ground state is 2.27 MeV. About 0.2% of the decays of ^{90}Y nuclei are to an excited state, which decays to the ground state of ^{90}Zr with the emission of a 1.75-MeV gamma ray.

^{106}Ru

The radioactive decay scheme of $^{106}Ru \xrightarrow{\beta^-} {}^{106}Rh$ is shown in Figure C.2-2. ^{106}Ru has a radioactive half-life of 367 days and decays by β^- emission to ^{106}Rh, which is also radioactive and decays with a half-life of 30 seconds. The maximum energy of the β^- emission from the decay of ^{106}Ru to ^{106}Rh is

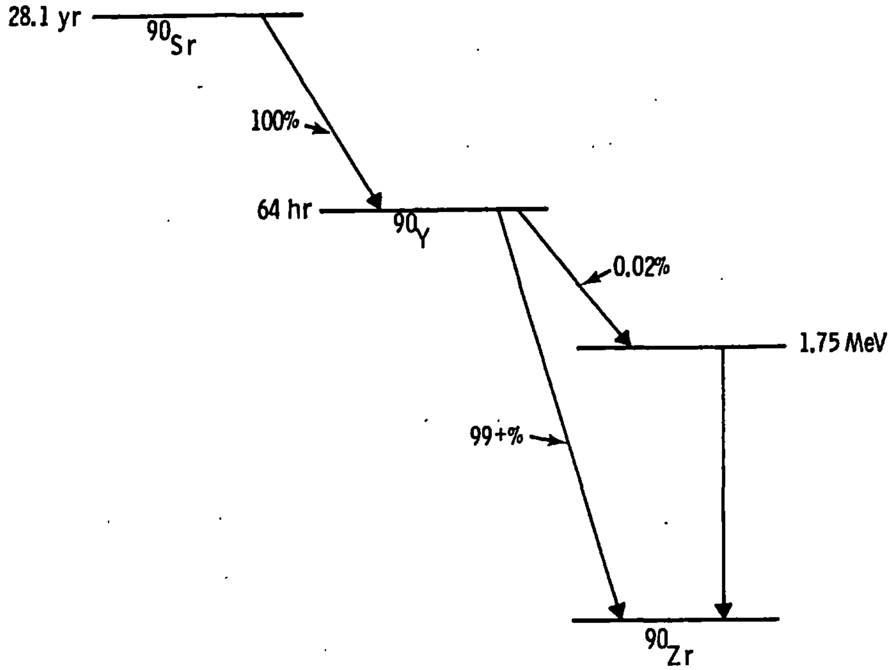


FIGURE C.2-1. Radioactive Decay Scheme for $^{90}Sr \xrightarrow{\beta^-} {}^{90}Y$

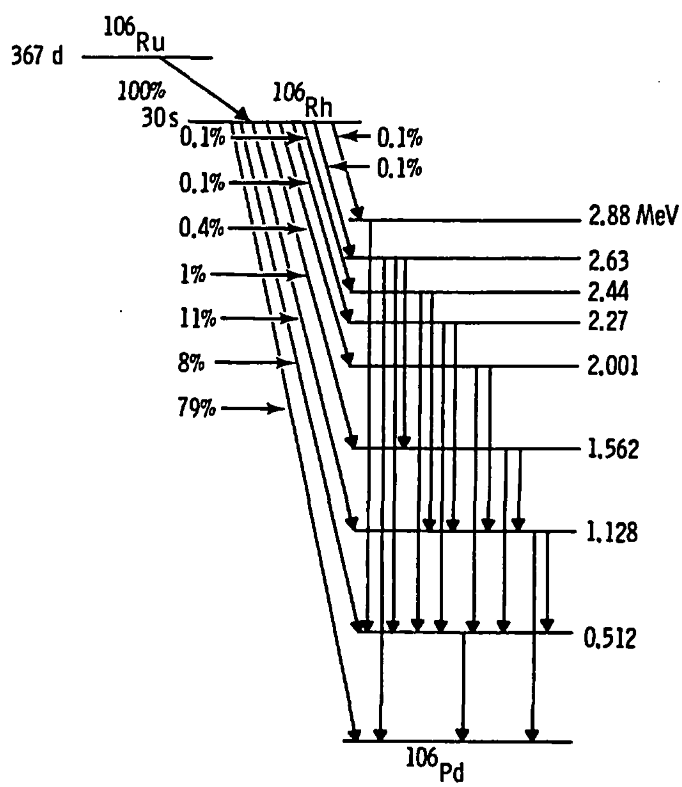


FIGURE C.2-2. Radioactive Decay Scheme for $^{106}Ru \xrightarrow{\beta^-} {}^{106}Rh$

only 0.0392 MeV, and there is no gamma emission associated with this decay. Approximately 79% of the decays of ^{106}Rh nuclei are to the ground state of ^{106}Pd, accompanied by the emission of a β^- particle having a maximum energy of 3.55 MeV. The remainder of the decays are to excited levels in ^{106}Pd and result in a complex gamma-ray spectrum with photon energies that range from 0.512 MeV to 2.63 MeV. The two most prominent gamma rays in this spectrum have energies of 0.512 MeV and 0.624 MeV.

^{134}Cs

The radioactive decay scheme of ^{134}Cs is shown in Figure C.2-3. ^{134}Cs has a radioactive half-life of 2.05 years and decays by β^- emission to excited levels of ^{134}Ba. Approximately 71% of the decays are to the 1.400 MeV level of ^{134}Ba and are accompanied by β^- emissions with a maximum energy of 0.662 MeV. The remainder of the decays are to higher ^{134}Ba energy levels and are accompanied by β^- emissions of lower energy. De-excitation of the ^{134}Ba nuclei results in a complex gamma-ray spectrum with photon energies that range from 0.475 MeV to 1.365 MeV. The two most prominent gamma rays in this spectrum have energies of 0.605 MeV (98% of the decays) and 0.796 MeV (88% of the decays).

^{137}Cs

The radioactive decay scheme of ^{137}Cs is shown in Figure C.2-4. ^{137}Cs has a radioactive half-life of 30 years and decays by β^- emission. In approximately 6.5% of the decays, the transition is to the ^{137}Ba ground state. The maximum energy of the β^- emissions for these transitions is 1.176 MeV. In the remaining 93.5% of the decays, the transition is to an excited state of ^{137}Ba with subsequent decay to the ground state by the emission of a 0.662-MeV gamma photon.

^{144}Ce

The radioactive decay scheme of $^{144}Ce \xrightarrow{\beta^-} {}^{144}Pr$ is shown in Figure C.2-5. ^{144}Ce has a radioactive half-life of 284 days and decays by β^- emission to ^{144}Pr. The β^- emission associated with this radioactive transformation has

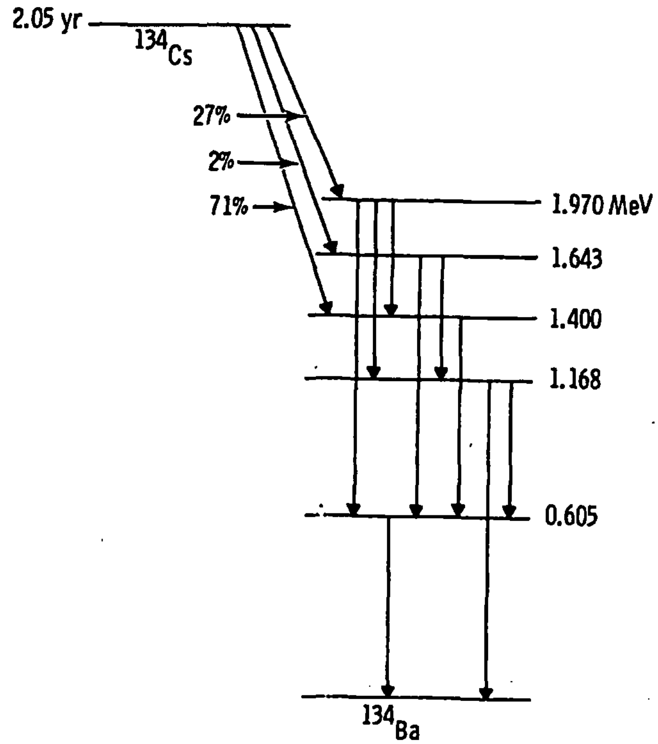


FIGURE C.2-3. Radioactive Decay Scheme for ^{134}Cs

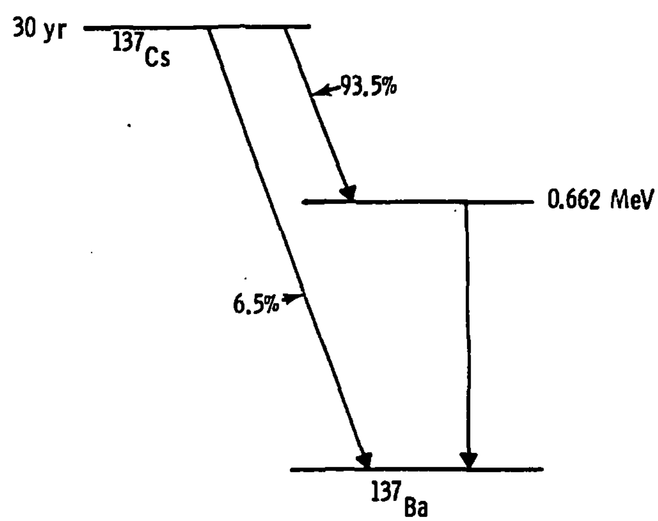


FIGURE C.2-4. Radioactive Decay Scheme for ^{137}Cs

a maximum energy of 0.309 MeV. ^{144}Ce decay is accompanied by a complex spectrum of very weak gamma rays that range in energy from 0.034 MeV to 0.134 MeV. ^{144}Pr is also radioactive and decays with a half-life of

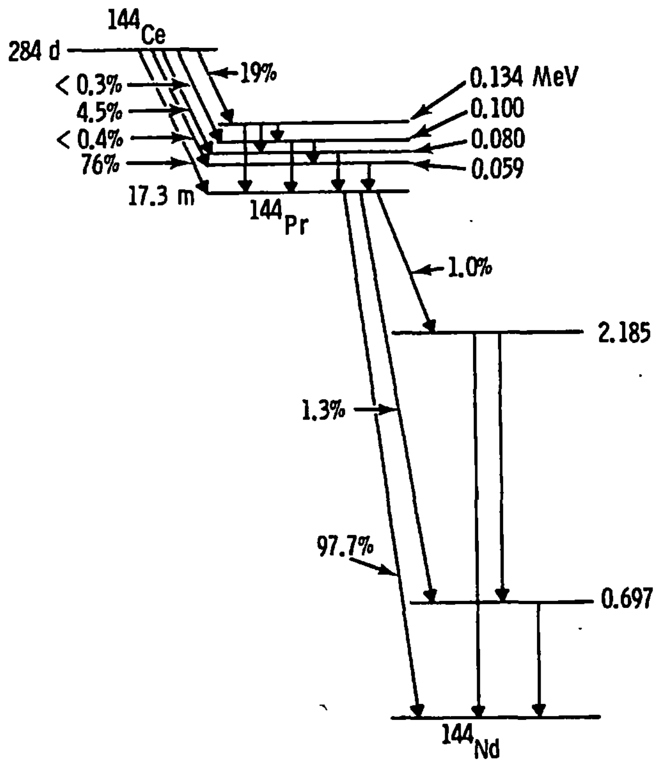


FIGURE C.2-5. Radioactive Decay Scheme for $^{144}Ce \xrightarrow{\beta^-} {}^{144}Pr$

17.3 minutes. Approximately 98% of the decays of ^{144}Pr are to the ground state of ^{144}Nd and are accompanied by β^- emission with a maximum energy of 2.99 MeV. Approximately 1% of the ^{144}Pr decays are to an excited state of ^{144}Nd at 2.185 MeV and approximately 1% are to an excited state at 0.695 MeV. De-excitation results in the emission of gamma rays having energies of 0.697 MeV (1.6% of the decays), 1.49 MeV (0.3% of the decays), and 2.19 MeV (0.7% of the decays).

C.3 DETAILS OF EXPOSURE RATE CALCULATIONS

This section provides details of the methods used to estimate average radiation exposure rates from fission product contamination of the containment building for the reference accident scenarios described in Chapter 8. Average exposure rates are estimated for gamma radiation from plateout on building surfaces and equipment and for radiation from contaminated sump water. The methodology uses equations for calculating photon fluxes from uniformly contaminated regular geometric sources given in the Reactor Shielding Design Manual.[6]

The cesium isotopes, ^{134}Cs and ^{137}Cs, are the major fission product radionuclides present in surface and sump water contamination 1 year after the reference accidents. These two isotopes comprise 96% of the scenario 1 radioactive inventory, 84% of the scenario 2 inventory, and 83% of the scenario 3 inventory. 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. In this study, 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 photon.

C.3.1 Calculations of Exposure Rates from Plateout on Building and Equipment Surfaces

This section describes the equations used to estimate average exposure rates from gamma radiation plated out on building and equipment surfaces as a result of the reference PWR accidents. As discussed in Chapter 8, the washdown of walls during and after an accident by moisture that condenses on these surfaces would result in wall contamination levels that are approximately a factor of 100 lower than floor contamination levels. An analysis to evaluate the effect on exposure rate calculations of changes in room size and changes in the distribution of radioactive contamination between the floor, walls, and ceiling of a room is described in Section C.4. The results of this analysis show that, for a room in which the contamination levels on walls and ceilings are much smaller than the average contamination level on the floor, the calculated average exposure rate is approximately equal to the exposure rate from a uniformly contaminated infinite plane. Therefore, the infinite plane equation is used to estimate exposure rates from plateout on containment building surfaces.

The equation for the photon flux at a point P located on the axis at a distance d from a plane disk source with uniform surface contamination (see Figure C.3-1) is:

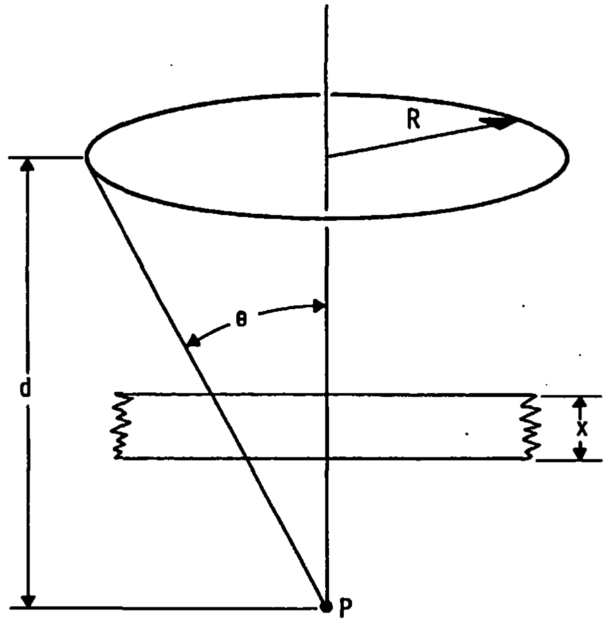


FIGURE C.3-1. Plane Disk Source

$$\phi = \frac{BS_A}{2} \; E_1(b_1) - E_1(b_1 \sec \theta) \qquad (C.2)$$

where:

ϕ = flux (photons/cm^2-sec)

B = dose buildup factor (dimensionless)

S_A = source strength (photons/cm^2-sec)

$b_1 = \sum \mu_i \, x_i$ (dimensionless)

μ_i = total macroscopic attenuation coefficient for the i^{th} shield material (cm^{-1})

x_i = thickness of the i^{th} shield material (cm)

$$E_n(b) = b^{n-1} \int_b^\infty \frac{e^{-t}}{t^n} \, dt \text{ for } n \geq 0.$$

B is assumed to be equal to 1 for air. (See Section C.3.2 for an explanation of the dose buildup factor.) Graphs for evaluating the logarithmic integral, $E_n(b)$, are given in Reference 6.

For a uniformly contaminated infinite plane, the photon flux in air at a distance d becomes:

$$\phi = \frac{S_A}{2} [E_1(\mu_a d)] \tag{C.3}$$

where: μ_a is the macroscopic attenuation coefficient for air. For 0.662-MeV gamma rays, μ_a is approximately equal to 0.0001 cm^{-1}. Thus, at a distance of 1 m from an infinite plane that is uniformly contaminated with ^{137}Cs, the photon flux is:

$$\phi = \frac{S_A}{2} [E_1 (0.01)]. \tag{C.4}$$

Conversion factors for converting from photon flux (in units of photons/cm^2-sec) to exposure rate (in units of R/hr) are given in Reference 6. For 0.662-MeV gamma radiation, the conversion factor is 1 photon/cm^2-sec = 1.4×10^{-6} R/hr. Average exposure rates from surface contamination, calculated using Equation C.4 and this conversion factor, are given in Table 8.3-1 of Chapter 8 for the PWR accident scenarios.

C.3.2 Calculations of Exposure Rates from Contaminated Sump Water

This section describes the equations used to estimate average exposure rates from gamma radiation from contaminated sump water. The sump water is modeled as a disk source of finite thickness with a uniform volume concentration of radioactivity.

The equation for the photon flux at a point P on the axis at a distance d from a disk of finite thickness with uniform volume contamination and with h<R (see Figure C.3-2) is:

$$\phi = \left[\frac{BSv}{2\mu_s}\left(E_2(b_1) - E_2(b_3) + \frac{E_2(b_3 \sec\theta)}{\sec\theta} - \frac{E_2(b_1 \sec\theta)}{\sec\theta}\right)\right] \tag{C.5}$$

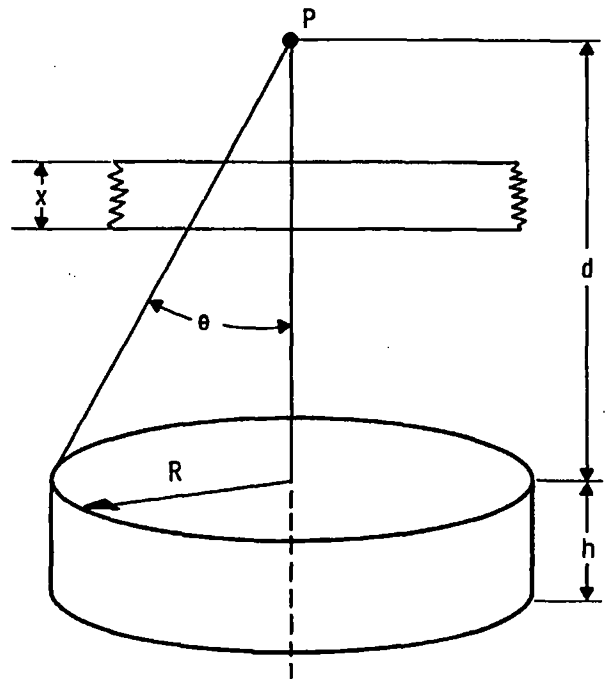


FIGURE C.3-2. Disk Source of Finite Thickness

where:

ϕ = flux (photons/cm^2-sec)

B = dose buildup factor (dimensionless)

S_v = source strength (photons/cm^3-sec)

μ_s = total macroscopic attenuation coefficient of the source material (cm^{-1})

$b_1 = \sum \mu_i x_i$ (dimensionless)

μ_i = total macroscopic attenuation coefficient for the ith shield material (cm-1)

x_i = thickness of the ith shield material (cm)

$b_3 = b_1 + \mu_s h$ (dimensionless).

Values of macroscopic attenuation coefficients for the source material (water) and for air and concrete for 0.662-MeV photons are:

μ_s (water) $= 0.085\ cm^{-1}$

μ_i (air) $= 0.0001\ cm^{-1}$

μ_i (concrete) $= 0.18\ cm^{-1}$.

The total attenuation coefficient (μ) for gamma radiation is the sum of a total absorption coefficient (μ_a) and a Compton scattering coefficient (σ_s). For the transmission of gamma radiation through thick absorbers, the measured exposure rate, is often found to exceed the estimated attenuated exposure rate because multiple scattering events return some photons of degraded energy to the detector. This empirical result is expressed in terms of a build-up factor, B, which is defined as:

$$B = \frac{\text{observed exposure rate}}{\text{estimated exposure rate}}.$$

For exposure rate calculations in which concrete shield material is assumed to be interposed between the sump water and the location where the exposure rate is calculated, it is necessary to utilize a dose build-up factor. This factor can be calculated from the equation:

$$B = A_1\ e^{-a_1 \mu x} + A_2\ e^{-a_2 \mu x} \tag{C.6}$$

where: A_1, A_2, a_1, and a_2 are empirical constants that are a function of gamma energy and x is the thickness of the concrete shield. Values of the empirical constants for 0.662-MeV gamma rays, taken from Reference 6, are:

$A_1 = 11.3$

$A_2 = 1-A_1 = -10.3$

$a_1 = 0.1$

$a_2 = 0.019$.

Estimated dose buildup factors for various thicknesses of concrete, calculated for 0.662-MeV gamma rays on the basis of Equation C.6, are shown in Table C.3-1.

TABLE C.3-1. Dose Buildup Factors for Concrete for 0.662-MeV Gamma Rays

Concrete Thickness (cm)	Dose Buildup Factor, B
30	10.3
45	16.6
60	25.6

Average gamma radiation exposure rates from sump water contamination, calculated using Equation C.5 with the appropriate dose buildup factor to determine the photon flux and using the dose conversion factor of 1 photon/cm^2 - sec = 1.4 x 10^6 R/hr, are given in Table 8.3-1 of Chapter 8 for the PWR accident scenarios.

C.4 ROOM MODEL FOR ESTIMATING RADIATION EXPOSURE RATES

The BWR decommissioning study(7) included an investigation of the relationship between room size, surface radionuclide contamination levels, and calculated gamma radiation exposure rates. The analysis evaluated the effect on the calculated exposure rate of changes in room size and changes in the distribution of radioactive contamination between the floor, the walls, and the ceiling of the room. Radiation exposure rates were calculated at a point in air 1 m above the floors of rooms of various sizes with 1 Ci/m^2 of ^{60}Co uniformly distributed on the room surfaces. Exposure rate calculations were also made for rooms whose walls and ceilings contained 50% and 10%, respectively, of the floor contamination level. For comparison purposes, the gamma exposure rate at a point in air 1 m from a uniformly contaminated infinite plane was also calculated.

The floor of the reference room was assumed to be square and the walls of the room were assumed to be 3 m high. Room surfaces were modeled as uniformly contaminated disk sources, and exposure rates were calculated using the applicable equations from the Reactor Shielding Design Manual, as described in Section C.3.

The results of this analysis are shown in Figure C.4-1. For very large, uniformly contaminated rooms, the exposure rate approaches a value equal to

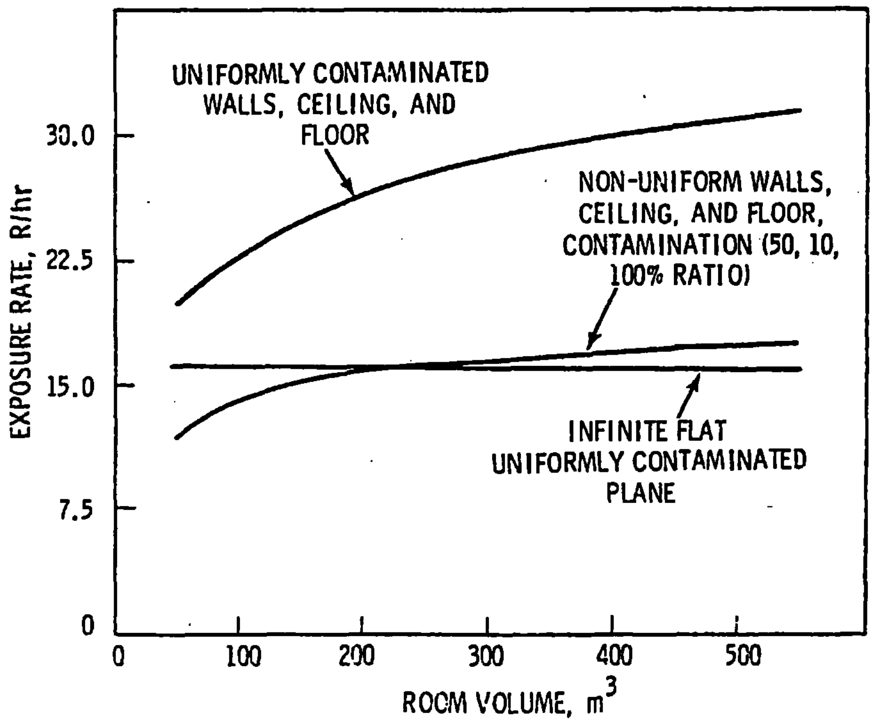


FIGURE C.4-1. Exposure as a Function of Room Volume for a ^{60}Co Deposition of 1 Ci/m^2 (from Reference 7)

2 times the exposure rate from a uniformly contaminated infinite plane. For rooms in which the contamination levels on the walls and ceiling are much smaller than the contamination level on the floor, the calculated average exposure rate is closely approximated by the exposure rate from a uniformly contaminated infinite plane.

For the reference accidents of this study, it is assumed that surface contamination is caused by the plateout on building surfaces of radioactivity injected into the building atmosphere with the escaping steam or water vapor from a loss-of-coolant accident. For these reference accidents, washdown of the walls by moisture that condenses on building and equipment surfaces during and after the accident results in wall contamination levels that are much lower than floor contamination levels. Therefore, the gamma radiation exposure rate from surface contamination is modeled by assuming a uniformly contaminated infinite plane.

C.5 SUMMARY OF TMI-2 CONTAMINATION DATA

A loss-of-coolant accident occurred at Three Mile Island Nuclear Station Unit 2 (TMI-2) on 28 March 1979 when the reactor pressurizer relief valve stuck open following a turbine-generator trip and the reactor operators mistakenly turned off the high-pressure injection system that had been adding water to the primary cooling system. A brief description of this accident is given in Section 3.8 of Chapter 3. The accident resulted in significant radioactive contamination of building surfaces and equipment in both the reactor building and the auxiliary and fuel handling building and in flooding of the reactor building basement with contaminated primary coolant water. This section summarizes the contamination data at TMI-2. It is useful to summarize this information since the accident which occurred on 28 March 1979 was the only accident at a large power reactor resulting in significant plant contamination.

Measured radioactive contamination levels and radiation exposure rates in the TMI-2 reactor building approximately 1 year after the accident are summarized from data presented in the final programmatic environmental impact statement for decontamination of TMI-2.[8] The purpose of this brief description of TMI-2 accident consequences is to provide a basis for comparison with the reference reactor accident scenarios of Chapter 8. The contamination condition of the TMI-2 reactor building 1 year after the accident is summarized in Table C.5-1.

Five entries were made into the TMI-2 reactor building between 23 July and 11 December 1980 to map radiation fields, measure surface contamination, and inspect for damage to the building and equipment at the 305-ft (equipment hatch) and 347-ft (main working floor) elevations. (The 305-ft level is approximately 7 m above the basement floor.) Entry teams ranged in size from 2 to 14 persons and times spent inside the reactor building varied from 20 to 120 minutes. The maximum whole-body dose received by any individual for each entry was 220, 340, 570, 460, and 650 mrem, respectively. On the first entry, breathing air was supplied by individual tanks of compressed air; on succeeding entries, air from the reactor containment building was drawn through individual HEPA filters by battery-powered fans into pressurized full-face masks.

TABLE C.5-1. Contamination Condition of TMI-2 Reactor Building Approximately 1 Year After the Accident[a]

I - Sump Water[b]	
Volume of sump water	2650 m^3
Total radioactivity in sump water	5.3×10^5 Ci
Average concentration of dissolved radionuclides in sump water	190 Ci/m^3
Average concentration of filterable radioactive solids in sump water	10 Ci/m^3
II - Contamination on Building Surfaces	
Average removable contamination on 305-ft elevation floor[c]	1400 $\mu Ci/m^2$
Average removable contamination on 305-ft elevation walls	1 $\mu Ci/m^2$
Average gamma radiation exposure rate at 305-ft elevation	500 mR/hr
Average removable contamination on 347-ft elevation floor[d]	670 $\mu Ci/m^2$
Average removable contamination on 347-ft elevation walls	2 $\mu Ci/m^2$
Average gamma radiation exposure rate at 347-ft elevation	250 mR/hr

III - Relative Amounts of Principal Fission Product Radionuclides

Source	Radionuclide ^{137}Cs	^{134}Cs	^{90}Sr
Plateout in Reactor Building	83%	13%	4%
Dissolved Radionuclides in Sump Water	86%	13%	1%
Filterable Solids in Sump Water	10%	1%	89%
Primary Coolant System Water	51%	8%	41%

IV - Physical Condition of Reactor Building and Equipment

1. Building lights are operable.
2. Polar crane may be inoperable because of adverse environment, long period without maintenance, and heat damage to electrical components.
3. Gross contamination of building ventilation system.
4. Minor structural damage to equipment and doors.
5. Equipment in flooded basement is grossly contaminated and possibly inoperable.

(a) Data are from Reference 8.
(b) Sump water is contaminated coolant water that is standing to a depth of 2.44 m (8 ft) in the reactor building basement.
(c) The 305-ft elevation is the equipment hatch level.
(d) The 347-ft elevation is the working floor level.

Following the accident, about 2650 m^3 (700,000 gal) of contaminated water (sump water) flooded the reactor building basement at TMI-2 to a level of about 2.44 m above the basement floor. This water was primarily from the reactor coolant drain tank that overflowed when the reactor pressurizer relief valve stuck open early in the accident. Analyses(9) of the sump water composition show that the water contained approximately 200 μCi/mL of dissolved radionuclides and filterable radioactive solids. The principal radionuclides in the water were ^{90}Sr, ^{134}Cs, and ^{137}Cs. A gamma radiation measurement of 40 to 45 R/hr was obtained remotely at a distance of approximately 2 m above the sump water surface.

Radioactive deposits (plateout) were present on most of the approximately 28,000 m^2 of exposed building and equipment surfaces inside the reactor building. Measurements of removable contamination from samples obtained by wiping or scraping at 28 locations during the first and second entries into the building yielded the following results (in μCi/m^2): measurements at eight locations on the 305-ft elevation floor ranged from 300 to 4700, with an average of 1400; measurements at four locations on the 305-ft elevation walls ranged from 0.6 to 2, with an average of 1; measurements at two locations on the 347-ft elevation floor ranged from 640 to 690, with an average of 670; and measurements at four locations on the 347-ft elevation walls ranged from 1 to 4, with an average of 2. The principal radionuclides in the plateout were ^{90}Sr, ^{134}Cs, and ^{137}Cs.

Plateout, the contaminated sump water, and suspended particles were the major sources of worker exposure to radiation. General gamma radiation levels (exclusive of "hot spots") were measured during the first three entries into the building and found to range from 100 to 700 mR/hr at the 305-ft elevation and from 30 to 600 mr/hr at the 347-ft elevation. The average gamma radiation levels were estimated to be 500 mR/hr and 250 mR/hr at the 305-ft and 347-ft elevations, respectively. Higher radiation fields occurred at localized "hot spots" where concentrations of plateout were present and above the open stairwell, which is not protected by an intervening floor or wall from radiation from the sump water. Typical measurements of the gamma radiation in such areas were 1 to 2 R/hr at the air coolers, 2 to 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.

Only minor physical damage to building surfaces and equipment was observed during entries into the TMI-2 reactor building. Very limited information is available on the condition of equipment and services that may be needed to assist in cleanup operations at TMI-2. The containment spray system was actuated on 28 March 1979, following a hydrogen burn within the reactor building, and may still be operable. Some of the building lights were activated during the second entry. The electrically powered polar crane is inoperable because of heat and water damage to electrical components and because of the long period without maintenance. The reactor building air cooling units, located adjacent to the equipment hatch at the 305-ft level, are operable but are grossly contaminated. The air-ducting at this level is also highly contaminated and may require special decontamination or replacement. Because the basement was flooded with contaminated water, equipment located there is highly contaminated, and some of it may be inoperable due to water damage.

Table C.5-2 shows a comparison of reported radioactive contamination and radiation exposure levels in the TMI-2 reactor building approximately 1 year after the accident with postulated radioactive contamination and estimated average radiation exposure levels for the three PWR accident scenarios described in Chapter 8. The contamination levels at TMI-2 tend to fall between the contamination levels postulated for the scenario 2 and scenario 3 accidents of this study.

TABLE C.5-2. 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 8 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.

REFERENCES

1. Reactor Safety Study, WASH-1400 (NUREG 75/014), U.S. Nuclear Regulatory Commission, Washington, D.C., October 1975.

2. M. J. Bell, ORIGEN, the ORNL Isotope Generation and Depletion Code, ORNL-4628, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1973.

3. Technical Bases for Estimating Fission Product Behavior During LWR Accidents, NUREG-0772, U.S. Nuclear Regulatory Commission, Washington, D.C., June 1981.

4. R. R. Sherry, NUREG-0772--Technical Bases for Estimating Fission Product Behavior During LWR Accident, Paper presented at the Ninth Water Reactor Safety Research Information Meeting, Washington, D.C., October 26-30, 1981. NUREG/CP-0024, Vol. 3, March 1982.

5. C. M. Lederer, J. M. Hollander, and I. Perlman, Table of Isotopes, Sixth Edition, John Wiley and Sons, Inc., New York, New York, 1967.

6. T. Rockwell, Reactor Shielding Design Manual, TID-7004, U.S. Atomic Energy Commission, Washington, D.C., March 1956.

7. 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.

8.. Final Programmatic Environmental Impact Statement Related to 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.

9. R. E. Brooksbank and W. J. Armento, Post-Accident Cleanup of Radioactivity at the Three Mile Island Nuclear Power Station, ORNL/TM-7081, Oak Ridge National Laboratory, Oak Ridge, Tennessee, February 1980.

APPENDIX D

GENERIC CLEANUP AND DECOMMISSIONING INFORMATION

Although each cleanup and decommissioning of an accident-damaged reactor facility is a unique operation with requirements that depend on the specific facility and the nature of the accident, all post-accident cleanup and decommissioning operations include some common activities and requirements. This appendix provides information on some of these common elements, including:

- decontamination of structures and equipment
- contamination control
- special tool and equipment requirements
- liquid waste treatment
- packaging, transportation, and disposal of wastes
- essential systems and services
- quality assurance
- environmental surveillance.

D.1 DECONTAMINATION OF STRUCTURES AND EQUIPMENT

Three basic methods can be used to remove radioactive materials from contaminated surfaces. These are: 1) dissolution of the surface film containing the radionuclides, 2) physical cleaning of the surface, and 3) physical removal of the contaminated structural material. Various techniques used for each of these methods are discussed in this section.

D.1.1 Surface Film Dissolution

The major use of surface film dissolution during cleanup and decommissioning of a reactor is for internal decontamination of components. Several chemical decontamination methods are available for the removal of radioactive contamination from inside surfaces of piping, tanks, and other equipment of the reactor coolant system and related systems (e.g., the CVCS). Some methods are designed for piping loops where the chemicals can be recirculated until the desired degree of decontamination is obtained. Others are designed to complete the decontamination in one pass, so that they can be used where recirculation is impractical. Selected chemical decontamination methods for reactor coolant systems, with associated advantages and disadvantages, are presented in Table D.1-1. These methods are discussed in the following paragraphs. Additional information on chemical decontamination is available in References 1 through 7.

TABLE D.1-1. Comparison of Selected Chemical Decontamination Methods

Method	Advantages	Disadvantages
Alkaline permanganate; citric/oxalic acid (AP-Citrox)	Has been successfully used to decontaminate reactors. Corrosion rates for stainless steel and Inconel are very low. Decontamination factors of 10 to 100 can be expected.	Generally requires repeated cycles to obtain good decontamination. Generates large volumes of waste solutions requiring processing. Large tankage volume required for chemical preparation and handling.
Alkaline permanganate; ammonium citrate (APAC)	Has been successfully used to decontaminate reactors. Corrosion rates, application temperatures, and application times similar to AP-Citrox process.	Decontamination factors 2 to 10 times lower than AP-Citrox process. Generates large volumes of waste solutions requiring processing. Large tankage volume needed for chemical preparation and handling.
EDTA and Citrox, 5% solution	Only one cycle and one rinse necessary. Contaminants can be removed from recirculating solution via a side stream through an on-line cation resin bed, if desired. Chemicals can be added directly to system water. Decontamination factors of 10 to 20 can be expected.	Recirculation time is lengthy, probably several days. Potentially damaging to system components. Additional R&D needed to determine optimum conditions.
Inhibited phosphoric acid, 10% solution	Fast, effective method for decontamination of carbon steel. Decontamination factors of 10 to 100 obtainable in 10 to 20 minutes. Lower decontamination factors are observed for stainless steel.	Possible redeposition of contaminants if acid remains in contact with carbon steel longer than 20 minutes. Corrosive to reactor systems. Produces large volumes of waste solutions.

(contd on next page)

TABLE D.1-1. (contd)

Method	Advantages	Disadvantages
OPG (oxalic-peroxide-gluconic) solution	Has been used successfully to dissolve UO_2 and decontaminate reactor coolant systems. High decontamination factors are achievable in short times (a few hours). Compatible with stainless steel, carbon steel, Inconel, and aluminum.	Uses large quantities of chemicals and produces large volumes of waste solutions. Additional chemical decontamination may be required after fuel particulate dissolution is complete.
CAN-DECON	One-step process. Has been used on CANDU reactors. Chemical reagents added directly to reactor coolant. Coolant is continuously passed through filters and ion-exchange resins to regenerate solution during decontamination operation.	Modest decontamination factors (∿3 to 6) achieved. Requires long reagent contact times. Can be corrosive to mechanical seals.
Dow Chemical NS-1	Has been successfully used to decontaminate reactor components. Waste volumes generated are large but volumes are less than with two-step solvents. Decontamination factors of 10 to 100 are possible.	Requires long solvent contact times, probably several days. Corrosion rates for stainless steel and Inconel are modest but higher than for AP-Citrox or APAC.

Alkaline Permanganate-Citrox

The alkaline permanganate (AP)-citric acid-oxalic acid (Citrox) process has been used successfully for the decontamination of stainless steel and Inconel parts of several reactor systems. AP is an oxidizing agent used to oxidize chromium in the corrosion film to Cr_2O_3, which can subsequently be dissolved in an alkaline solution. It is used as a pretreatment step in multi-step decontamination programs to expose the remaining corrosion film matrix to subsequent chemical dissolution. Citrox is a reducing agent consisting of a mixture of citric (0.2 M) and oxalic (0.3 M) acids with a corrosion inhibitor. Citrox is highly corrosive to carbon steel and to 400 series stainless steel. A representative procedure for system decontamination by the AP-Citrox process might include the following:

1. AP circulation for 4 hours at 105°C
2. water rinse until the pH is less than 10
3. citrox circulation for 8 hours at 80°C
4. water rinse until the conductivity of the rinse water is less than 50 µmho.

High decontamination factors (typically 10 to 100) can be expected from this treatment. Large quantities of chemicals are required, and the process results in the generation of large quantities of liquid wastes (up to 5 system volumes) that require treatment and disposal. Demineralization, evaporation, and solidification methods can be used for final disposal of the chemical solutions.

Alkaline Permanganate-Ammonium Citrate

Alkaline permanganate-ammonium citrate (APAC) is also a two-step decontamination process. It has been used to decontaminate the Plutonium Recycle Test Reactor (PRTR) and the Shippingport reactor. Corrosion rates, application temperatures, and application times are very similar to the AP-Citrox process. Ethylenediaminetetraacetic acid (EDTA) can be added to the ammonium citrate to act as a complexing agent and prevent redeposition of solubilized fission products. The advantages and disadvantages of the APAC process are similar to those for the AP-Citrox process; however, decontamination factors achieved with the APAC process are 2 to 10 times lower.

EDTA and Citrox

This decontamination method utilizes the chelating agent EDTA in combination with citric/oxalic acid in a weak (5%) solution at controlled pH. The chemicals can be added directly to the system water, and contaminants can be removed from the recirculating solution via a side stream through an on-line cation resin bed, if desired. While extended recirculation times are expected to be required, probably several days to one week, only one system volume of the decontamination solution is needed and there are no intermediate water flushes. Thus the volume of contaminated liquid requiring treatment and disposal is significantly reduced from that of two-step processes. Final disposal of the decontamination solution can be accomplished by conventional demineralization, evaporation, and solidification methods.

Inhibited Phosphoric Acid

Phosphoric acid rapidly defilms and decontaminates carbon steel surfaces. At 60-70°C, inhibited dilute (10%) phosphoric acid solutions can be used to achieve decontamination factors in the range of 20-100 in approximately 20 minutes. If the acid remains in contact with the carbon steel surface longer than 20 minutes, a ferrous phosphate film develops and deposits on the surfaces along with the contamination. Decontamination factors for stainless steel are about one order of magnitude lower than for carbon steel. Decontamination with a phosphoric acid solution is generally followed by one or more water rinses, a passivating rinse employing ammonium hydroxide, and a rust-inhibiting rinse for protective considerations.

OPG

Oxalic-peroxide-gluconic solution (OPG) is relatively fast acting, 1 to 4 hours at a temperature of 80°C, and it is compatible with carbon steel, stainless steel, Inconel, and aluminum. Decontamination factors of 10 to 100 can be achieved. The solution was used for dissolution of UO_2 during decontamination of the PRTR. Several system volumes (one to five) might be needed to achieve the desired decontamination if fuel particulate dissolution is necessary. To remove radioactive contamination from piping containing fuel particulates, a system flush with OPG can be followed by a rinse and a flush with AP-Citrox, APAC, or EDTA-Citrox. Contaminated OPG solution can be processed for disposal by filtration, evaporation, and solidification methods.

CAN-DECON

CAN-DECON is a proprietary chemical decontamination process developed in Canada to decontaminate CANDU reactors.[8] The process involves the addition of dilute (0.1 wt%) chemical reagents directly to the liquid in the reactor coolant system. The solvent is recirculated at temperatures up to 150°C for several hours in a carbon steel system or for several days in a stainless steel system. Decontamination factors have been low, in the range of 3 to 6 for the CANDU reactors. CAN-DECON is a one-step process. A continuous stream of coolant is passed through filters and cation exchange resins to strip radionuclides from the process chemicals during the decontamination operation. This effectively regenerates the process solution. When the process is completed, the reagent and remaining dissolved corrosion products are removed by mixed-bed resins or a cation bed and a mixed bed in series.

Dow Chemical NS-1

NS-1 is a Dow Chemical Company proprietary high-concentration chemical decontamination process. The process consists of circulating the reagent mixture through the system at 120°C for 100 to 200 hours under a nitrogen blanket to dissolve the contaminated oxide film. The process was used at Peach Bottom 2 and 3 to decontaminate the regenerative heat exchangers. Decontamination factors for these operations ranged from 2 to 10. Recent tests in preparation for use of this process at Dresden 1 have shown that much higher decontamination factors are attainable. This is a one-step process, and waste volumes generated are lower than with two-step processes. Solvent contact times are long; and corrosion is slightly higher than with other processes described in this section.

Some of the chemical decontamination solutions listed in Table D.1-1 employ chelating agents to form complexes with the radioisotopes that are removed from the surfaces being cleaned. The shallow-land disposal of decontamination solutions containing chelating agents can result in potentially undesirable radioisotope migration conditions at the disposal site. A report by Duquid et al.[9] points out the possibility for chelating agents to form stable, soluble complexes with transition metals, rare earths, and transuranics. The complexed isotopes may migrate with water passing through soil much faster than isotopes in a positive ionic (cation) form. Chelating agents from one waste package may accelerate the leach rate of radionuclides and metals from other wastes buried at the site. They may also remove previously absorbed radionuclides from soil, leaving them in a soluble, chelated state.

Because of the potential for complications arising from the disposal of chelating agents, the NRC has imposed restrictions on the near surface disposal of wastes containing these materials. These restrictions, which are contained in Part 61 of 10 CFR,[10] are discussed in Section D.5.2. In choosing a decontamination technique, care should be taken to ensure that the wastes from processing of the decontamination solution can be disposed of in accordance with the disposal requirements of 10 CFR Part 61.

D.1.2 Physical Cleanup of Surfaces and Equipment

This section describes some physical methods for the removal of smearable radioactive contamination from surfaces such as walls, floors, and tank exteriors, and from equipment surfaces. Appropriate combinations of these methods can be used for the decontamination of an accident-damaged reactor facility. Physical methods considered include:

- remote decontamination using containment spray systems
- hose wash
- high-pressure water jet or steam jet
- vacuuming
- janitorial techniques
- strippable coatings
- sludge removal with portable recirculating vacuum filter system
- electropolishing
- vibratory finishing
- ultrasonic cleaning.

While these methods can be used for the decontamination of a variety of surfaces, they may not be effective or even applicable to the decontamination of untreated concrete because of the adsorption of contaminated liquids into the body of the material. Concrete decontamination can require the complete removal of a surface layer. Physical techniques for surface removal are described in Section D.1.3.

Remote Decontamination

The containment spray system, which can be operated from outside the containment, can be used to carry out a remote wash of containment surfaces following an accident. This system is similar to the sprinkler system used in public buildings, except that the flow of water is much greater. The primary advantage of a remote wash is that it provides an initial reduction in radiation levels with very little worker exposure. The disadvantages are that a large volume of contaminated waste water is generated and that only surface areas reached by the spray are decontaminated. A reduction of exposure levels by factors of 2 to 5 is achievable in some areas of the containment by this method.

The containment spray system may be ineffective in reducing surface contamination if the contamination has been present for a length of time such that the contamination has dried and hardened on the surface.

Hose Wash

Hose wash is a decontamination technique that allows workers to stay some distance from the radiation source. It offers the advantages of flow rate control, flow pattern, and directional properties and can be effective in the decontamination of hard-to-reach areas that cannot be adequately cleaned by remote operation of the building spray system.

The holdback-carrier technique can be used effectively with hose wash. In this technique, enough nonradioactive cesium is added to the water so that the number of atoms of inactive cesium in the solution far exceeds the number of atoms of radioactive cesium on the surface being cleaned. As the holdback-carrier solution is washed down a contaminated wall, the radioactive cesium on the wall is replaced by nonradioactive cesium. By using holdback carriers, the contaminant transferred to the wash is more likely to remain in solution than to be redeposited on the surface.

Because of low impact forces, hose wash is less effective than high pressure water jet or steam jet techniques. If the surface to be cleaned is covered with oil or grease, hose wash is ineffective. Depending on conditions, hose wash decontamination factors range from 2 to 100. Flow rates for hose wash are typically about 0.003 m^3/sec.

High-Pressure Water Jet or Steam Jet

High-pressure water or steam jets are quite effective for some types of surface decontamination work. They are particularly effective in decontaminating surfaces covered with oil or grease. High-pressure water jets utilize a high-pressure, positive-displacement pump with a hand-held lance or gun for delivery. The selective use of nozzles can improve decontamination efficiency by matching nozzle flow patterns to specific work tasks. High-pressure water jet systems can produce pressures as high as 100 MPa. The high impact force of these units makes them effective tools for the removal of hard-to-remove contamination, and decontamination factors as high as 1000 may be obtained. Water jets can be turret-mounted to reduce operator fatigue during long hours of use and can be used in conjunction with the holdback-carrier technique discussed previously. Disadvantages of these units are the large volumes of contaminated water produced and the difficulty in controlling the scattering of contaminants released from surfaces by this method.

Steam-jet decontamination is similar to water-jet decontamination. Water or reagents are introduced into an injector system and mixed with steam. The water-steam mixture is expelled through a nozzle at pressures that are typically less than 7 MPa. The delivery rate is about 0.0015 m^3/sec at 5 MPa.

Vacuuming

In areas where dust has accumulated, dry vacuuming is an effective method for the removal of contamination. The vacuuming involves the use of a specially equipped machine with roughing and HEPA filters in the exhaust stream for the retention of radioactive particles. Worker exposure from airborne radioactivity may be increased by the vacuuming activity and protective respiratory equipment must be worn. Dry vacuuming does not work well on crusted deposits; therefore, it is used primarily in areas where dust has not been wetted or crusted.

Wet vacuuming can be used to decontaminate areas where contaminants adhere tightly to surfaces. The method involves scrubbing with water and industrial detergents and then vacuuming the resulting solution. The wash solution is filtered and stored in barrels until it can be solidified for disposal.

Janitorial Techniques

Janitorial techniques are hands-on decontamination procedures including sweeping, wiping, wet-mopping, and scrubbing. Remote and semiremote decontamination methods should be used first to the maximum extent possible so as to minimize the radiation dose to workers.

For small quantities of loose contamination on floors or other surfaces, brushing, sweeping, or dry vacuuming is often effective. For more tenacious contaminants, various cleansing compounds are used in combination with hand-wiping and scrubbing techniques. Several proprietary decontamination solutions are available.[a] Ordinary household detergents are quite effective but produce sizable quantities of waste water that may require special processing. Aerosol-type foaming cleansers are effective and eliminate the wastewater problem, but their use produces sizable quantities of contaminated wiping material. Trichloroethylene, Freon-113, and other solvents are effective degreasing agents that can be used to decontaminate equipment surfaces covered with a layer of oil or grease. The use of these solvents generates contaminated organic solutions that must be processed.

Contaminated floors require scrubbing either by industrial floor scrubbers or by hand, followed by wet vacuuming, and possible detergent-cloth wiping. A final reagent/rinse mopping then completes the effort.

Overhead areas may require damp scouring with reagents followed by rinses and cloth wipes. High-elevation work above floors involves the use of bosun chairs, scaffolding, and telescoping platforms to reach all surfaces. The area above the polar crane may be reached by using the crane beams as a staging platform.

(a) Reference 4 contains a list of proprietary decontamination solutions.

Strippable Coatings

This method involves the application and subsequent removal of a strippable coating. As the coating is removed, it adheres to and takes with it the surface contamination. Strippable coatings are commonly applied to decontaminated areas to facilitate subsequent decontamination if recontamination should occur.

Sludge Removal

Post-accident cleanup will include the removal and processing of contaminated sludge from sumps, tanks, and the floors of some areas of the containment building. Hands-on removal of the sludge, using shovels and scrapers, results in high radiation exposure to decommissioning workers. A procedure for sludge removal used at TMI-2[7] involves resuspension of the sludge by flushing or by dislodging it with the use of a high-pressure water jet. The sludge is then collected with a portable recirculating vacuum filter system (RVFS) that vacuums the material and deposits the solids on disposable filter cartridges.

Electropolishing

Electropolishing is an electrochemical process used on metal objects to remove a thin layer of the exterior surface and attached contamination.[11] The process is illustrated schematically in Figure D.1-1. The method commonly employs a tank containing an acid solution as an electrolyte and a low-voltage, high-current electrical source. The phosphoric acid in the decontamination tanks is recirculated through a filter that accumulates much of the contaminated solids removed from the surface. Small tools and parts can be cleaned in a very short time to nondetectable levels, and the process is effective on all surfaces exposed to the electrolyte.

Electropolishing can also be used for in-situ decontamination of the internal surfaces of cylindrical tanks. The electrolyte is sprayed onto the tank wall from nozzles mounted on rotating arms. A power supply provides current to these arms and, via the electrolyte stream, to the tank wall. Contaminated electrolyte is collected at the tank bottom and returned to an electrolyte handling system for cooling, filtration, and radiation monitoring.

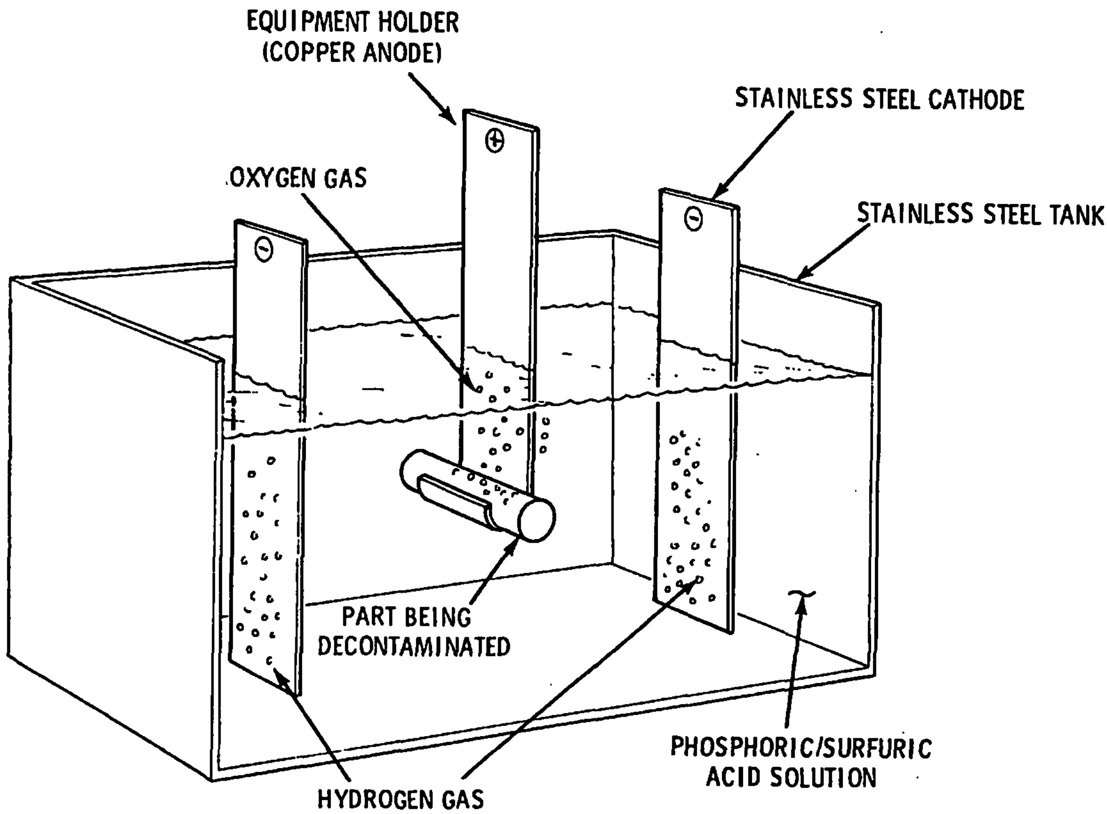


FIGURE D.1-1. Schematic Diagram of Electropolishing Cell

Vibratory Finishing

Vibratory finishing is a surface-finishing technique that employs a vibrating tub of loose media through which flows a liquid chemical compound.[12] The technique is illustrated schematically in Figure D.1-2. The energy from the tub causes the media to scrub the surfaces of the objects being finished, while the liquid compound flushes away the material removed by the scrubbing action. The process is effective on external and internal surfaces, in threads, and in holes. It is being developed as a decontamination technique for processing surface-contaminated metallic and nonmetallic (e.g., plastic) waste. Vibratory finishing will not usually decontaminate objects to the nondetectable levels obtainable with electropolishing. However, it removes essentially all of the smearable contamination as well as contaminated surface scale and corrosion.

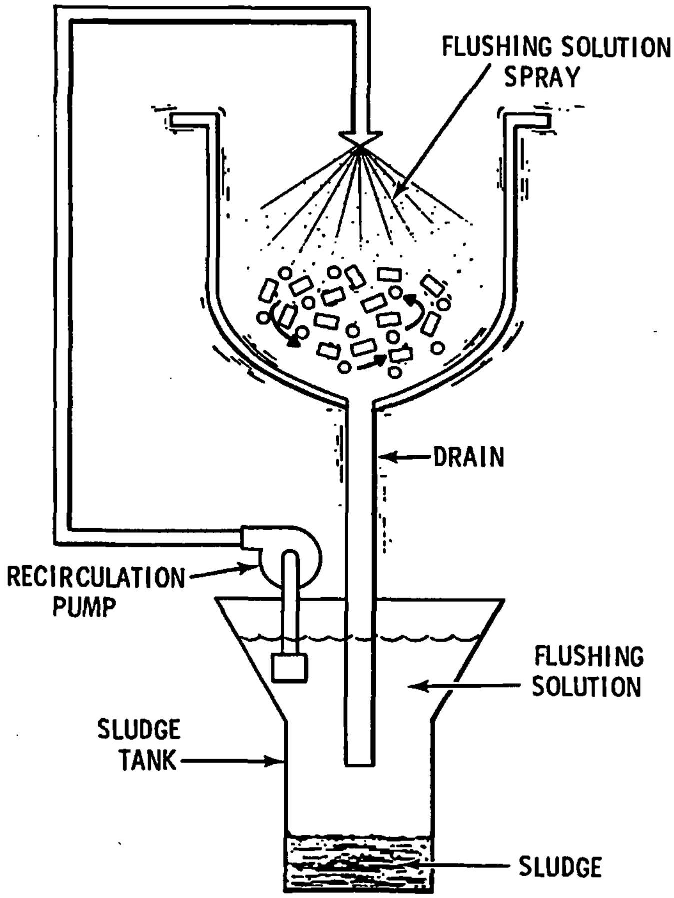


FIGURE D.1-2. Vibratory Finishing System

Ultrasonic Cleaning

For cleaning applications in which small objects are immersed in a chemical or detergent bath, ultrasonics provides an effective method for agitation of the cleaning solution. Ultrasonic cleaning therefore combines the advantages of chemical cleaning and mechanical action. It is particularly suitable for irregularly shaped objects that contain crevices and inaccessible areas.

A typical ultrasonic cleaning system is shown schematically in Figure D.1-3. The system consists of three basic components: a generator, a transducer, and a cleaning tank. The generator converts utility line power at a relatively low frequency of 60 Hz to a more useable form of electrical energy at relatively high frequencies in the range from 18 to 90 kHz. The

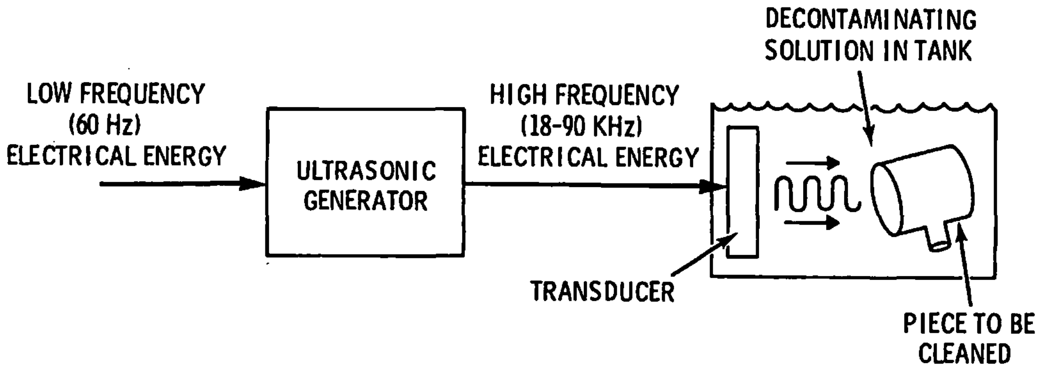


FIGURE D.1-3. Ultrasonic Cleaning System

transducer converts these relatively high frequency electrical impulses to low amplitude mechanical energy of the same frequency--18 to 90 kHz. The cleaning tank contains the liquid cleaning medium through which the mechanical energy is propagated in the form of supersonic waves to the object being cleaned.

D.1.3 Removal of Structural Material

Some concrete in nuclear facilities is contaminated below the surface and cannot be decontaminated to release levels by physical surface cleaning alone. In addition, some of the concrete and structural steel in the biological shield surrounding the reactor vessel is activated as a result of neutron bombardment. In both instances, the structural materials must be physically removed and disposed of during decommissioning.

Several criteria should be considered when selecting a material-removal method for a particular location in the plant. The method chosen should minimize personnel radiation exposure and airborne contamination dispersion. In addition, the size and weight of removed materials should facilitate packaging and shipping for offsite disposal.

The major methods available for concrete surface removal include:

- sandblasting
- vacuum blasting
- jack hammer
- pneumatic or hydraulic impactor
- concrete spaller

- flame cutting
- thermic lance cutting.

A comparison of these surface removal techniques is presented in Table D.1-2.[13] The techniques are discussed in the following paragraphs. The use of controlled blasting with explosives for bulk concrete removal is also described.

TABLE D.1-2. Comparison of Major Concrete Surface Removal Techniques

Technique	Advantages	Disadvantages	Type of Rubble Produced	Size of Air Filtration System Required
Sandblasting	Useful for removing thin layers and paint	Contamination embedded in pores not effectively removed. Generates large quantities of dust and airborne particulates	Small particles	Large
Vacuum blasting	Useful for limiting the spread of dust and abrasive	Vacuum pickup not efficient on irregular surfaces	Small particles	Medium
Jack hammer	Proven technique	Awkward to use on walls. Generates moderate quantities of dust	Medium-size pieces and small particles	Medium
Pneumatic or hydraulic impactor	Proven technique	Limited to large accessible facilities. Generates moderate quantities of dust	Medium-size pieces and small particles	Medium
Concrete spaller	Proven technique. Controlled rate of material removal	Awkward to use on irregular surfaces or in cramped quarters. Cutting through steel rebar slows cutting speed and damages drill	Medium-size pieces and small particles	Small
Flame cutting	Cuts both concrete and steel without difficulty. Adaptable for remote operation	Generates large quantities of toxic gases and smoke. Hot gases can damage HEPA filters, making contamination control difficult	Small particles	Large
Thermic lance cutting	Cuts both concrete and steel without difficulty. Adaptable for remote operation	Generates moderate quantities of toxic gases and smoke. Hot gases can damage HEPA filters, making contamination control difficult	Small particles	Large

Sandblasting

Sandblasting, where the surface is mechanically eroded away, removes only a minimal surface thickness and produces large quantities of small, contaminated particles. Sandblasting primarily removes paint and a little of the concrete surface. It does not effectively remove contamination from the pores in the concrete or from expansion joints. A large exhaust and air filtration system is needed with this method to control contaminated dust. This technique is relatively slow if the contamination penetrates beyond a thin surface layer.

Vacuum Blasting

Vacuum blasting is an abrasive blasting technique in which the nozzle is surrounded by a concentric exhaust air cone to remove the blast dust and abrasive (see Figure D.1-4). The cone through which the debris and spent abrasive is exhausted is connected by a hose to a vacuum system.

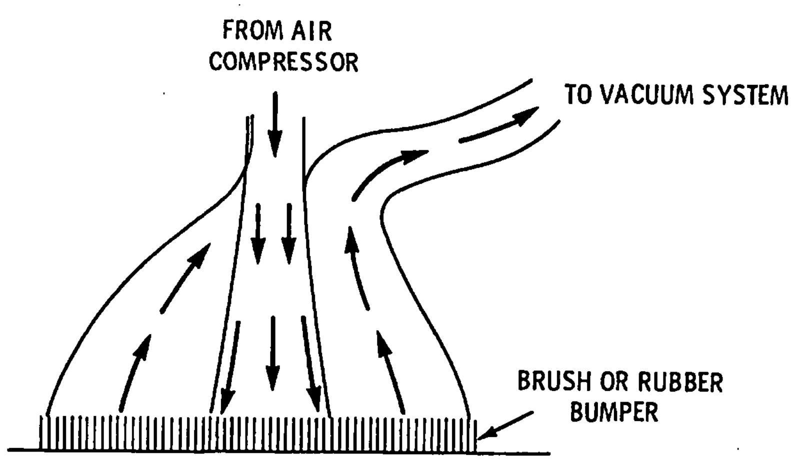


FIGURE D.1-4. Cross Section of a Vacuum Blaster Nozzle

Commercial vacuum blasting units are available that use a cyclone separator to remove the abrasive from the exhaust air stream and that filter the air for collection of the radioactive dust. The abrasive material is reused. Units can be floor-mounted or hand-held, thus allowing either semiremote or hands-on operation. Vacuum blasting is most useful when the unit can be held perpendicular to the surface being cleaned. On irregular surfaces, reflection of abrasives can be a problem and vacuum pickup is less efficient. Decontamination factors of 1000 can be achieved with this technique.

Jack Hammer

Jack hammers, powered by compressed air, are readily available and are easily operated by one man. They are used to chip off the surface material deep enough to remove the contamination. Because they are difficult to position on walls and ceilings, jack hammers are used primarily on floors. A medium-size air filtration system is necessary to control the dust produced by the use of this equipment.

Impactor

Impactors (or hoe rams), similar in operation to jack hammers but much larger, have been used successfully in several decontamination projects.[13,14] An impactor, powered either pneumatically or hydraulically, uses a pick chisel point that is driven into the concrete surface with high-energy impacts several times per second. The use of impactors also requires an air filtration system for dust control.

Concrete Spaller

The concrete spaller, shown schematically in Figure D.1-5, is a lightweight, fully portable rock-splitting tool consisting of a split collar with a sharp triangular ridge around the circumference, mounted on a travelling shaft which has a tapered end. The spaller is operated by inserting the expanding bit into a predrilled hole and activating the device hydraulically, causing the concrete surrounding the bit to be spalled off. Use of the spaller permits localized concrete removal to depths of 50 to 75 mm with no explosions and relatively little dust. (The principal source of the dust is the drilling of the hole into which the splitting tool is inserted.) The hole pattern and the spacing between holes are important parameters in the effectiveness of this technique.

The concrete spaller is selected as the reference device used in this study for removal of contaminated concrete surfaces.

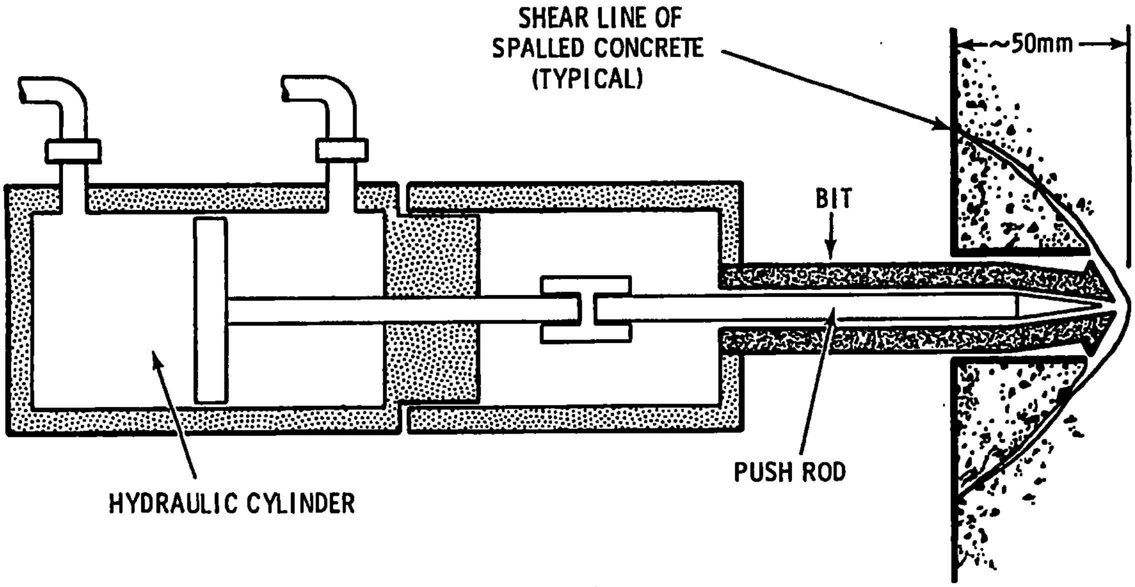


FIGURE D.1-5. Schematic of Concrete Spaller

Flame Cutting

Flame cutting of concrete consists of a thermite reaction process whereby a powdered mixture of iron and aluminum is ignited in an oxygen jet at a temperature of approximately 9000°C, resulting in rapid decomposition of the concrete in contact with the jet. The mass flow through the flame-cutting nozzle clears away the decomposed concrete, leaving a clean kerf. Reinforcing rods in the concrete add iron to the reaction to sustain the flame and assist the reaction.

Flame cutting results in the production of copious quantities of toxic gases and smoke. The gases and smoke may be removed by a squirrel cage blower, and directed through a flexible duct that houses a water fogger to hold down smoke particulate. The high gas temperatures preclude the use of HEPA filters for contamination control, making the flame-cutting technique unsuitable for use on radioactive concrete without precooling the effluent gas.

Thermic Lance Cutting

The thermic lance consists of an iron pipe packed with a combination of steel, aluminum, and magnesium wires through which a flow of oxygen gas is maintained. The thermic lance utilizes a thermite reaction at the tip of the iron pipe, in which the constituents are completely consumed and temperatures in the range from 2500°C to 6000°C are generated. Thermic-lance cutting, like flame cutting, results in the production of large quantities of smoke and hot gases.

Controlled Blasting with Explosives

Controlled blasting is ideally suited for demolition of massive or heavily reinforced, thick concrete sections. The process consists of drilling holes at preselected locations in the concrete, loading the holes with explosives, and detonating using a delayed firing technique. The method is well suited to the removal of activated concrete in large structures such as the biological shield that can be enclosed within ventilation-confinement envelopes to effectively control the spread of dust and airborne contamination. Placement of blasting mats over the affected region prevents flying debris from penetrating the confinement envelope. Fog sprays of water, typically used from 1 minute before to about 15 minutes after blasting, help settle the dust from the explosion. Although blasting sequences are designed to minimize air pressure surges, the ventilation enclosures must be designed to withstand those pressure surges that do occur. Similarly, attention must be given to the ventilation system to close dampers during blasting to prevent surge damage to filters.

Various types of explosives are available for use in demolition applications. The selection of the best type of explosive and of the appropriate blasthole design for a given demolition application should be done by a qualified blasting expert. In this study, the services of a certified blasting technician are assumed to be retained for the duration of bulk concrete removal activities.

D.2 CONTAMINATION CONTROL

Decontamination of the containment following an accident at an LWR requires large-scale transfers of personnel and equipment between highly contaminated areas and less contaminated areas. Many decommissioning operations, particularly cutting operations required for equipment disassembly and concrete removal operations, have the potential for generating significant amounts of airborne radioactive contamination. To minimize the spread of contamination during these activities and to reduce the hazard to decommissioning workers, effective methods of contamination control are required.

Requirements for contamination control during decommissioning can be categorized as follows:

- isolation of contaminated areas
- local mitigation of contamination sources
- collection of contamination.

Contamination control measures in each of these categories are described in the following subsections.

D.2.1 Isolation of Contaminated Areas

Barriers are used to isolate contaminated areas and to minimize the spread of radioactivity from highly contaminated areas to less contaminated areas.

During accident cleanup activities following a reactor accident, an enclosure may be required on the outside of personnel and equipment hatches to serve as an interface between the contaminated containment and the outside environment. Depending on the severity of the accident and the amount of contamination and extent of damage inside the containment, the requirements for this enclosure might be as follows:

- serve as an interface to permit rigid control of equipment, supplies, and personnel entering and leaving the contaminated zone
- serve as an extended contamination barrier so that contaminated items can be brought out of the containment for examination, decontamination, and packaging.

The requirements for an interface enclosure can range from a simple "greenhouse" as described in the following paragraph to a steel-frame, metal-sided building large enough to accommodate facilities for the decontamination and packaging of contaminated equipment and for the temporary storage of contaminated waste.

One type of barrier commonly used in the nuclear industry to isolate contaminated areas is a "greenhouse." A greenhouse is constructed by covering a framework, usually steel scaffolding or wood frame, with plastic sheeting and sealing all joints. Overlapping flaps of plastic are generally used for the door. The greenhouse is connected either to the plant ventilation system or to a portable system (see Section D.2.3.2), which prevents outward leakage of contamination by drawing a slight vacuum on the greenhouse. Greenhouses can be semipermanent, portable structures that can be moved from one location to another as needed, but are more often temporary confinement structures that are dismantled and discarded after each job.

In many cases, construction of a complete greenhouse is unnecessary. A simple plastic curtain partitioning off one section of a room may be all that is required to isolate a contaminated area. The type and degree of isolation required depends on the equipment or structures involved, the associated level and mixture of radioactive contamination, the ventilation balance (direction of airflow), and the cleanup or decommissioning operation being performed.

D.2.2 Local Mitigation of Contamination Sources

Mechanical or physical measures can be used to limit the spread of radioactive contamination. Two methods that have been successfully used are 1) water sprays to reduce airborne dust dispersion, and 2) painting of contaminated surfaces to prevent smearing.

The wetting of dust with water or other liquids is one of the oldest methods of contamination control and can be very effective if properly used. Water sprays are widely used to control fugitive dust emissions from construction sites. The spraying of water containing detergent (as a wetting agent) has been used in the nuclear industry to reduce dust concentrations in air during waste exhumation operations.[15] To be effective, the liquid application must be designed to blanket the dust source completely and to wet the dust particles thoroughly. Various types, sizes, and patterns of spray nozzles are used, depending on the physical properties of the dust, the type and size of the dust source, and the degree of control desired. Water sprays can be used in combination with other contamination control techniques, and are commonly used for dusty operations such as concrete removal.[14]

Nonflammable, strippable coatings can be used to seal porous surfaces (e.g., concrete) to prevent penetration of contamination into the surfaces. Paint can be used to seal smearable contamination already present on surfaces to prevent subsequent contamination spread.[14] Spraying is generally the easiest and quickest method of application. Painting is especially useful in high-traffic areas, where smearable contamination is likely to be picked up and spread around on shoe covers and equipment wheels.

D.2.3 Collection of Contamination

Collection of radioactive contamination before it can be dispersed (preferably as it is generated) reduces the need for additional decontamination subsequent to some cleanup and decommissioning activities. Various collection methods can be used. Vacuum collection and portable ventilation systems are discussed in this section.

D.2.3.1 Vacuum Collection

Contaminated materials can be collected as they are generated by using vacuum systems. A dust shield with a vacuum attachment can be installed on the tool (e.g., concrete spaller or scrubber) being used. As the contaminated dust is generated, it is drawn into the vacuum system and deposited in a collection drum. The outlet air is filtered (with roughing and HEPA filters) to prevent the collected contamination from being expelled.

Various designs for vacuum collection systems are possible, depending on the required operating characteristics. One such system, shown schematically in Figure D.2-1, is described in Reference 16. This system, originally designed for collection of contaminated soil, uses a standard 0.21-m^3 waste drum to collect the contaminated material. When the drum is filled, it is capped and sealed for disposal. A special, commercially available vacuum lid, employing a cyclone baffle arrangement to enhance dust settling, is modified to accept an inexpensive, disposable roughing filter. A HEPA filter and power/vacuum-blower unit, mounted on a steel pallet, complete the system. The system is reported to be capable of pulling up to $\sim$28 m^3/min of air at 110-mm-Hg vacuum, and is estimated to cost less than $5000.

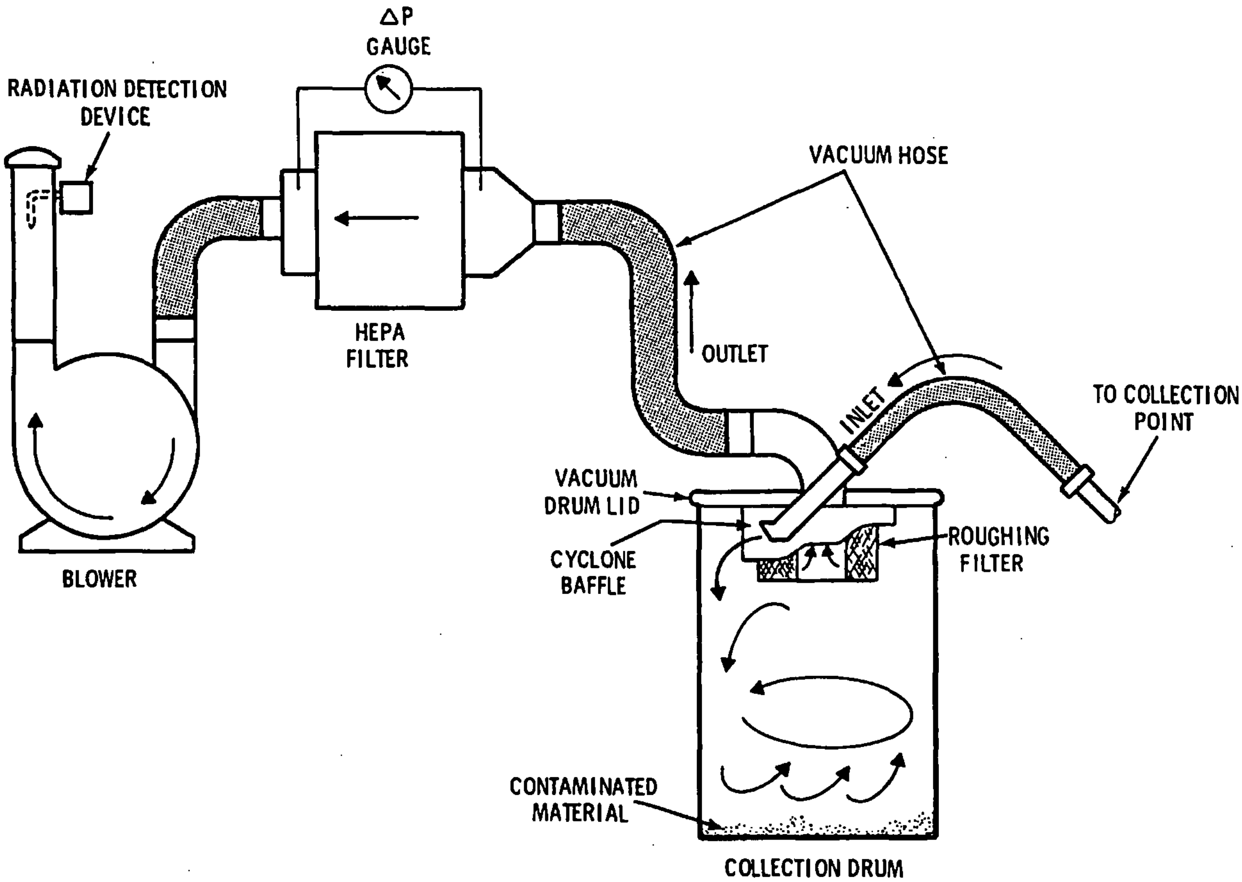


FIGURE D.2-1. Vacuum Collection System Schematic

D.2.3.2 Portable Ventilation Systems

Portable ventilation systems can be used to confine and collect airborne particulates generated during decommissioning operations. General design information concerning such systems is discussed at length in Reference 17. Two portable ventilation systems, a work enclosure and a fume exhauster, are discussed here.

Portable Filtered Ventilation Enclosure. A typical portable filtered ventilation enclosure unit is illustrated in Figure D.2-2. A large squirrel-cage blower is coupled with a high-efficiency particulate air (HEPA) filter preceded by a glass-fiber roughing filter, all mounted on a wheeled cart. A flexible duct couples the cart unit to the enclosure unit that surrounds the work area and confines the materials being emitted. Roughing

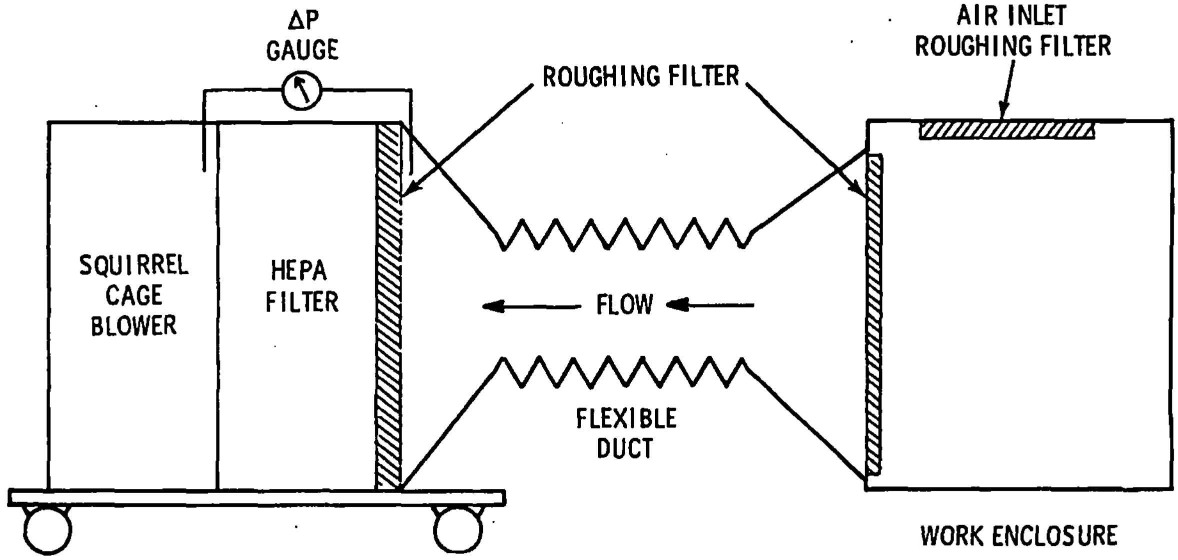


FIGURE D.2-2. Portable Filtered Ventilation Enclosure

filters are installed at both the inlet and the outlet of the enclosure unit. The enclosure unit may have whatever shape best performs the required function at a particular location. A simple, rectangular open-faced box will suffice for many applications.

Radiation detection devices are used to monitor the buildup of radioactive material on the filters. A differential pressure gauge is installed across the HEPA filter to monitor the increasing pressure drop as particulates build up on the filter. Filters are changed when either the dose rate from the collected radioactive particles or the differential pressure across the HEPA filter reaches a predetermined level.

Portable Filtered Fume Exhauster. Another type of portable filtered ventilation system, a fume exhauster, is illustrated in Figure D.2-3. This system has an electrostatic precipitator coupled with a roughing filter, HEPA filter, air-handling motor, squirrel-cage blower, and one or two free-standing intake ducts. The fume exhauster is used to collect radioactive and nonradioactive particulates at the point of generation. This high-volume ventilation system captures all types of particulate matter with efficiencies of greater than 97% for the electrostatic unit and at least 99.95% for the HEPA filter. The advantages of this unit are its portability, its ability to

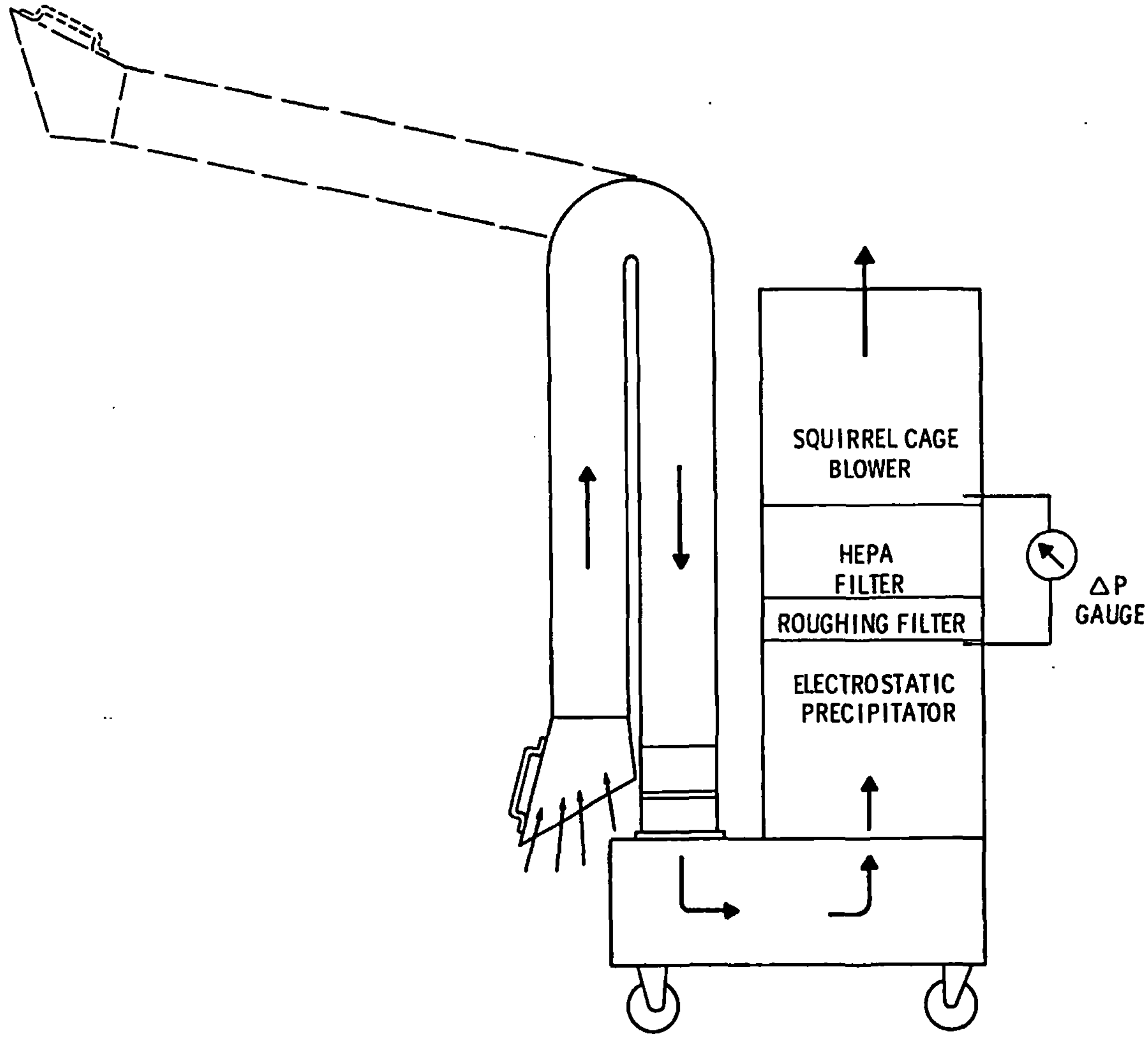


FIGURE D.2-3. Portable Filtered Fume Exhauster

handle large volumes of particulate-laden air, and its generation of relatively small amounts of solid wastes (HEPA filters).

D.3 SPECIAL TOOL AND EQUIPMENT REQUIREMENTS

Special tool and equipment requirements for post-accident cleanup and decommissioning are identified during planning and preparation. Designs and specifications are prepared for each item required. When an item is procured, it is inspected to verify that it meets specifications and complies with applicable Quality Assurance (QA) and safety requirements. It is then tested to ensure that it performs as required. The testing also serves to train personnel in the use of the equipment and to provide data on its operation.

Special tools and equipment items postulated to be needed for accident cleanup and decommissioning of the reference PWR are shown in Table D.3-1. The function of each item is given as well as the number required for accident cleanup and for each decommissioning alternative. Descriptions of special cutting tools (particularly underwater cutting devices) are given in Section F.3 of Reference 18 and in Section G.2 of Reference 19.

D.4 LIQUID WASTE TREATMENT

Alternatives for the treatment of accident-generated water and of decontamination solutions from accident cleanup operations are described in this section.

In studies[18,19] of reactor decommissioning following normal shutdown, the processing of decontamination liquids was assumed to be accomplished using the existing plant radwaste system or a temporary portable radwaste system. Because of the large volumes and high specific activities of the contaminated liquids requiring treatment during accident cleanup, the use of existing plant facilities or small portable systems is generally not adequate for liquid waste treatment. Therefore, auxiliary treatment systems are assumed to be constructed and used. Some alternative liquid waste treatment systems are described in this section.

During accident cleanup, it may be possible to decontaminate the accident-generated water and reuse this water for building decontamination purposes. While this would not reduce the volume of contaminated liquids requiring treatment (the same liquids would undergo treatment more than once), it would reduce the total volume of water requiring disposal. Estimates of requirements and costs of liquid waste treatment made in this study are based on the assumption that treated accident water is reused for decontamination purposes whenever this is practical.

TABLE D.3-1. Special Tools and Equipment for Post-Accident Cleanup and Decommissioning of the Reference PWR

Item	Number of Items Required[a] Post-Accident Cleanup[b]	DECON	SAFSTOR	ENTOMB	Major Function
Underwater manipulator	1	1	0	1	Positioning and movement of underwater cutting devices
Underwater plasma-arc torch[d]	2	2	0	1	Sectioning reactor vessel and internals, steam generators, tanks, and cutting piping, equipment, and structural members; welding
Underwater oxyacetylene torch[d]	2	2	0	1	
Arc saw	1	1	0	1	
Portable plasma-arc torch[d]	2	4	2	4	
Portable oxyacetylene torch[d]	2	2	2	2	
Guillotine pipe saw[d]	4	10	2	10	Cutting piping
Power-operated reciprocating hacksaw[d]	4	10	2	10	Sectioning piping and equiment
Closed-circuit, high-resolution TV systems	2	2	2	2	Observation and control of remote and underwater operations
Underwater lights and viewing aids	AR[c]	AR	AR	AR	Illuminating and observing underwater operations
Underwater tools (e.g., impact wrenches, bolt cutters, tongs)	AR	AR	AR	AR	Underwater disassembly, handling, and packaging operations
Submersible pump with disposable filters[d]	4	4	2	4	Rapid cleanup and draining of pool water
High-pressure water jet	4	2	2	2	Surface decontamination
Scaffolding and safety nets	AR	AR	AR	AR	Access and worker protection for elevated work locations
Shielded vehicle with manipulator arms and interchangeable tools	1	1	1	1	For remote operations in areas with high radiation dose rates
Mobile chemical decontamination unit	0	5	5	5	Decontamination of liquid and solid radwaste equipment
Mobile chemical mixing and heating unit	0	4	4	4	Decontamination of drain systems
Power-operated mobile manlift	2	9	3	9	Safe access to heights
9100-kg mobile hydraulic crane	2	3	0	3	Removal and packaging of contaminated piping and equipment
9100-kg forklift	2	6	1	6	Handling material and loading trucks
Rigging materials (e.g., chokers, grapples, winches)	AR	AR	AR	AR	Handling of piping and equipment

(contd on next page)

TABLE D.3-1. (Contd)

Item	Number of Items Required[a] Post-Accident Cleanup[b]	DECON	SAFSTOR	ENTOMB	Major Function
Concrete drill with HEPA-filtered dust collection system	0	4	1	4	Drilling holes in concrete for blasting or surface spalling
Concrete surface spaller	0	4	1	4	Removal of contaminated concrete surfaces
Front-end loader (light-duty)	0	3	1	3	Cleanup and packaging tasks
Vacuum cleaner (HEPA-filtered)[d]	3	3	3	3	Cleanup tasks
Portable ventilation enclosure[d]	10	10	3	10	Contamination control
Filtered-exhaust fan unit	0	4	0	4	Contamination control
Supplied-air plastic suit[d]	250	250	100	250	Personnel respiratory and body-surface protection from radioactive contaminants
Polyurethane foam generator	0	2	2	2	Contamination control during HVAC work
Paint sprayer	0	0	4	0	Immobilization of contamination
Incinerator	1	1	1	1	Incineration of combustible wastes

(a) Based on a taskwise analysis of the activities involved, plus spares as required.
(b) Numbers are for cleanup following scenario 2 accident.
(c) AR = as required.
(d) Anticipated not to be reusable following post-accident cleanup.

D.4.1 Liquid Waste Sources

Liquid wastes that require treatment and disposal during accident cleanup operations include liquids directly generated during the accident (accident-generated water), and liquids contaminated during the cleanup (decontamination solutions). Accident-generated water includes containment building sump water and reactor coolant system water. Decontamination solutions include water-based decontamination liquids from cleanup of the containment and other contaminated areas, reactor coolant system flush and drain water, and chemical decontamination liquids from building and reactor coolant system decontamination. Volumes of liquid waste estimated to require processing during accident cleanup at the reference PWR following the accidents postulated in this study are given in Table E.4-3 of Appendix E.

Containment Building Sump Water

Sump water is water that is released from the primary coolant system during an accident and accumulates in the lower levels (e.g., the basement and the building sumps) of the containment building. Sump water may constitute a very large volume of water with high concentrations of fission product radionuclides. Filterable solids and oil and grease are also likely components of containment building sump water, and the presence of these materials must be considered in evaluating processing alternatives.

Reactor Coolant System Water

The reactor coolant system contains water contaminated with fission products and reactor core debris from the accident. Because of dilution by makeup water following the accident, the radionuclide concentration in this water may be lower than it is in the sump water. Although removal of contamination from the primary water is necessary to permit safe access to the reactor vessel for defueling, the decontamination process and subsequent makeup must still maintain adequate boron to provide neutron-absorption capability during defueling operations.

Water-Based Decontamination Liquids

Initial decontamination of the containment interior involves the use of water sprays and high-pressure hose washes. A large volume of water may be generated from these operations, but radionuclide concentrations in the water will be relatively low. These wash liquids normally drain down into the containment building sump and could be added to existing accident-generated water for processing.

Reactor Coolant System Flush and Drain Water

Flushing and draining of the reactor coolant system commences following defueling of the core. As much as 1000 m^3 of water may be used for this operation. This water becomes contaminated with radioactive fission products and with fuel debris (UO_2 and activated cladding and structural material).

Chemical Decontamination Liquids

Chemical decontamination operations in the containment building generate chemical waste solutions having a full range of detergents and complexing agents. Chemical decontamination of the reactor coolant system may involve the use of aggressive chemicals such as alkaline permanganate and citrox as well as chelating agents such as EDTA. The volumes of chemical decontamination liquids generated in decontamination of the containment interior are smaller than the volumes of water-based decontamination liquids generated. Chemical and water-based decontamination liquids have similar radioactivity concentrations. The use of chemical decontamination solutions may require special packaging and disposal considerations to meet criteria for the disposal of chelating agents. These disposal criteria are discussed in Section D.5.2.

D.4.2 Liquid Waste Treatment Alternatives

Three alternatives exist for the treatment of liquid wastes from LWR post-accident cleanup operations. These alternatives are:

- interim onsite storage
- direct immobilization using a binder material
- processing of the wastes through treatment systems that utilize filtration, evaporation, and demineralization techniques.

D.4.2.1 Interim Onsite Storage

Onsite storage of accident-generated water and of water-based and chemical decontamination solutions involves the transfer of these liquids to tanks within the plant, if available, or to newly constructed exterior tanks. Onsite storage would be a temporary measure to allow decay of the radioactivity prior to immobilization or processing of the liquid wastes. It could be used in conjunction with the SAFSTOR decommissioning alternative. The feasibility of this alternative would depend on the severity of the accident and on the quantities and specific activities of accident-generated water and decontamination solutions requiring storage. Construction of exterior tanks for the storage of large volumes of liquid waste could be a costly and time-consuming operation. The storage of liquids with high levels of radioactivity would require that the tanks be shielded to reduce radiation levels in areas near the tanks.

D.4.2.2 Direct Immobilization

Direct immobilization involves the fixing of liquid wastes in a solid matrix using a binder material such as bitumen, cement, or vinyl ester styrene (VES). The solidified liquids are packaged in 0.21-m^3 steel drums or in disposable steel liners and either temporarily stored onsite or transported to a commercial shallow-land burial facility for disposal. Processes that use binder materials to immobilize liquid waste to make it acceptable for transport and disposal are described in Appendix H of Reference 7. High levels of radioactivity are present in accident water and in some decontamination liquids from the cleanup and decommissioning of reactors following serious accidents. Immobilization of these liquids would require the construction of a waste immobilization and handling facility that could be remotely operated.

Immobilization of accident water from a severe (scenario 2 or scenario 3) accident would result in a very large number of containers with high surface dose rates. For example, immobilization of the sump water from the scenario 3 PWR accident (1600 m^3 of water containing 2.5×10^6 Ci of radioactivity) would require the processing of about thirteen thousand three hundred 0.21-m^3 steel drums (assuming 0.12 m^3 of liquid waste per drum). The radioactivity per drum would be approximately 200 Ci and the drum surface radiation level would be about 160 R/hr.

Direct immobilization without evaporation can be used for the treatment of relatively low-volume, low-specific-activity chemical decontamination solutions generated during post-accident decontamination operations.

D.4.2.3 Waste Processing

Processes for the treatment of liquid wastes utilize filtration, evaporation, and ion exchange.

Filtration is a physical process in which particles suspended in a liquid are separated from it by forcing the liquid through a porous medium. As the liquid passes through the porous material, the suspended particles are trapped on the surface of the medium or within the body of the medium itself. Loading of the filter medium requires removal of the filter cartridge and its replacement with a fresh one. Filtration is used as a first step in the processing of liquid wastes. It is not an appropriate treatment process in and of itself because much of the radioactivity is in solution and cannot be removed by filtering.

Evaporation is the removal of a relatively volatile component from a solution by boiling the solution. The contaminants are retained in the concentrated solution (the evaporator bottoms) while the relatively clean volatiles are vaporized and then condensed as processed water or distillates. Evaporation is appropriate for treatment of liquid wastes with low to moderate concentrations of dissolved solids and for chemical decontamination solutions.

Ion exchange involves the removal of ionic species from an aqueous phase. A well-known application of ion exchange is "water softening"--the substitution of sodium ions for calcium and magnesium ions in water to reduce its hardness. For the removal of radioactive contaminants from liquid waste, both natural and synthetic zeolites (aluminosilicate minerals) and synthetic organic resins are used. The ion exchange material is placed in demineralizer vessels through which the liquid containing the ionic contaminants flows. As the radioactive liquid comes in contact with the ion exchange media, specific ions are preferentially removed.

Vessels containing ion exchange media are referred to as zeolite liners, cation liners, and mixed-bed liners (liners containing both anion and cation organic resins). These designations refer to the type of ion exchange material in the vessel. Zeolites are better suited than organic resins for the cleanup of liquid wastes containing high concentrations of radioactivity because of the greater radiation stability of inorganic ion exchange media.

Ion exchange is appropriate for the treatment of accident water and water-based decontamination solutions. It is not appropriate for the treatment of chemical decontamination solutions that contain high concentrations of chemicals and detergents. Treatment of chemical decontamination solutions with ion exchange media leads to chemical breakdown and plugging of the media. These solutions are normally treated by evaporation or are immobilized directly in a solidification facility.

Two ion exchange systems for the treatment of accident water and water-based decontamination solutions have been used to treat contaminated liquids during cleanup operations at TMI-2.[7] These systems are the EPICOR-II system, which has been successfully used to process contaminated water from the auxiliary and fuel handling building (AFHB), and the submerged demineralizer system (SDS), a zeolite-based system used to process containment building sump water, which contains much higher concentrations of radioactivity than the AFHB water. The systems are described in References 6, 7, and 20.

D.4.3 Disposal of Processed Water

Processed water is the liquid effluent from the treatment of accident water and water-based decontamination solutions. The principal radioactive constituent of processed water is tritium (12.8-year half-life), which is not removed by any of the treatment processes described in Section D.4.2. Alternatives for the disposal of this water include:

- onsite storage
- solidification and offsite disposal at a shallow-land burial ground
- controlled release to the river
- controlled release to the atmosphere via evaporation.

Onsite storage would be a temporary measure that might be employed to allow tritium 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.) This alternative might also be required if other alternatives are unavailable or unacceptable. Onsite storage would require the construction of tanks with a capacity to contain the water from processing operations. A criterion for processed water storage would be that the content of radioactivity stored in each tank should be limited such that a tank failure would not result in greater than 10 CFR Part 20 (Table II, Col. 2) concentrations at the nearest drinking water intake.

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).

There are no technical or federal regulatory obstacles to implementing the offsite disposal alternative. There are, however, potential state restrictions on the use of low-level waste disposal facilities to dispose of this material. There is currently a shortage of shallow-land burial space, and restrictions might be imposed on the use of this space for the disposal of relatively innocuous tritiated waste. In addition, the logistics of shipping 21,000 steel drums might discourage the use of this alternative. Assuming 120 drums per shipment, 175 truck shipments would be required to transport this material.

The controlled release of processed water to the nearby river might be used to dispose of this water. Criteria governing the potential discharge of radioactive effluent to the river include: 1) the requirements of 10 CFR Part 20, 2) 10 CFR Part 50 Appendix I criteria for offsite radiological exposure, 3) Clean Water Act criteria related to EPA's Primary Drinking Water standards,[21] 4) state and local ordinances restricting discharges to the river, and 5) limitations imposed by the amended, possession-only facility license. In the event the controlled-release alternative is used, specifications for discharge rates and dilution factors would be established and monitoring procedures would be implemented to ensure that applicable criteria that limit radionuclide concentrations in the river are not exceeded.

The controlled release of processed water to the atmosphere can be effected by natural or forced evaporation. Natural evaporation consists of placing the water in a lined pond. The water is allowed to evaporate and vapor containing tritium is released to the atmosphere. Natural evaporation can be improved by heating the water entering the pond or by the use of a spray system. Forced evaporation utilizes one or more of the cooling towers on the plant site. To control the tritium release from forced evaporation, the cooling tower could be operated intermittently under favorable meteorological conditions. Tritium releases from the forced evaporation alternative require control to conform to the levels in 10 CFR Part 20 Appendix B for operating personnel and to the dose criteria in 10 CFR Part 50 Appendix I for offsite radiological exposure.

D.5 PACKAGING, TRANSPORTATION, AND DISPOSAL OF WASTES

The cleanup and decommissioning of an accident-damaged reactor results in significant quantities of radioactive wastes requiring treatment, packaging, and disposal. These wastes result from the accident and from decontamination and disassembly operations. The management of these radioactive wastes is described in this section.

D.5.1 Waste Characterization

The types of waste generated in accident cleanup and decommissioning operations are shown in Figure D.5-1. The waste is characterized as primary waste, which is the form of the radioactive material at the time it is generated, and secondary waste, which is waste generated by treatment of the primary waste. Major factors governing the management of this waste are its physical and chemical forms. Some important characteristics of the waste types identified in Figure D.5-1 are briefly described in the following paragraphs.

Accident Gases

The radioactive fission product gases are mostly short-lived (e.g., ^{88}Kr, ^{133}Xe, and ^{135}Xe) or will have been removed from containment (e.g., ^{85}Kr and ^{89}Kr) at the time of entry into the containment to begin cleanup operations. Gases are not further considered in this section.

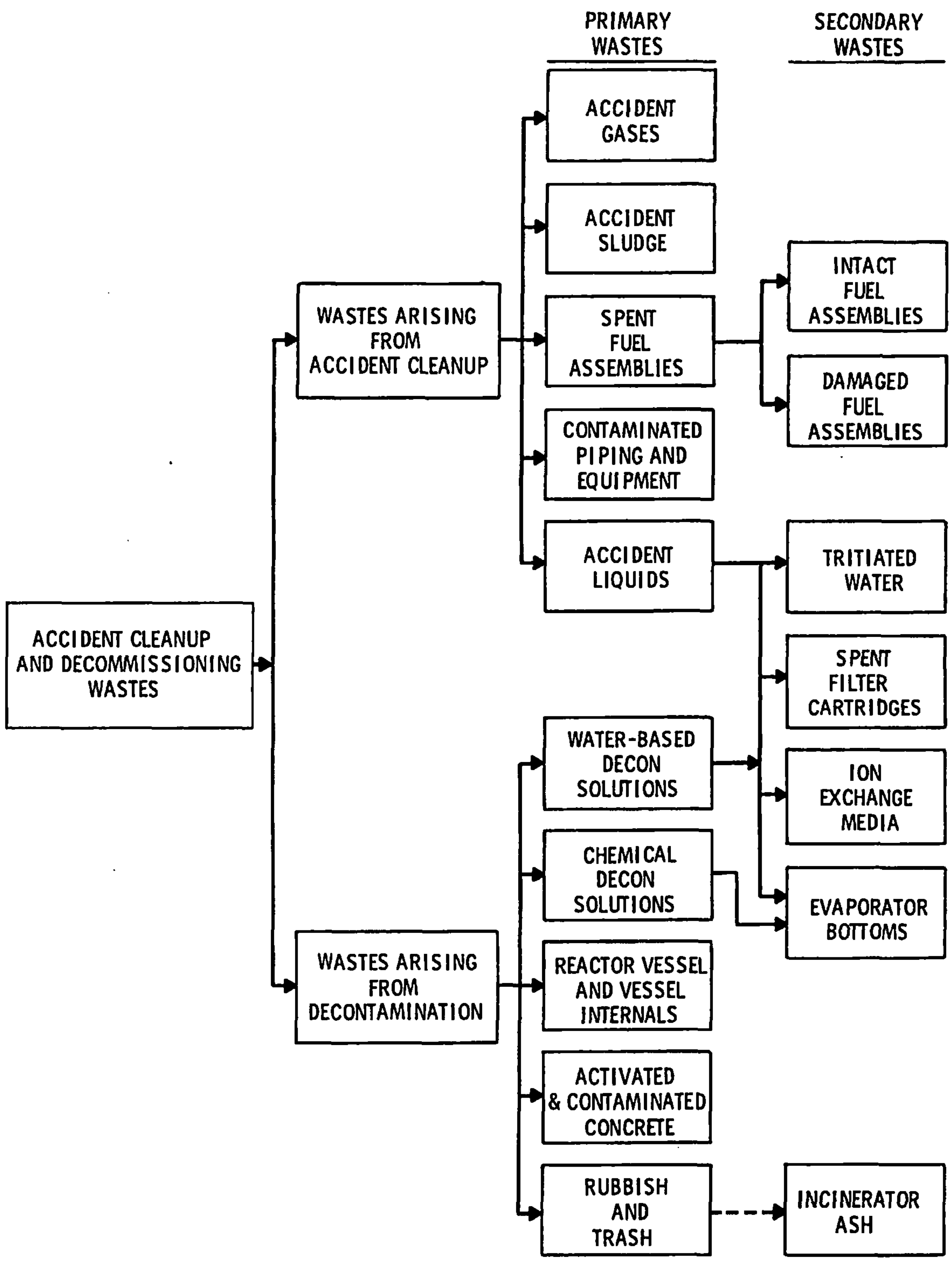


FIGURE D.5-1. Characterization of Waste Forms from Accident Cleanup and Decommissioning

Accident Sludge

For some accident scenarios, radioactive sludge is present in the containment building sump and in tanks that contain contaminated accident water. The chemical and physical characteristics of the sludge are relatively complex and variable. The sludge is contaminated with fission products and represents a relatively low-volume, high-specific-activity form of waste requiring solidification prior to disposal.

Spent Fuel Assemblies

The irradiated fuel assemblies in the reactor core must be removed from the reactor vessel before decontamination of the reactor coolant system can proceed. For some accident scenarios, these fuel assemblies will be damaged. The amount of cladding failure, fuel melting, or cracking and crumbling of fuel material depends on the severity of the accident. To limit the further dispersal of fission product contamination, damaged fuel assemblies are packaged in stainless steel canisters as soon as they are removed from the reactor vessel.

Contaminated Piping and Equipment

Some of the piping and equipment contaminated during normal operation or as a result of the accident cannot be decontaminated and requires packaging and disposal as radioactive waste. Equipment items requiring disposal include tanks, motors, pumps, valves, filters, instrumentation, and other components. Contact exposure rates for these materials range from a few mR/hr to thousands of R/hr.

Accident Water

Accident water includes water that collects in the containment building basement (sump water) and contaminated reactor coolant system water. This water contains tritium, radioactive fission products, sodium, and boron. The sump water also includes sludge and suspended and dissolved solids and colloids. The reactor coolant system water may contain fuel particles and transuranics. Processing of accident water using ion exchange, evaporation, or filtration techniques leads to the generation of secondary waste forms consisting of tritiated water, process solids in the form of loaded ion exchange materials and evaporator bottoms, and solid waste in the form of spent filter cartridges.

Water-Based Decontamination Solutions

Post-accident decontamination procedures include the use of water to wash down internal building surfaces and equipment. This water becomes contaminated with the radioactive materials it washes off the surfaces. The chemical and physical characteristics of the water depend on the contamination levels of the surfaces washed, the procedure used for application of the water, and the extent to which detergents are used. In general, water-based decontamination solutions contain suspended solids, fission products, chemical contaminants, and detergents. The solutions represent a relatively large volume of low-specific-activity liquid waste. Processing of these solutions results in the generation of secondary waste forms, including spent filter cartridges, loaded ion exchange materials, and evaporator bottoms.

Chemical Decontamination Solutions

Post-accident decontamination operations also use chemical solutions (e.g., strong detergent solutions or foam-type decontamination agents). The effectiveness of these solutions may be greater than that of water-based solutions so that the resultant liquid may have higher specific activities than would water-based solutions. Processing of these solutions using evaporation results in the generation of secondary waste in the form of evaporator bottoms.

Reactor Vessel and Vessel Internals

The reactor vessel and the structural hardware and other components within the reactor vessel become radioactive as a result of neutron activation during normal operation of the reactor and are also contaminated with fission products released from damaged fuel during the accident. These wastes represent a low- to relatively high-activity source of solid waste material with contact exposure rates that range from a few mR/hr to thousands of R/hr. Their removal during post-accident cleanup and decommissioning may be complicated as a result of damage sustained during the accident.

Activated and Contaminated Concrete

This material includes activated and contaminated concrete from the biological shield and contaminated concrete from the floors and walls of the containment structure. The concrete is removed by chipping and blasting and the rubble material ranges in size from fine particles to moderate-sized chunks. It has a relatively low specific activity and can be shipped in unshielded containers (usually steel or plywood boxes).

Rubbish and Trash

Radioactive trash generated during post-accident decommissioning operations consists of compactible and noncompactible solid material, some of which is also combustible. The compactible and combustible solids consist of disposable clothing, rags, plastic covers, laydown pads, and miscellaneous trash. Noncompactible solids consist of tools, hoses, safety goggles, miscellaneous construction materials, and other small items of equipment used by decommissioning personnel. The form and specific activity of the solid waste generated by post-accident cleanup and decommissioning crews is comparable to the solid waste generated during decontamination operations at operating LWRs and other nuclear facilities.

Tritiated Water

Tritiated water is accident water that has been processed for the removal of radioactive fission products but still contains the tritium originally present in the liquid. This water may be held in onsite storage tanks for decay of the tritium or disposed of as described in Section D.4.3.

Spent Filter Cartridges

Spent filter cartridges are a form of secondary waste arising from the treatment of radioactive liquids. Filter cartridge assemblies are typically right-circular cylinders that are used to remove particulates from liquid waste; the contaminated particulates are deposited on the filter. They represent a low- to very-high-specific-activity form of waste, with their specific activity dependent on the contaminants in the waste stream that is processed.

Ion Exchange Media

The use of ion-exchange media in the form of organic resins or zeolites to remove fission product contaminants from liquid results in the generation of process solids in the form of contaminated ion exchange materials. The specific activities and radionuclide contents of the loaded ion exchange media will vary with the liquids processed, the contaminants removed from the liquids, and the capacity of the specific media to retain these contaminants. Loaded ion exchange media represent a relatively high volume of process solid waste with specific activities in the low- to very-high range relative to expended ion exchange media generated during decontamination of operating LWRs.

Evaporator Bottoms

The use of evaporation techniques to reduce liquid waste volumes results in the generation of process solids in the form of evaporator bottoms or sludges. The physical characteristics of these process solids depend on the solids content of the liquids evaporated and the equipment used for evaporation. These characteristics can range from slurries containing 10-20 wt% solids to sludges with solids contents in excess of 50 wt%. The specific activities of these process solids also vary over a wide range. Evaporation of decontamination solutions can lead to relatively high concentrations of chelating agents (e.g., EDTA) in these process solids.

Incinerator Ash

Incinerator ash is produced as a result of the incineration of combustible trash to reduce its waste volume. Incineration results in a volume reduction of about a factor of 50 to 100 with a corresponding increase in the specific activity of the ash.

D.5.2 Alternatives for Waste Management

Waste management alternatives for the wastes from post-accident cleanup and decommissioning, described in Section D.5.1, are shown in Tables D.5-1 through D.5-3. Management of these wastes includes treatment or conditioning as necessary to solidify the wastes and reduce their volumes, packaging (discussed in Section D.5.3), followed either by onsite storage or by shipment

TABLE D.5-1. Waste Management Alternatives for Solid Wastes from Accident Cleanup and Decommissioning

Waste Management Alternative	Waste Type: Spent Fuel Assemblies	Contaminated Piping and Equipment	Reactor Vessel and Vessel Internals	Activated and Contaminated Concrete	Rubbish and Trash
Treatment					
Compaction					X
Incineration					X
Disassembly/sectioning		X[a]	X		
Packaging					
0.21-m^3 steel drum				X	X
Disposable steel liner		X	X		
Plywood box		X		X	X
Special container	X[b]	X[c]			
Shipment					
Unshielded		X	X	X	X
Shielded	X	X	X		
Storage or Disposal					
Interim onsite storage	X	X	X	X	X
LLW burial ground		X	X	X	X
Interim storage at federal repository		X	X		
ISFSF fuel storage	X				
Deep geologic disposal[d]	X	X	X		

(a) X denotes alternative considered for waste type.
(b) Requires special liner designed for spent fuel cask.
(c) Some equipment items are packaged by capping the piping connections and using the equipment outer shell as the container.
(d) This alternative not currently available.

TABLE D.5-2. Waste Management Alternatives for Liquid Wastes from Accident Cleanup and Decommissioning

Waste Management Alternative	Waste Type: Accident Water	Waste Type: Water-Based Decon Solution	Waste Type: Chemical Decon Solution
Treatment			
Filtration	X[a]	X	
Ion Exchange	X	X	
Evaporation	X	X	X
Conditioning			
Immobilize/cement	X	X	X
Immobilize/vinyl ester styrene	X	X	X
Packaging			
0.21-m^3 steel drum	X	X	X
Disposable steel liner	X	X	X
Shipment			
Unshielded	0[b]	X	X
Shielded	X	X	X
Storage or Disposal			
Interim onsite storage	X	X	X
LLW burial ground	X	X	X

(a) X denotes alternative considered for waste type.
(b) 0 denotes alternative for tritiated water only.

of the wastes to an offsite storage or disposal facility. Shipments may be made in shielded or unshielded containers depending on package surface exposure rates.

Onsite storage of radioactive wastes is assumed to be a temporary measure, since it is unlikely that a nuclear power plant site could qualify as a permanent waste repository because of such factors as nearby population densities and hydrology. Temporary onsite storage of wastes may be necessary if adequate facilities for permanent offsite disposal are not available.

TABLE D.5-3. Waste Management Alternatives for Process Solids from Accident Cleanup and Decommissioning

Waste Management Alternative	Waste Type: Accident Sludge	Spent Filter Cartridges	Ion Exchange Media	Evaporator Bottoms	Incinerator Ash
Conditioning					
Dewatering	X[a]		X		
Immobilize/cement	X		X	X	X
Immobilize/vinyl ester styrene	X		X	X	
Packaging					
0.21-m^3 steel drum	X	X	X	X	X
Disposable steel liner	X	X	X	X	
Shipment					
Unshielded					
Shielded	X	X	X	X	X
Storage or Disposal					
Interim onsite storage	X	X	X	X	X
LLW burial ground	X	X	X	X	X
Interim storage at federal repository	X	X	X		
Deep geologic disposal[b]	X	X	X		

(a) X denotes alternative considered for waste type.
(b) This alternative not currently available.

Onsite storage of some wastes is likely if either SAFSTOR or ENTOMB is chosen as the alternative for decommissioning the plant after accident cleanup operations are completed.

Most radioactive wastes from post-accident cleanup and decommissioning operations are assumed to be disposed of by shallow-land burial. The NRC has amended its rules in Title 10 of the Code of Federal Regulations to add a new

Part 61[10,22] which provides licensing procedures, performance objectives, and technical requirements for the issuance of licenses for the land disposal of "low-level" radioactive waste. A waste classification system has been developed and incorporated in Part 61 for the purpose of defining waste concentrations and packaging and disposal requirements so that the health and safety of the public and the long-term protection of the environment is not compromised as a result of shallow-land burial operations.

Three classes of wastes are defined by Part 61 requirements:

1. Class A wastes are wastes for which there are no stability requirements but which must be disposed of in a segregated manner from other wastes. These wastes are defined in terms of maximum allowable concentrations of certain isotopes and minimum requirements on waste form that are necessary for safe handling.
2. Class B wastes are wastes which need to be placed in a stable waste form and disposed in a segregated manner from unstable waste forms. These wastes are defined in terms of requirements for stable waste form as well as in terms of allowable concentrations of isotopes and minimum handling requirements.
3. Class C wastes are wastes which need to be placed into a stable waste form, disposed in a segregated manner from nonstable waste forms, and disposed of so that a barrier is provided against potential inadvertent intrusion after institutional controls have lapsed. These wastes are defined in terms of allowable concentrations of isotopes and requirements for disposal.

Maximum allowable waste concentrations for the three waste classes are shown in Table 1 of 10 CFR Part 61.55. Wastes containing concentrations higher than the upper limits specified in the table would be generally unacceptable for near-surface disposal. The disposal of such wastes would be subject to case-by-case determinations depending on the specific waste forms and disposal techniques.

The physical and chemical characteristics and the packaging requirements for wastes that are considered acceptable for disposal by shallow-land burial are also defined in Part 61. Table 1 of 10 CFR Part 61.55 indicates that radioactive wastes containing chelating agents in concentrations greater than 0.1% are not permitted for near-surface disposal except as specifically approved by the NRC.

The regulations in 10 CFR 61 can have a direct effect on the choices of decontamination and waste processing methods employed for post-accident cleanup and decommissioning. Choices should ensure that wastes intended for shallow-land disposal meet the waste characteristics, radioisotope concentration limits, and packaging requirements set forth in the regulations.

Disposal requirements for spent fuel, the reactor vessel and vessel internals, process solids from the treatment of radioactive liquids, or other highly radioactive or long-lived wastes are not defined at the time this study is being prepared. Some of these waste materials may not be suitable for disposal by shallow-land burial, and a requirement for deep geologic disposal or some other form of disposal may be forthcoming. At the present time only shallow-land burial grounds for the disposal of low-level wastes exist. There are no deep geologic disposal facilities, and acceptance criteria for proposed facilities are not well defined at this time. Interim storage at a federal repository may be required for wastes that cannot be disposed of by shallow-land burial. Independent, away-from-reactor, spent fuel storage facilities (ISFSF) would be one way of storing spent fuel on an interim basis. Similar temporary facilities might be used for long-lived and highly radioactive reactor vessel internals.

The waste management alternatives chosen in this study for purposes of evaluating the costs of cleanup and decommissioning, based on alternatives listed in this section, are described in Appendix E for accident cleanup operations and in Appendix H for decommissioning operations.

D.5.3 Packaging of Radioactive Wastes

A variety of containers are used for the packaging of radioactive wastes from accident cleanup and decommissioning operations. Details of packaging requirements are given in subsequent appendices where the specific cleanup and decommissioning alternatives are described.

Trash and contaminated equipment and hardware of low specific activity are packaged and shipped in 0.21-m^3 steel drums and in plywood boxes. Solidified liquids are packaged and shipped in 0.21-m^3 steel drums or in disposable steel cask liners.

Disposable steel cask liners are used for packaging and shipping the bulk of the activated materials from the reactor vessel and vessel internals. Specially constructed steel boxes are used where size and radiation exposure considerations make packaging in cask liners unfeasible. In some cases, lead shielding must be added to the packages to reduce the surface dose rates of the containers to acceptable limits. In other cases, less-activated component pieces are used to surround the more activated pieces to provide the required shielding without sacrificing part of the container volume.

Where external contamination levels allow, certain equipment items (e.g., heat exchangers and small tanks) are packaged by capping the piping connections with welded metal covers and using the items' outer shells as the containers. Larger items, such as steam generators, may be cut into sections, after which each section is capped and handled as its own container.

The liners containing spent filters and zeolite and organic resins from the submerged demineralizer system used to process accident water (see Section E.4.1 for a description of the demineralizer system) serve as their own primary containers for the shipment of these wastes. Because of high surface radiation dose rates, these liners are placed inside lead casks for shipment.

D.5.4 Shipment of Radioactive Wastes

To facilitate comparisons with earlier studies[18,19] of reactor decommissioning following normal shutdown, in this study the undamaged spent fuel and irradiated fuel channels are assumed to be shipped by rail to an independent spent fuel storage facility (ISFSF). Damaged fuel is assumed to be shipped by rail to a federal repository for examination and/or interim storage. All other wastes are assumed to be shipped by exclusive-use truck to a shallow-land burial ground for disposal or to a federal repository for treatment and interim storage. The ISFSF, the shallow-land burial ground, and the federal repository are all assumed to be located 1600 km from the site of accident cleanup and decommissioning operations.

Rail shipment of spent fuel is standard operating practice at many operating reactors including the reference plants. References 23 and 24 discuss rail shipment of spent fuel.

Truck shipment of radioactive wastes is assumed to be a contracted activity performed by a transportation company with special equipment for handling radioactive waste and specially trained drivers. Several commercial companies with these capabilities exist. The hauler is assumed to have the appropriate NRC license and permits from the Department of Transportation (DOT) before he handles the radioactive waste material.

All shipments of radioactive material must be made in compliance with federal, state, and local regulations as discussed in Chapter 5 of Volume 1. Federal (DOT and NRC) transportation regulations establish container specifications, dose rate limits, and handling procedures to ensure the safety of the public and of the transportation workers during shipment of radioactive materials. In addition, for highway transport, state agencies regulate vehicle sizes and weights and, in some cases, transportation routes and times of travel.

Dose rates for highway shipments in exclusive-use vehicles must not exceed the following values (49 CFR 173.393):

- 1000 mrem/hr at ~1 m (i.e., 3 ft) from the external surface of a package shipped inside a closed vehicle
- 200 mrem/hr at any point on the external surface of a closed vehicle or an exposed shipping container (e.g., a shielding cask)
- 10 mrem/hr at 2 m from the external surface of the vehicle
- 2 mrem/hr at any normally occupied position in the vehicle.

These dose rate limits are illustrated in Figure D.5-2 for closed truck transport.[25] All of these criteria must be met for each shipment.

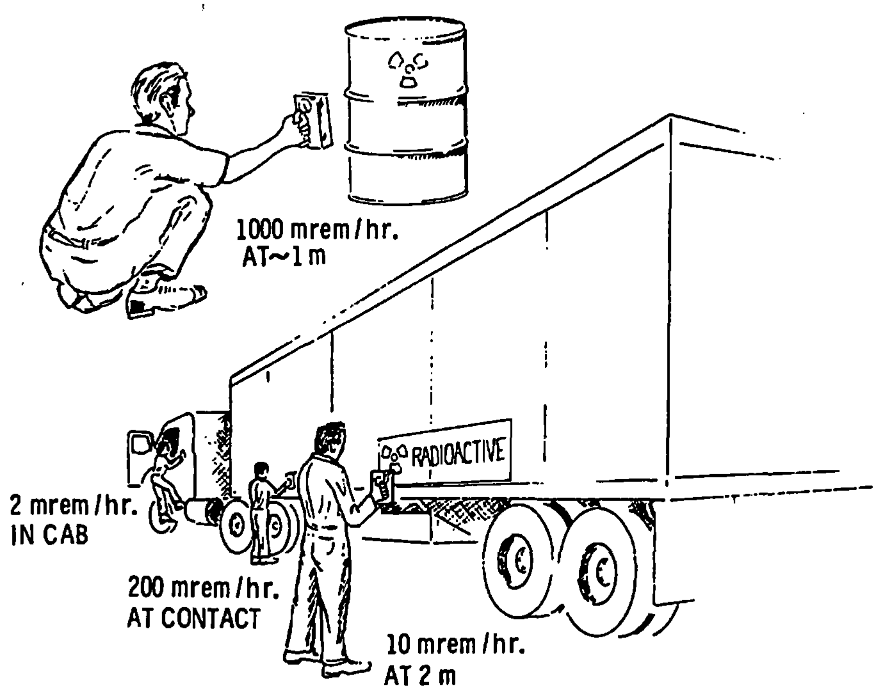


FIGURE D.5-2. Radiation Dose Limits for Closed Exclusive-Use Vehicles (from Reference 25)

D.6 ESSENTIAL SYSTEMS AND SERVICES

All or parts of certain facility systems and services must be available during accident cleanup and decommissioning operations until all radioactive material is either removed or secured in place, to prevent the release of significant quantities of radionuclides to the environment. Some systems and services are required for cleanup and disassembly activities. Other systems provide health and safety protection to the decommissioning workers and the public. If these essential systems and services have been lost or damaged as a result of the accident, they must be repaired or replaced with temporary systems and services before cleanup and decommissioning can proceed.

The essential systems and services are listed in Table D.6-1 together with the justification for their retention during accident cleanup and decommissioning operations.

TABLE D.6-1. Essential Systems and Services for Accident Cleanup and Decommissioning

System or Service	Justification
Electrical Power	Operation of electrical equipment, including HVAC, lighting, and radiation monitoring
HVAC Systems	Ventilation and radioactive contamination confinement
Water Supply (service and domestic systems)	Decontamination cleanup, fire protection, and potable water
Fire Protection System	Health and safety
Compressed Air Systems (control and service)	Operation of pneumatic controls and tools; personnel fresh air supply
Communications Systems	Facilitate and coordinate decommissioning activities
Radiation Monitoring Systems	Personnel safety considerations
Radwaste Systems	Treatment of radioactive liquids and solids
Spent Fuel Cooling and Cleanup System	Cleanup and cooling of water in spent fuel storage pool while spent fuel is there, and during defueling and reactor vessel/internals removal
Closed Cooling Water Systems	Secondary cooling of other systems
Chemical Feed System	Radwaste handling, water demineralization, and reactivity control in the reactor core
Fuel Oil System	Auxiliary power
Security Systems	Public safety and plant protection considerations.

For decommissioning by the SAFSTOR or ENTOMB alternatives, certain systems and services are required during the continuing care period. These include:

- electrical power
- radiation monitoring systems
- security systems.

D.7 QUALITY ASSURANCE

Quality assurance (QA) is an important part of the accident cleanup and decommissioning effort. A QA program consists of all of the programmed events necessary to ensure that cleanup and decommissioning activities are performed in accordance with established procedures, that proper safety considerations are observed, and that adequate documentation is maintained. During the planning and preparation phase, as detailed procedures are developed, QA portions are included.

Regulations and guidance pertaining to QA in the construction and operation of nuclear power plants are contained in several documents, including:

- 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"
- Regulatory Guide 1.58, Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel for the Construction Phase of Nuclear Power Plants
- Regulatory Guide 1.88, Collection and Storage of Nuclear Power Plant Quality Assurance Records
- Regulatory Guide 1.143, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants
- NRC's Standard Review Plan,[26] Section 17.1, "Quality Assurance During the Operating Phase."

While these documents do not specifically address cleanup and decommissioning, they do contain guidance on such topics as the design, purchase, and fabrication of equipment, the training of personnel, and the maintenance of QA records--topics that are pertinent to these operations.

The essential components of a QA program for decommissioning are described in detail in Section G.5 of Reference 19.

D.8 ENVIRONMENTAL SURVEILLANCE

Radiological environmental surveillance is of concern during the decommissioning of any nuclear facility and particularly during the cleanup and decommissioning of an accident-damaged nuclear power reactor where operations may be complicated by unusual or unforeseen difficulties. The following objectives are relevant to environmental surveillance for accident cleanup and decommissioning:

- detection of sudden changes and evaluation of long-term trends of (radionuclide) concentrations in the environment, with the intent of detecting failure to adequately control releases and then to initiate appropriate actions
- assessment of the actual or potential exposure of people to radioactive materials or radiation present in their environment, or estimation of the probable upper limits of such exposure
- determination of the fate of contaminants released to the environment, with the intent of detecting previously unconsidered mechanisms of exposure
- demonstration of compliance with applicable regulations and legal requirements concerning releases to the environment.

Methods, procedures, and performance criteria for environmental surveillance are discussed in detail in Reference 27.

The radiological environmental surveillance program on and in the vicinity of the TMI-2 site during decontamination operations is described in Section 11 and Appendix M of Reference 7. The licensee's monitoring program includes the sampling of air, milk, water, fish, aquatic plants, sediments, miscellaneous food products, and exposure rates in the environs on and around the TMI facility out to a distance of about 35 km. The licensee's air sampler network consists of eight stations that are sampled weekly using both air particulate filters and charcoal cartridges. Milk is sampled semimonthly from five farms in the offsite area. Water is sampled semimonthly from eight stations. Fish, aquatic plants, and aquatic sediments are sampled periodically, as well as miscellaneous food products as they become available. Radiation exposure rates in the area around the site are monitored with the use of thermoluminescent dosimeters (TLDs) that are exchanged monthly at 20 locations and quarterly at 53 additional stations. The groundwater monitoring program includes samples from 15 observation and monitoring wells. In addition to the licensee's monitoring program, radiological monitoring programs are conducted at the TMI site by the Commonwealth of Pennsylvania, the state of Maryland, the U.S. Environmental Protection Agency, the U.S. Department of Energy, and the U.S. Nuclear Regulatory Commission.

The monitoring program at the TMI-2 site is subject to change based on review of the results obtained and any requests for additional sampling. It is anticipated that the level of effort will be reduced as contaminated materials are removed from the site and the associated potential for radioactive release is reduced.

The basic environmental monitoring program used in this post-accident study as the basis for estimating the costs of environmental surveillance during cleanup and decommissioning is shown in Table D.8-1. This monitoring program is taken from Appendix G of Reference 19. The program is consistent with the guidelines developed in Reference 27 and with the monitoring effort at the TMI-2 site.

An abbreviated version of the environmental monitoring program for active decommissioning is carried out during continuing care (for the SAFSTOR and ENTOMB decommissioning alternatives). Special surveillance requirements would be included for emergency situations involving radionuclide releases (e.g., fire or malicious acts) that would require prompt emergency actions to minimize public risk. Changes in background levels, in environmental radiation accumulations (e.g., fallout from nuclear weapons testings), and especially in land usage and population distribution may, over a period of years, justify modifications to the continuing care surveillance program. The program is anticipated to be reviewed and revised as appropriate at the following times:

- after all fuel and source material have been removed from the plant

TABLE D.8-1. Basic Environmental Monitoring Program for Post-Accident Decommissioning

Sample Type	Frequency	Analysis	Analytical Detection Limit[(a)]	Number of Sampling Stations Onsite	Number of Sampling Stations Offsite
Terrestrial Samples					
Air particulate	Weekly	Gross Beta	0.002 pCi/m^3	2	4
	Monthly composite	Gross Alpha	0.002 pCi/m^3	2	4
		Gamma Scan[(b)]	0.3 pCi/m^3/isotope	2	4
Air radioiodine	Weekly[(c)]	^{131}I	0.1 pCi/m^3	2	4
Direct radiation	Quarterly	TLD[(d)]	1.25 mrem/quarter increase	8	10
Rainfall	Monthly	Gross Beta	0.5 pCi/L	1	2
		Tritium	1000 pCi/L	1	2
		Gamma Scan[(e)]	25 pCi/L/isotope	1	2
Soil	Semiannually	^{89}Sr, ^{90}Sr	0.01 pCi/g (dry)	3	3
		Gamma Scan	0.1 pCi/g/isotope (dry)	3	3
Vegetation	Semiannually	^{89}Sr, ^{90}Sr	5.0 pCi/kg (wet)	2	3
		Gamma Scan	50 pCi/kg/isotope (wet)	2	3
Animals	Semiannually	^{89}Sr, ^{90}Sr	5 pCi/kg (wet)	2	3
		Gamma Scan	50 pCi/kg/isotope (wet)	2	3
Milk	Monthly	^{89}Sr, ^{90}Sr	1.0 pCi/L	-[(f)]	5[(g)]
		^{131}I[(c)]	0.5 pCi/L	-	5[(g)]
		Gamma Scan	50 pCi/L/isotope	-	5[(g)]
Aquatic Samples					
Surface water	Monthly	Gross Beta	0.5 pCi/L	3	3
		Tritium	1000 pCi/L	3	3
		Gamma Scan[(e)]	25 pCi/L/isotope	3	3
Well water	Quarterly	Gross Beta	0.5 pCi/L	-	3
		Tritium	1000 pCi/L	-	3
		Gamma Scan[(e)]	25 pCi/L/isotope	-	3
Bottom sediment	Semiannually	Gamma Scan	0.1 pCi/g/isotope (dry)	3	3
Vegetation	Semiannually	^{89}Sr, ^{90}Sr	5 pCi/kg (wet)	2	3
		Gamma Scan	100 pCi/kg/isotope (wet)	2	3
Shoreline soil	Semiannually	Gamma Scan	0.1 pCi/g/isotope (dry)	-	3
Fish	Semiannually	^{89}Sr, ^{90}Sr	5 pCi/kg (wet)	2	3
		Gamma Scan	100 pCi/kg/isotope (wet)	2	3

(a) Analytical detection limit is that concentration that is three standard deviations above the average concentration in a blank sample, ensures accuracy limits of ±25%.
(b) Performed if gross beta exceeds 0.1 pCi/m^3.
(c) ^{131}I analyses needed only for first month following shutdown.
(d) Thermoluminescent dosimeter.
(e) Performed if gross beta exceeds 10 pCi/L.
(f) Indicates no sample taken.
(g) Includes one sample from a local milk processor.

- approximately 10 years after decommissioning is completed
- after significant decay of some prominent isotopes has occurred (^{60}Co will decay by about a factor of 10 in 18 years. ^{90}Sr and ^{137}Cs will decay by about a factor of 10 in 100 years).

As experience is gained and a data base is developed, modifications to the environmental program can be expected. The monitoring program can also be eliminated if additional decommissioning is performed to release the site for unrestricted use (for the SAFSTOR and ENTOMB alternatives).

REFERENCES

1. J. A. Ayres, ed., Decontamination of Nuclear Reactors and Equipment, The Ronald Press, New York, New York, 1970.

2. R. L. Dillon, et al., Literature Review of Dilute Chemical Decontamination Processes for Water-Cooled Nuclear Reactors, EPRI NP-1033, Pacific Northwest Laboratory for Electric Power Research Institute, March 1979.

3. J. A. Partridge and R. E. Lerch, Chemical Decontamination of Metals, HEDL-SA-1880, Hanford Engineering Development Laboratory, Richland, Washington, October 1979.

4. W. J. Manion and T. S. LaGuardia, Decommissioning Handbook, DOE/EV/10128-1, Nuclear Energy Services, Inc. for U.S. Department of Energy, November 1980.

5. L. D. Perrigo and J. R. Divine, Decontamination Methods, PNL-SA-7770 REV 1, Pacific Northwest Laboratory, Richland, Washington, May 1980.

6. Environmental Assessment - Use of Epicor II at Three Mile Island, Unit 2, NUREG-0591, U.S. Nuclear Regulatory Commission, Washington, D.C., October 1979.

7. Final Programmatic Environmental Impact Statement Related to 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.

8. P. J. Pettit, et al., "Decontamination of the Douglas Point Reactor by the CAN-DECON Process," Materials Performance, pp. 34-38, January 1980.

9. J. O. Duguid, J. L. Means and D. A. Crerar, "Migration of Radioactive Wastes: Radionuclide Mobilization by Complexing Agents," Science, Vol. 200, pp. 1477-1480, 30 June 1978.

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

11. R. P. Allen, et al., Electropolishing as a Decontamination Process: Progress and Application, PNL-SA-6858, Pacific Northwest Laboratory, Richland, Washington, April 1978.

12. M. W. McCoy, H. W. Arrowsmith, and R. P. Allen, Vibratory Finishing as a Decontamination Process, PNL-3336, Pacific Northwest Laboratory, Richland, Washington, October 1980.

13. J. M. Halter and R. G. Sullivan, Contaminated Concrete Surface Layer Removal, presented at Fourth International Conference on Contamination Control, Washington, D.C., September 10-13, 1978.

14. W. D. Kittinger, B. F. Ureda, and J. W. Carroll, Experience and Techniques in Atomics International's Recent Decommissioning Programs, presented at Health Physics Society Annual Meeting, Philadelphia, Pennsylvania, July 8-13, 1979.

15. D. H. Card, Early Waste Retrieval Interim Report, TREE-1047, EG&G Idaho, Inc., Idaho Falls, Idaho, February 1977.

16. C. C. Meinhardt and G. M. Holter, Interim Report on Mini-Vacuum System, RHO-CD-524, Rockwell Hanford Operations, Richland, Washington, May 1978.

17. N. I. Sax, Dangerous Properties of Industrial Materials, Fifth Edition, VanNostrand Reinhold Company, Section 2, "Industrial Air Contaminant Control," by B. Feiner, pp. 40-115, 1979.

18. 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.

19. 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.

20. Safety Evaluation Report Related to the Operation of the Submerged Demineralizer System at Three Mile Island Nuclear Station, Unit No. 2, NUREG-0796, U.S. Nuclear Regulatory Commission, Washington, D.C., June 1981.

21. U.S. Code of Federal Regulations, Title 40, Part 141, "National Interim Primary Drinking Water Regulations" (40 CFR 141), U.S. Government Printing Office, Washington, D.C., 1978.

22. Draft Environmental Impact Statement on 10 CFR Part 61 "Licensing Requirements for Land Disposal of Radioactive Waste," NUREG-0782, Vol. 1, U.S. Nuclear Regulatory Commission, Washington, D.C., September 1981.

23. Technology for Commerical Radioactive Waste Management, DOE/ET-0028, Vol. 4, "Rail Transport of Spent Fuel," pp. 6.2.3 through 6.2.19, U.S. Department of Energy, Washington, D.C., May 1979.

24. Alternatives for Managing Waste from Reactors and Post-Fission Operations in the LWR Fuel Cycle, ERDA-76-43, Vol. 3, "Transportation of Spent Fuel," pp. 22.13 through 22.17, U.S. Energy Research and Development Administration, Washington, D.C., May 1976.

25. Radioactive Materials Transportation Manual, Nuclear Transport Division, Tri-State Motor Transit Co., Joplin, Missouri, 1974.

- cleanup of the reactor coolant system
- solidification and packaging of wastes from accident cleanup operations.

The sequence of accident cleanup tasks and their relationship to decommissioning activities is shown schematically in Figure E.0-1. The rationale for accident cleanup is discussed in Section E.1. Because the processing of accident water and defueling of the reactor must be accomplished even if a decision is made to refurbish rather than to decommission an accident-damaged reactor, the same accident cleanup tasks would be required whether the reactor was refurbished for restart or decommissioned. In Figure E.0-1, a decision point relating to restarting the reactor or completing the 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.

Because accident cleanup activities would be similar whether the reactor is refurbished for restart or decommissioned, the requirements, costs, and safety analysis given in this report are considered to be a good representation independent of the ultimate use of the plant. However, this study does not include a consideration of activities related to refurbishment or restart of a reactor following the accident cleanup period.

Accident cleanup activities are independent of the alternative (DECON, SAFSTOR, or ENTOMB) chosen to decommission the facility, although the methods used to complete certain tasks may vary with the decommissioning alternative.[a] The work required to complete each task will certainly be influenced by the severity of the accident.

Details of accident cleanup are discussed in this appendix, including cleanup methods and procedures, schedules and manpower requirements, and

(a) In this study, cleanup methods are assumed not to vary substantially with decommissioning alternative.

APPENDIX E

DETAILS OF ACTIVITIES AND MANPOWER REQUIREMENTS FOR ACCIDENT CLEANUP AT A REFERENCE PWR

This appendix provides details of the technical requirements and manpower needs for accident cleanup, to supplement the discussion in Chapter 10 of Volume 1. A discussion of the rationale for accident cleanup is also provided.

The reactor accidents postulated in this report 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. The first activities following stabilization of an accident consist of 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 safely defuel the reactor, placing the fuel in a configuration that is safe from nuclear criticality and/or fuel meltdown, and
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 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

26. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-75/087, U.S. Nuclear Regulatory Commission, Washington, D.C., September 1975.

27. J. P. Corley, et al., A Guide for Environmental Radiological Surveillance at ERDA Installations, ERDA-77-24, Pacific Northwest Laboratory for U.S. Energy Research and Development Administration, March 1977.

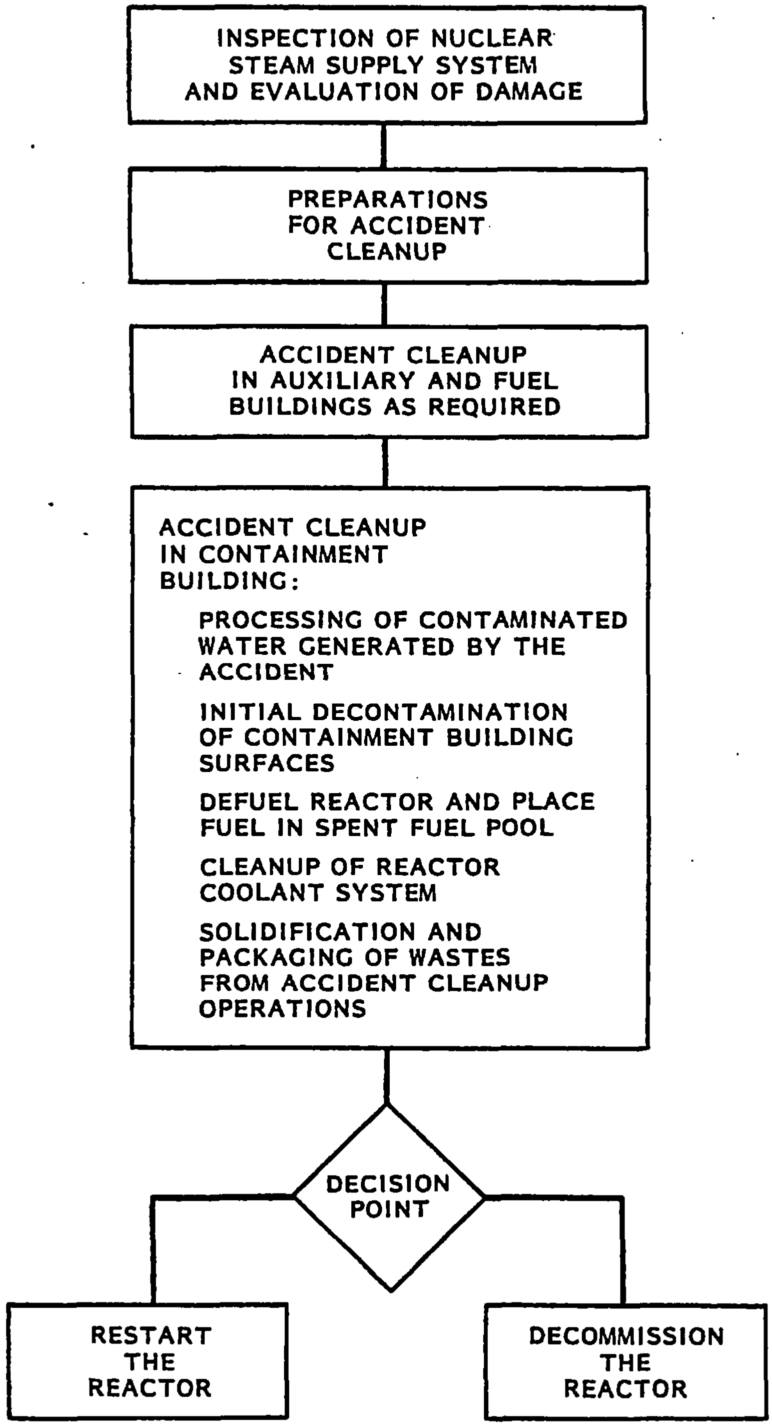


FIGURE E.0-1. Sequence of Accident Cleanup Tasks

occupational radiation doses. The costs of accident cleanup are discussed in Appendix F. Decommissioning activities that follow accident cleanup are discussed in Appendix G.

E.1 RATIONALE FOR ACCIDENT CLEANUP

The defueling of the reactor and the processing of 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. In this report, these cleanup tasks are assumed to be performed prior to other operations of whichever alternative (DECON, SAFSTOR, or ENTOMB) is chosen to decommission the facility. The rationale for their early completion is given in the following paragraphs.

The objective of reactor defueling is to remove all fuel and damaged reactor parts from the reactor pressure vessel. Defueling of the reactor ensures that no further damage to plant or equipment can occur as a result of an accidental criticality. Defueling also eliminates the possibility of any additional dispersal of radioactive fission products from damaged fuel. Defueling results in reduced radiation exposures to workers engaged in cleanup and decommissioning operations in the vicinity of the reactor vessel. Defueling must be accomplished prior to decontamination of the spent fuel pool and refueling canal and to decontamination or disposal of equipment needed for defueling operations. It is a prerequisite to the decontamination or disposal of reactor coolant system components and piping.

Steps in defueling the reactor include:

- removal of the reactor vessel head and inspection of the core
- removal of the structural components (i.e., vessel internals) above the fuel
- removal of intact fuel assemblies and removal and packaging of damaged fuel assemblies
- removal of loose fuel element debris
- interim storage of spent fuel.

Accident cleanup activities must establish conditions within the reactor building that allow defueling to proceed. This requires the venting of any radioactive gases present in the reactor building, the processing of contaminated accident water, and some decontamination of building surfaces to reduce levels of penetrating radiation in selected work areas sufficiently to permit reasonable occupancy times without excessive radiation exposures to workers engaged in defueling operations. In keeping with ALARA principles, decontamination efforts during accident cleanup are restricted to those operations that result in the greatest reduction in residual radioactive contamination with the least radiation dose to decontamination workers. These operations include washdown of containment building surfaces using the containment spray system and water jet equipment, draining of accident water from basements and sumps, and installation of temporary shielding around localized hot spots. Hands-on decontamination work using mops and wipes with assorted cleaners is performed in those instances where significant reductions in local area radiation dose rates can be readily achieved.

In addition to reducing the radiation exposures to workers engaged in building cleanup and reactor defueling operations, the prompt processing of contaminated accident water is needed for the following reasons:

- transfer of the radioactivity from a mobile form in the water to a less mobile form with a reduced volume to facilitate disposal
- reduce the inventory of contaminated water in the building basement so that equipment needed for decommissioning operations or for maintenance of the facility in a shutdown condition is not adversely affected by the water
- provide surge capacity for the temporary storage of water generated by decontamination activities
- alleviate potential psychological stress to the local population that could come from the continued presence of large quantities of contaminated water in the building.

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.

There are several reasons for chemical decontamination of the reactor coolant system during accident cleanup. These reasons include:

1) Decontamination of the reactor coolant system serves to reduce the radiation dose to workers within the containment, regardless of which decommissioning alternative is chosen or whether restart is chosen. It preserves all the options for facility decommissioning or reactor restart in a way that is quite effective in terms of radiation exposure reduction.

2) If DECON is chosen as the alternative for plant decommissioning, it is necessary to drain the reactor coolant system and remove any fuel debris or loosely held contamination that might be dispersed during dismantling and disposal of coolant system components. Because chemical decontamination of the coolant system is largely a remote operation, the radiation dose to decontamination workers is likely to be considerably less than the dose to workers engaged in dismantling and packaging the equipment for disposal without prior decontamination.

3) If the SAFSTOR alternative is chosen for plant decommissioning, coolant system decontamination might be deferred until the end of the safe storage period when radiation exposure levels are somewhat reduced. However, failure to perform the chemical decontamination before placing the facility in safe storage might preclude ever taking such action, since the pumps, valves, and associated equipment required to handle the decontamination solutions would likely be unusable after an extended storage period, and the costs in dollars and occupational radiation dose of the equipment refurbishment required to accomplish chemical decontamination at a later date could be significant.

For the ENTOMB alternative, this accident cleanup step might include only minimal decontamination by flushing to empty the coolant system of contaminated water and to remove lightly held or water-soluble radioactivity. However, failure to perform the chemical decontamination would result in significantly higher radiation doses to workers performing entombment tasks in the vicinity of the reactor coolant system.

The programmatic environmental impact statement (PEIS) on decontamination of TMI-2[1] addresses the question of the relationship of accident cleanup activities to alternatives for future restart or for decommissioning of that facility. A major conclusion of the TMI-2 PEIS is that the need for accident cleanup is not significantly affected by an early decision to decommission or to refurbish and restart the facility. A decision to decommission or to refurbish does not affect the goals of accident cleanup or the tasks that must be accomplished during cleanup. A decision about future use of the reactor might, however, affect the choice of procedures used to accomplish certain cleanup tasks. For example, a decision about whether some equipment was to be disposed of or reused could affect the choice of chemicals used to decontaminate the equipment. However, destructive decontamination is rejected as a viable alternative for reactor building decontamination in the TMI-2 PEIS.

Destructive decontamination involves the use of corrosive chemicals for surface decontamination without regard to damage to the surface. An advantage of using corrosive chemicals is that they remove part of the surface so that radionuclides that are strongly adsorbed or chemically bonded to the surface are also removed. Disadvantages to the use of corrosive chemicals include:

- an increased health and safety hazard for workers
- equipment used to collect and process spent liquid may be adversely affected by the harsh chemicals
- special measures may be needed to protect equipment needed for subsequent cleanup or decommissioning operations.

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 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 either the decommissioning or refurbishment of the facility following the completion of accident cleanup.

Based on the above considerations, it is assumed in this report that the tasks included in accident cleanup are necessary and the procedures used to accomplish these tasks are essentially independent of whether the facility is ultimately restarted or decommissioned, and if decommissioned, of the alternative (DECON, SAFSTOR, or ENTOMB) chosen. The work required to accomplish each accident cleanup task is, however, affected by the severity of the accident. Technical requirements for accident cleanup are discussed in the following subsections. The costs of accident cleanup are discussed in Appendix F.

E.2 DETAILS OF PREPARATIONS FOR ACCIDENT CLEANUP

A period of planning and preparation precedes the actual performance of cleanup operations within a reactor facility that has been involved in an accident. Planning and preparation activities and the manpower requirements for their performance are described in this section.

In the studies[2,3] of reactor decommissioning following normal shutdown, planning and preparation activities for decommissioning were assumed to take place during the final 18 months to 2 years of reactor operation. Since accidents are unplanned events, preparations for accident cleanup and for the subsequent decommissioning must begin after the accident has occurred and the plant is shut down. Many planning and preparation tasks must be completed before accident cleanup operations begin, thereby delaying the start of these operations. Other planning and preparation tasks cannot be completed during the initial preparations phase and must be deferred until some cleanup renders the accident-damaged facility more accessible for detailed examination.

If the auxiliary and fuel buildings are contaminated as the result of an accident, decontamination of these buildings must precede accident cleanup inside the containment. The auxiliary and fuel buildings house the spent fuel storage and handling facilities and other essential systems and services (e.g., the coolant water treatment system and the radwaste treatment systems) needed for cleanup inside the containment building. Cleanup operations in these buildings can proceed concurrently with planning and preparation for containment building cleanup. (For example, cleanup of the auxiliary and fuel building at TMI-2 began about 6 weeks after the accident.) However, for convenience of analysis, planning and preparation activities and auxiliary and fuel building cleanup are treated as entirely separate operations in this report.

E.2.1 Planning and Preparation Activities

Several planning and preparation activities must be performed prior to the operational phase of containment building cleanup. These activities include:

- containment entry and data acquisition
- venting of radioactive gases (e.g., krypton-85)
- 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.

E.2.1.1 Containment Entry and Data Acquisition

Data on the post-accident radiological and physical condition of the plant are obtained and analyzed during planning and preparation activities. These data provide a basis for planning accident cleanup operations and for selecting appropriate methods and equipment to perform the cleanup. The data also provide information needed to prepare documentation for regulatory agencies as discussed in Section E.2.1.3.

Radiation surveys are performed to measure contamination levels and radiation exposure rates inside the damaged containment. These surveys provide information on which to base decisions about decontamination requirements and work procedures as well as baseline data for later use in judging the effectiveness of cleanup operations. Initial radiation surveys are made as detailed as possible without excessive exposure to personnel performing the surveys.

Additional data needed for planning cleanup and decommissioning operations include information about the operational status of plant systems and services (such as radiation detectors, ventilation equipment, electrical services, cranes, radwaste equipment, etc).

Initial entries into the containment building may provide only limited information or information of general nature, especially following a severe accident where radiation levels are high and time of access is limited. More detailed information about the condition of the facility can be obtained after initial decontamination of building surfaces to reduce exposure rates is completed. Detailed information about the status of the damaged fuel core may be available only after the reactor vessel head is removed.

At TMI-2,[1] initial entries into the containment building for the purpose of measuring surface contamination and radiation exposure levels and of inspecting any damage to plant and equipment were made by teams of 2 to 14 persons who remained inside the containment for time periods ranging from 20 minutes to 2 hours. The maximum whole body doses received by individual members of these survey teams ranged from less than 100 mrem to about 650 mrem. Entry teams made radiation measurements, took samples of surface radioactivity by wiping and scraping surfaces, inspected for damage, took photographs of the interior, performed minor maintenance functions, and removed a few small, loose items for subsequent laboratory analysis.

Radiation surveys continue during cleanup operations to evaluate the effectiveness of these operations. A comprehensive radiation survey, taken after chemical decontamination of the reactor coolant system is completed, provides the basis for planning final decommissioning operations.

E.2.1.2 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 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 at times greater than 1 year following the accident are the relatively long-lived cesium isotopes and ^{90}Sr which plate out on building surfaces or are retained in the accident water. An exception is the noble gas ^{85}Kr, which has a 10.7-year half life and which can also constitute a major radiological hazard to cleanup and decommissioning workers. (At TMI-2, the amount of ^{85}Kr remaining in the reactor building atmosphere 15 months after the accident was estimated at 44,000 Ci.[1]) The ^{85}Kr must be removed from the containment building atmosphere so that workers can begin the tasks necessary to clean the building, maintain instruments and equipment, and eventually remove the damaged fuel from the reactor core. (Removal of the ^{85}Kr from the reactor building atmosphere at TMI-2 was estimated to reduce the radiation dose rate for workers by about a factor of 4.)[4]

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

- purging
- selective absorption
- charcoal adsorption
- gas compression and storage
- cryogenic processing.

Purging involves the controlled release of air from inside the reactor building by way of filtering and monitoring equipment that leads to the building ventilation stack. The building ventilation system is equipped with valves to control the rate of air release and with trains of filters to remove fine particulate radioactive material from the air before it is discharged to the stack. As the air bearing the ^{85}Kr leaves the reactor building, it is replaced by fresh air from the outside that enters through an open valve.

Selective absorption is a process whereby air is withdrawn from the containment building, the krypton is separated from this air, and the decontaminated air is returned to the building. The contaminated air passes through a column in which liquid Freon absorbs the krypton while allowing the other gases to pass through unchanged. The krypton which is thus removed can be stored in high-pressure or low-pressure gas cylinders.

Charcoal adsorption is a process by which the contaminated air from the containment building is passed through large tanks containing charcoal. The krypton adheres to the surface of the charcoal which may be at normal temperature or refrigerated. The charcoal from this process is then isolated and stored.

Gas compression is a process by which the air containing the radioactive krypton gas in the containment building is drawn off into pressurized storage containers. These pressurized containers are then stored in sealed sections of piping. The gas is maintained under pressure in storage until the ^{85}Kr has decayed to levels judged acceptable for release.

Cryogenic processing is the condensation of ^{85}Kr from the air by bringing it into direct contact with liquid nitrogen (-196°C). The liquified ^{85}Kr is collected, restored to a gas form, and stored to allow decay. An alternative to storage is to transport the containers of the separated krypton (whether from the cryogenic or the selective absorption processes) to a burial ground for disposal or to a remote area for release to the environment.

An environmental assessment evaluating alternatives for ^{85}Kr removal from the TMI-2 reactor building[4] concluded that the potential health impact on the public of using any of the above alternatives for krypton removal, including purging, is negligible. All of the alternatives except purging require the construction of special equipment or facilities which could delay ^{85}Kr removal for periods of 9 months to 4 years. The environmental assessment concluded that purging was the quickest and safest for workers at TMI-2 to accomplish, and this method was used for krypton removal.

Purging of the TMI-2 reactor building atmosphere occurred during the period from June 28 to July 11, 1980. This purging reduced the airborne concentration of radioactivity in the building by about a factor of 10^7. Measurements of air concentrations of ^{85}Kr outside the reactor building showed that the point of maximum exposure during the purge was at a location about 0.6 km from the site.[1] If a person had remained at this location throughout the purge, he/she would have received a beta skin dose of 4.5 mrem and a whole body gamma dose of 0.05 mrem.

Based on the above considerations, in this report it is assumed that ^{85}Kr is removed from the containment building by controlled venting of the building atmosphere. This controlled venting occurs over a period of several weeks during preparations for accident cleanup. A description of the containment building ventilation system is given in Section B.1.5.

E.2.1.3 Documentation for Regulatory Agencies

Existing regulations, guides, and standards that apply to a nuclear power reactor that has been involved in an accident are discussed in Chapter 5 of Volume 1. At the start of planning and preparation activities, the current status of these requirements must be reviewed by the licensee who must accomplish the cleanup and decommissioning in compliance with their provisions. The cleanup and 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 authority for regulating nuclear fuel cycle activities.

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. Regulations pertaining to termination of the operating license are set forth in Section 50.82 of Part 50 of Title 10 of the Code of Federal Regulations. Regulatory Guide 1.86 describes methods acceptable to the NRC for satisfying the requirements of Section 50.82. Documentation that must be provided by the licensee includes:

- a description of the current facility status
- a description of the ultimate facility status
- proposed changes to the technical specifications
- descriptions of proposed cleanup and decommissioning operations and associated environmental and safety precautions
- safety and environmental analyses of cleanup and decommissioning operations and of any resultant releases of radioactivity
- safety and environmental analyses of the plant in its ultimate status.

Consistent with the intent of 10 CFR 51, Licensing and Regulatory Policy and Procedures for Environmental Protection, and in keeping with the purposes of the National Environmental Policy Act (NEPA), before decommissioning begins, an environmental impact statement or environmental assessment is needed describing the probable effects of the proposed cleanup and decommissioning actions. The licensee is required to provide supporting information to assist the NRC in the preparation of these documents. As an illustration of the type of documentation required, as of June 1981 (27 months after the accident), the following environmental statements and assessments had been prepared for the decontamination of TMI-2:

1) Final Programmatic Environmental Impact Statement (PEIS) Related to Decontamination and Disposal of Radioative Wastes from the TMI-2 Accident.(1) The PEIS is an overall study of the activities necessary for decontamination of the facility, defueling, and disposition of the radioactive wastes. It is intended to provide an overall evaluation of the environmental impacts that could result from these activities.

2) Final Environmental Assessment for Decontamination of the TMI-2 Reactor Building Atmosphere.(4) This document provides an assessment of information considered by the NRC in arriving at a recommendation for the preferred method of removing ^{85}Kr from the

containment building so that workers can begin the tasks necessary to decontaminate the building and remove the damaged fuel from the reactor core.

3) Environmental Assessment on the Use of EPICOR-II at TMI-2.[5] This is an evaluation of the effect on public health and safety of the use of the EPICOR-II system for the cleanup of radioactive contaminated waste water which had accumulated in the Unit 2 auxiliary building tanks. The document includes a consideration of the environmental impacts of the use of EPICOR-II and a discussion of alternatives to the EPICOR-II system.

4) Safety Evaluation Report on the Operation of the Submerged Demineralizer System at TMI-2.[6] This is an evaluation of the effect on public health and safety of the decontamination of reactor building sump water and reactor coolant system water using the submerged demineralizer system (SDS) followed by polishing in EPICOR-II. The evaluation only considers the processing of the contaminated water and does not consider the disposition of the processed water.

Additional impact statements and assessments may be required for specific decontamination or decommissioning operations at TMI-2 as cleanup work at that plant continues.

The cleanup and decommissioning of an accident-damaged reactor is also subject to constraints imposed by statements, orders, and amendments to the facility license issued by the NRC subsequent to the accident. NRC actions in connection with the decontamination of TMI-2 are detailed in Reference 1. 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.

The time requirement for furnishing information to regulatory agencies, issuing environmental statements and assessments, and securing regulatory approvals to go ahead with specific cleanup tasks is a critical factor in determining when actual cleanup operations can begin. At TMI-2, initial cleanup of the auxiliary and fuel handling building (AFHB) began about 6 weeks after the accident. However, the processing of accident water from the TMI-2 reactor building basement did not begin until about 30 months after the accident.

E.2.1.4 Design, Fabrication, and Installation of Special Equipment

Planning and preparation includes the identification and procurement of special tools and equipment required for accident cleanup and decommissioning. A list of special tools and equipment is given in Section D.3. Some items, such as cutting tools or decontamination equipment, can be identified early in the planning stage before actual cleanup begins. Other items, such as special tools for the removal of damaged fuel from the reactor core, may not be identified until the initial building decontamination is completed, the reactor pressure vessel head is removed, and a visual inspection of the fuel core is made.

Major facilities and equipment items required for cleanup of an accident-damaged reactor include the following:

- 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 mock-up 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 radwastes 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.

Many of these facilities and equipment items require design and development work as well as actual fabrication and testing. Evaporator/solidification facilities, volume reduction incinerators, and laundry facilities are commercially available and can be purchased or rented.

The demineralizer system postulated for treatment of the contaminated water is described in Section E.4.1. It consists of a filtration and ion exchange system patterned after the Submerged Demineralizer System (SDS) installed at TMI-2.[6] The system is designed to operate under water for radiation shielding and cooling and is installed in the spent fuel pool during preparations for accident cleanup. Processed accident water is stored in 1000-m^3-capacity carbon steel tanks to await reuse for building decontamination or controlled discharge to the river. The tanks are constructed adjacent to and outside of the building in which the demineralizer is located. One tank is assumed to be required for processed water storage following the scenario 1 accident; two tanks are required following the scenario 2 accident; and three tanks are required following the scenario 3 accident. The number of tanks required is based on the estimated volumes of water that must be processed during cleanup following the reference accidents.

Onsite waste storage structures needed for interim storage of radioactive wastes include a warehouse for low-activity wastes that are packaged in 0.21-m^3 steel drums or in plywood boxes and a shielded facility for high-activity wastes packaged in steel drums or liners. The warehouse is a sheet-metal building on a concrete foundation. The shielded facility is shown conceptually in Figure E.2-1. Basically, the shielded facility consists of underground concrete cells with concrete cover blocks that are thick enough to provide the necessary radiation shielding. A mobile gantry crane enclosed in a sheet-metal building is used to place the radioactive waste in and retrieve the waste from the cells. The building is provided primarily for weather protection, since the external surfaces of waste containers will be free of smearable contamination when they are placed in the storage cells. The dimensions of the onsite waste storage structures are chosen to accommodate the volumes of waste expected to result from accident cleanup operations.

The extraction of damaged fuel from the fuel core requires special handling tools. Fuel assemblies that have experienced structural damage sufficient to cause them to break apart if they are lifted from the top using normal defueling equipment must be lifted by special devices that support the bottom and sides of the assembly. Conceptual requirements for possible tools for fuel removal are given in Section E.4.1. Extensive design and development work may be required to provide this equipment. In addition, a mockup of a section of the reactor core is used to test this defueling equipment and train operators in procedures for its use. Because of uncertainties in the physical condition of the fuel, some of the design and fabrication of fuel handling equipment may be delayed until cleanup operations have proceeded to the point of reactor pressure vessel head removal so that detailed inspection of the core is possible.

Damaged fuel assemblies that are removed from the reactor core following the scenario 2 and scenario 3 accidents require overpacking in stainless steel

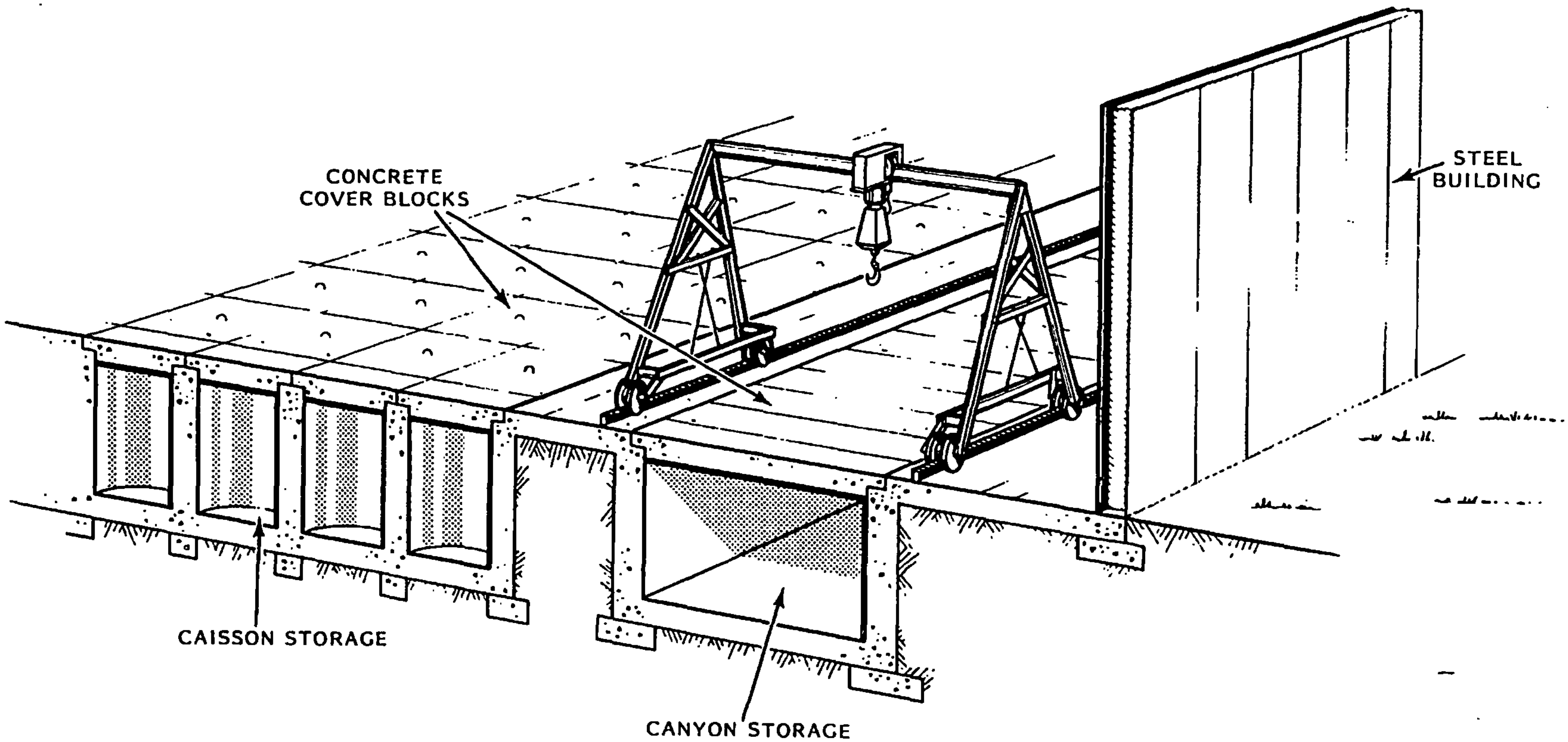


FIGURE E.2-1. Conceptualized Shielded Facility for Onsite Storage of High-Activity Wastes canisters prior to interim storage in the spent fuel pool. Canistered fuel cannot be accommodated in existing fuel storage racks that are designed to store normal fuel elements. Therefore, a sufficient number of new racks capable of storing the canistered fuel assemblies must be fabricated and installed in the spent fuel pool prior to the start of defueling operations.

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.

E.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.

E.2.1.6 Selection and Training of Accident Cleanup Staff

The selection and training of operations staff for accident cleanup is an important part of planning and preparation. Staffing requirements are identified during this period, and key positions are filled with qualified engineering and operating personnel.

The postulated staff organization for preparations for cleanup is described in Section E.2.4. The postulated staff organization for accident cleanup is described in Section E.4.4.

Detailed knowledge of and familiarity with the facility being decontaminated increases the effectiveness of the cleanup staff. Consequently, positions are assumed to be filled, whenever possible, with utility personnel familiar with the construction and operation of the plant, to capitalize on experience and minimize training requirements. Additional training required to perform specific cleanup tasks is provided, with special emphasis given to the use of new and unique equipment and procedures. This results in improvements in efficiency and reduces occupational exposures when actual cleanup operations are performed inside the plant.

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 individuals 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.

The actual procedures that may be required for the removal of damaged fuel from the reactor core cannot be known with certainty until the reactor vessel head is removed and visual inspection of the damaged fuel is accomplished. Accordingly, final training in the use of tools and procedures for defueling the reactor may be delayed until cleanup operations have progressed to a point where the reactor vessel head can be removed. For training in the performance of fuel removal, a full-scale mockup of a section of the reactor vessel is used.

E.2.1.7 Removal of Accumulated Spent Fuel

Because space in the spent fuel pool is needed for the filter/demineralizer system used to process contaminated water and for temporary storage of fuel from reactor defueling operations, it is necessary to remove the spent fuel already stored in the pool from prior plant refuelings. The fuel is assumed to be transported to an off site facility (an independent spent fuel storage installation, ISFSI) for interim storage. Shipment and storage costs for the transfer of accumulated spent fuel from the spent fuel pool to an ISFSI are assumed to be charged to reactor operations but are shown as an optional item in planning and preparations costs.

Excess fuel storage racks are also removed to increase the available space in the spent fuel pool. Excess storage racks are packaged and transported to a shallow-land burial ground for disposal.

Defueling following the scenario 2 and scenario 3 accidents includes a requirement for overpacking the fuel assemblies in stainless steel canisters. Existing storage racks do not have the proper dimensions to accommodate the canistered fuel. Prior to defueling operations for the scenario 2 and scenario 3 accidents, the existing storage racks are removed and replaced with racks that can accommodate the canistered fuel. The costs of replacement storage racks are included in the special equipment costs.

E.2.2 Time Requirements for Preparations for Accident Cleanup

Time requirements for planning and preparation depend on several factors including the severity of the accident, the time needed to design, fabricate, install, and test special facilities and equipment, and the time required to secure regulatory approvals for specific 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. Delays by the licensee in responding to requests for information and/or delays in the review process that precedes the issuing of regulatory approvals could significantly delay the start of accident cleanup operations.

A minimum of 9 months are assumed to be required to discharge the accumulated spent fuel from the spent fuel pool and ship it to an 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 continuously available to transport the fuel. Based on experience at TMI-2, as much as 2-1/2 years could be required to prepare the documentation, install and test new equipment, and secure the necessary regulatory approvals to begin the accident cleanup operations. In this study, planning and preparation activities that precede accident cleanup of the containment building are estimated to require approximately 1 to 3 years, depending on accident severity.[a]

E.2.3 Occupational Doses for Preparations for Accident Cleanup

The major source of occupational radiation dose during preparations for accident cleanup is the dose received by workers who enter the containment building to measure contamination levels and radiation exposure rates, assess the damage to the building and equipment, install monitoring systems, and make minor repairs to essential systems and equipment. The assumed average external whole-body doses received by these workers are summarized in Table E.2-1. Average dose rates are based on accident scenario information presented in Chapter 8 of Volume 1. The containment building is assumed to be vented for the removal of ^{85}Kr prior to the initial entry of workers into the building. Workers are assumed to spend an average of 1 hour inside the building during each entry. All personnel entering the building wear protective clothing and full-face respirators. For entries into containment following the scenario 1 accident, the average individual worker dose per entry is

TABLE E.2-1. Estimated Occupational Doses to Workers Entering Containment During Preparations for Accident Cleanup

	Accident Scenario		
	Scenario No. 1	Scenario No. 2	Scenario No. 3
Number of Entries into Containment	12	18	24
Average Time per Entry (hours)	1	1	1
Average Dose Rate (rem/hr)	0.03	0.25	1.5
Number of Workers per Entry	10	10	10
Total Accumulated Occupational Dose (man-rem)	3.6	45	360

(a) 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. See Section F.1.

0.03 rem and the total accumulated occupational dose for all entries is 3.6 man-rem. For entries into containment following the scenario 2 accident, the average individual worker dose per entry is 0.25 rem and the total accumulated occupational dose for all entries is 45 man-rem. For entries into containment following the scenario 3 accident, the average individual worker dose per entry is 1.5 rem, and the total accumulated occupational dose for all entries is 360 man-rem.

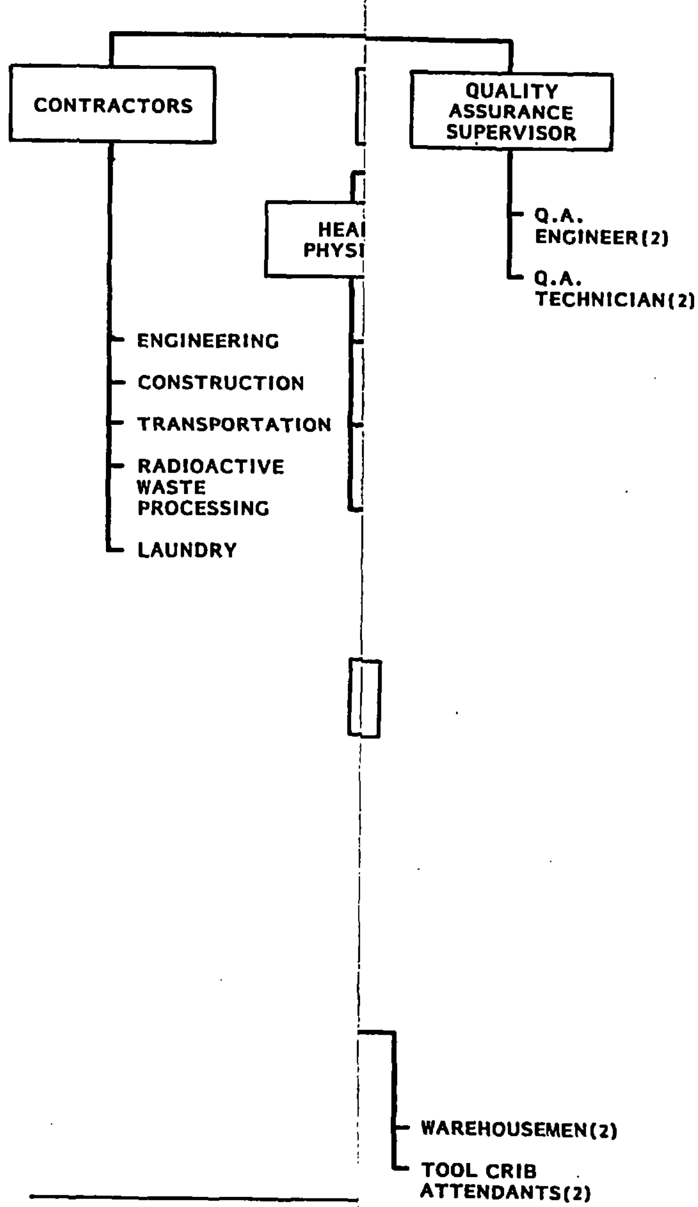
E.2.4 Staff Requirements for Preparations for Accident Cleanup

A postulated staff organization for preparations for accident cleanup is shown in Figure E.2-2. The 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 branch has the responsibility to maintain the reactor in a safe shutdown condition. In addition to operations in the reactor control room, this responsibility entails the following activities:



NOTES:

1) NUMBERS IN PAREN OF PERSONS IS NOT PERSONNEL REQUIRI

2) THE SECURITY FORC PLANT MAINTENANC

FIGURE E.2-2. Postulated Staff Organization for Preparations for Accident Cleanup

- maintain and repair systems required to keep the reactor in a safe shutdown condition.
- monitor and maintain auxiliary systems such as plant communications, heating, ventilation and air conditioning, etc.

The plant operations staff is responsible for the transfer of spent fuel from the fuel storage pool to rail casks for shipment to an ISFSI. They also assist the cleanup planning staff in the aquisition of data on the radiological and physical condition of the containment building and in the installation and testing of systems required for accident cleanup.

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

Staff not specifically involved in preparations for accident cleanup or required to maintain the plant in a safe shutdown condition are not shown in Figure E.2-2. Overhead staff involved in general plant management, plant stores, personnel administration, public communications, medical services, etc., are not shown in the figure and are not included in estimates of staffing costs given in Appendix F.

Estimated staff labor requirements for utility staff involved in preparations for accident cleanup are shown in Table E.2-2. Labor requirements are given on a man-year-per-year basis and are shown for each of the PWR accident scenarios.

In addition to the utility staff involved in preparations for accident cleanup, shown in Figure E.2-2 and Table E.2-2, contractors are hired to provide specific services that include the following:

- engineering assistance in preparing documentation for regulatory agencies, designing special tools and equipment, and preparing work plans and work schedules
- design, fabrication, and installation of major facilities needed for accident cleanup such as the demineralizer system for processing accident water, structures for interim storage of radioactive waste, and a mock-up of the reactor vessel

TABLE E.2-2. Estimated Utility Staff Labor Requirements for Preparations for Accident Cleanup

Position	Staff Labor Requirements (man-years/year) for Preparations for Accident Cleanup		
	Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
Plant Superintendent	1.0	1.0	1.0
Assistant Plant Superintendent	1.0	1.0	1.0
Consultants	3.0	6.0	10.0
Secretaries and Work Processors	8.0	10.0	12.0
Site Support Staff			
Health and Safety Supervisor	1.0	1.0	1.0
Health Physicist	1.0	1.0	1.0
Senior Health Physics Technician	8.0	8.0	12.0
Health Physics Technician	16.0	16.0	24.0
Protective Equipment Attendant	4.0	8.0	8.0
Industrial Safety Specialist	1.0	1.0	1.0
Industrial Safety Technician	2.0	2.0	2.0
Security Supervisor	1.0	1.0	1.0
Security Shift Supervisor	4.0	4.0	4.0
Security Patrolman	48.0	48.0	48.0
Contracts and Accounting Supervisor	1.0	1.0	1.0
Accountant	1.0	1.0	2.0
Contracts Specialist	1.0	1.0	1.0
Insurance Specialist	1.0	1.0	1.0
Procurement Specialist	1.0	1.0	1.0
Clerk	2.0	4.0	4.0
Quality Assurance Supervisor	1.0	1.0	1.0
Quality Assurance Engineer	1.0	2.0	2.0
Quality Assurance Technician	1.0	2.0	2.0
Subtotals	96.0	104.0	117.0
Plant Operations Staff			
Plant Operations Supervisor	1.0	1.0	1.0
Plant Chemist	1.0	1.0	1.0
Chemist	2.0	2.0	2.0
Reactor Operations Engineer	1.0	1.0	1.0
Engineer	2.0	2.0	2.0
Reactor Operations Shift Supervisor	4.0	4.0	4.0
Senior Reactor Operator	8.0	8.0	8.0
Reactor Operator	16.0	16.0	16.0
Utility Operator	16.0	16.0	16.0
Technician	12.0	12.0	12.0
Craft Supervisor	1.0	1.0	1.0
Crew Foreman	4.0	4.0	4.0
Maintenance Mechanic	16.0	16.0	16.0
Instrument Technician	16.0	16.0	16.0
Warehouseman	4.0	4.0	4.0
Tool Crib Attendant	4.0	4.0	4.0
Subtotals	108.0	108.0	108.0
Cleanup Planning Staff			
Cleanup Planning Supervisor	1.0	1.0	1.0
Engineering Supervisor	1.0	1.0	2.0
Engineer	6.0	8.0	12.0
Estimator	1.0	2.0	4.0
Draftsman	2.0	4.0	6.0
Crew Leader	1.0	2.0	2.0
Utility Operator	4.0	8.0	16.0
Craftsman	8.0	12.0	16.0
Laborer	4.0	8.0	16.0
Subtotals	28.0	46.0	75.0
Totals	245.0	276.0	324.0

- specialized waste processing services such as evaporation of contaminated solutions and incineration of combustible wastes
- transportation of radioactive wastes to offsite storage or disposal facilities
- laundry services.

The staff shown in Figure E.2-2 is augmented as described in Section E.3.4 to provide personnel for accident cleanup in the auxiliary and fuel buildings.

E.3 DETAILS OF ACCIDENT CLEANUP IN THE AUXILIARY AND FUEL BUILDINGS

The focus of the accident cleanup activities following a reactor accident is on decontamination of the containment building and defueling of the reactor. In addition to containment building contamination and fuel core damage, a serious accident might also result in fission-product contamination of the auxiliary and fuel buildings. (The 28 March 1979 accident at TMI-2 provides an example of such contamination of these other buildings.) The auxiliary and fuel buildings contain many components of safety-related systems (e.g., the tanks, pumps, and piping and the 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 Volume 1.) The contamination includes radioactive plateout on building and equipment surfaces, small concentrations (puddles) of contaminated water in sumps and on floors where fluid has leaked from pipes and tanks, and fission product contamination of the liquid in tanks, pipes and other components of the emergency core cooling system (ECCS), and the chemical and volume control system (CVCS). The amount of radioactivity in tanks and pipes is assumed to be 20,000 Ci contained in 200 m^3 of liquid. This radioactivity is predominantly ^{134}Cs, ^{137}Cs, and ^{90}Sr. 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 auxiliary building cubicles that contain the filters, demineralizers, and holdup tanks for the CVCS.

Decontamination of the auxiliary and fuel buildings has as goals the reduction of general area radiation exposure levels to about 1-3 mR/hr and the reduction of exposure levels in work areas (e.g., areas that contain coolant water treatment system and radwaste treatment system components) to about 10-30 mR/hr.

Details of accident cleanup methods and procedures, manpower requirements, and occupational radiation doses for accident cleanup in the auxiliary and fuel buildings are discussed in the following subsections.

E.3.1 Procedures for Accident Cleanup in the Auxiliary and Fuel Buildings

The sequence of tasks postulated in this study for accident cleanup in the auxiliary and fuel buildings includes 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 demineralizer system used to process contaminated water.
3. Install the demineralizer system in the spent fuel pool.
4. Flush CVCS tanks and pipes and process the contaminated liquid through the 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 in 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 and to verify the effectiveness of cleanup operations.

The decontamination of building and equipment surfaces includes the following activities:

- removal and packaging of debris and of contaminated non-essential items for storage or disposal
- removal of loosely adhering contamination by vacuuming or by hosing down the contaminated surfaces
- mopping and wet vacuuming to remove small volumes of contaminated liquids that leak from pumps, valves, and flanges
- wiping or scrubbing to remove contamination that adheres firmly to a surface
- scrubbing and mopping of floors.

The details of decontamination procedures are given in Appendix D.

The demineralizer system is described in Section E.4.1. One leg of the system is installed and used for processing contaminated liquids. This also provides a trial run to prepare for processing the containment building accident water that is contaminated to a level almost an order of magnitude greater than that of contaminated auxiliary building liquids.

Accident cleanup in the auxiliary and fuel buildings includes the decontamination and the maintenance or repair of systems or items of equipment that are needed for accident cleanup in the containment building. These systems or equipment items include:

- reactor bleed holdup tanks and associated equipment
- reactor coolant pump water cooling and seal water systems
- miscellaneous waste holdup tanks and associated equipment
- coolant evaporator system components
- spent fuel storage pool water cleanup system
- fuel transfer equipment and handling cranes
- makeup and purification demineralizers and filters.

Radioactive wastes from accident cleanup in the auxiliary and fuel buildings include sludge, process solids from the treatment of contaminated liquids, chemical decontamination solutions, contaminated equipment, and miscellaneous trash. The volumes of waste generated during cleanup and the assumed alternatives for treatment, packaging, and disposal of these wastes are given in Table E.3-1.

E.3.2 Schedules and Cleanup Worker Requirements for Accident Cleanup in the Auxiliary and Fuel Buildings

Accident cleanup in the auxiliary and fuel buildings is postulated to begin during preparations for cleanup of the containment building and to be substantially completed before containment building cleanup begins. The time required for cleanup of the auxiliary and fuel buildings depends on the extent of radioactive contamination. A sequence and schedule for accident cleanup in these buildings is shown in Figure E.3-1. The total time requirement for cleanup in these buildings following the scenario 2 and scenario 3 accidents is estimated to be about 2.2 years.

Cleanup of 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 E.2-2. This cleanup staff includes decontamination crews, a crew that provides construction and maintenance support, and waste processing and waste packaging crews, as shown in Figure E.3-2.

The personnel required to actually complete the cleanup tasks in the auxiliary and fuel buildings are shown in Figure E.3-1. Some tasks must be performed in high radiation areas where workers receive their occupational

Waste Type		ted e	Treated Volume (m^3)	Number of Packages	Average Radioactivity per Package (Ci)	Average Surface Radiation Level(b) (R/hr)
Sludge	Immobi		3	14	3	2.4
Process Solids						
Filter Cartridges			0.3	1	20	16
Zeolite Liners			0.9	3	6 800	5 450
Organic Resin Liners			0.3	1	100	80
Chemical Decontamination Solutions	Immobi vinyl		375	1 875	0.053	0.043
Trash						
Compactible, Combustible	Combus immobi		101	480	0.35	0.28
Compactible, Noncombustible	Compac		326	1 550	0.035	0.028
Noncompactible			822	235	0.105	0.084
Contaminated Equipment						
LSA Materials			185	53	0.25	0.20
High-Activity Materials			57	20	5	4
Spent Fuel Storage Racks		5	28	14	0.25	0.2

(a) All waste shipments are by truck.
(b) Based on 0.8 R/hr per curie.
(c) Seven drums per cask. Two casks per tru
(d) One liner per cask. Two casks per truck
(e) Sole use van can transport 120 steel dru
(f) Cask has dimensions of 1.63 m OD by 2.34
(g) Three plywood boxes per truck shipment.

TABLE E.3-1. Packaging and Disposal Requirements for Radioactive Wastes from Accident Cleanup in the Auxiliary and Fuel Buildings

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

Procedures for accomplishing these tasks are given in this section.

Accident cleanup operations are assumed to reduce general area radiation exposure rates in the containment building to the values shown in Table E.4-1. For the scenario 1 accident, cleanup operations are assumed to reduce the radiation exposure rate on the operating floor level to approximately the value that existed during reactor operations prior to the accident.

TABLE E.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

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. An 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

TABLE E.3-3. Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations For Accident Cleanup in the Auxiliary and Fuel Buildings

Worker Category	Estimated Worker Requirements[a] (man-yr)	Estimated Occupational Dose: Total[b] (man-rem)	Estimated Occupational Dose: Individual Average (man-rem/man-yr)	Adjustment[c] Factor	Adjusted Worker Requirement (man-yr)
Crew Leader	17.0	210	12.4	2.5	42.5
Utility Operator	25.0	315	12.6	2.6	65.0
Laborer	25.3	317	12.6	2.6	65.8
Craftsman	44.5	559	12.6	2.6	115.7
Health Physics Technician	17.2	212	12.4	2.5	43.0
Totals	129.0	1613			332.0

(a) Based on Figure E.3-1.
(b) Based on Table E.3-1.
(c) Increase in worker requirements necessary to reduce average individual dose to 5 man-rem/man-year.

adjustment factor of about 2.6 must be applied to the various worker categories to bring the estimated occupational radiation dose for individual workers down to 5 rem/year, resulting in an adjusted cleanup worker requirement of 332 man-years. This adjusted manpower requirement is used in computing staff labor costs for accident cleanup in the auxiliary and fuel buildings following the scenario 2 and scenario 3 accidents (see Section F.2).

E.4 DETAILS OF ACCIDENT CLEANUP IN THE CONTAINMENT BUILDING

Procedures for accident cleanup in the containment building following a serious accident at the reference PWR power station are described in this section. Work schedules, estimated occupational doses, and estimated staff labor requirements based on these procedures are also presented. Time and manpower requirements for accident cleanup depend on accident severity. The requirements given in this section are based on the three accident scenarios described in Chapter 8 and Appendix C.

E.4.1 Procedures for Accident Cleanup in the Containment Building

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

hoses and vacuums or who mop and scrub to remove surface contamination, workers engaged in the processing and packaging of radioactive wastes including contaminated liquids, health physics support personnel who perform monitoring activities within contaminated areas, and craftsmen who disassemble contaminated equipment or who provide maintenance and construction support within contaminated areas.

Dose calculations are based on time and manpower requirements shown in Figure E.3-1. Exposure hours are estimated on the basis that workers engaged in decontamination activities and in the installation or repair of systems and equipment needed for accident cleanup spend an average of 5 hours working in a radiation area during an 8-hour shift. (Initially, when contamination levels require the use of bulky protective clothing and of respiration devices, the time spent in a radiation zone might be only 4 hours per shift. As the decontamination work progresses, the time spent in a radiation zone is assumed to increase to 6 hours per shift.) The processing and packaging of radioactive wastes and the performance of radiation surveys are assumed to involve 6 hours of radiation zone work per 8-hour shift.

Worker exposure hours are multiplied by an estimated average dose rate for each cleanup task to obtain the total occupational dose for the task. The dose rate for some cleanup tasks will initially be much higher than the assumed average, but cleanup efforts will result in a reduction in dose rate as decontamination proceeds.

The total estimated occupational radiation dose for accident cleanup in the auxiliary and fuel buildings is about 1600 man-rem.

E.3.4 Staff Requirements for Accident Cleanup in the Auxiliary and Fuel Buildings

The cleanup worker requirement for accident cleanup in the auxiliary and fuel buildings is 129 man-years, based on the cleanup schedule shown in Figure E.3-1. This estimate of cleanup worker requirements includes only the labor required to actually complete the tasks shown in the figure and does not include the extra labor needed to maintain compliance with occupational radiation dose limits. The adjustment in cleanup worker requirements to comply with occupational radiation dose limits is shown in Table E.3-3. An

TABLE E.3-2. Estimated Occupational Radiation Doses for Accident Cleanup in the Auxiliary and Fuel Buildings[(a)]

Cleanup Task	Average Dose Rate (rem/hr)	Crew Leader Exposure (man-hr)	Crew Leader Dose (man-rem)	Utility Operator Exposure (man-hr)	Utility Operator Dose (man-rem)	Laborer Exposure (man-hr)	Laborer Dose (man-rem)	Craftsman Exposure (man-hr)	Craftsman Dose (man-rem)	Health Physics Technician Exposure (man-hr)	Health Physics Technician Dose (man-rem)	Task Totals Dose (man-hr)	Task Totals Exposure (man-rem)
Decontaminate Fuel Building	0.010	3 600	36.0	7 200	72.0	7 200	72.0	3 600	36.0	3 600	36.0	25 200	252
Remove Accumulated Fuel from Spent Fuel Pool	0.010	222	2.2	444	4.5	444	4.5	--(b)	--	222	2.2	1 332	14
Install and Test Demineralizer System	0.010	800	8.0	1 600	16.0	1 600	16.0	3 200	32.0	800	8.0	8 000	80
Flush CVCS Tanks & Process Contaminated Liquids	0.010	480	4.8	960	9.6	960	9.6	--	--	480	4.8	2 880	29
Decontaminate Auxiliary Building	0.015	5 000	75.0	10 000	150.0	10 000	150.0	5 000	75.0	5 000	75.0	35 000	525
Process and Package Wastes from Cleanup Operations	0.005	6 240	31.2	12 480	62.4	12 480	62.4	--	--	6 240	31.2	37 440	187
Construction and Maintenance Support	0.010	5 200	52.0	--	--	--	--	41 600	416.0	5 200	52.0	52 000	520
Survey Auxiliary and Fuel Buildings	0.005	240	1.2	--	--	480	2.4	--	--	480	2.4	1200	6
Totals		21 782	210	32 684	315	33 164	317	53 400	559	22 022	212	163 052	1613

(a) Number of figures shown is for computational accuracy only.
(b) A dash indicates that, for the specified task, that particular staff category is not used.

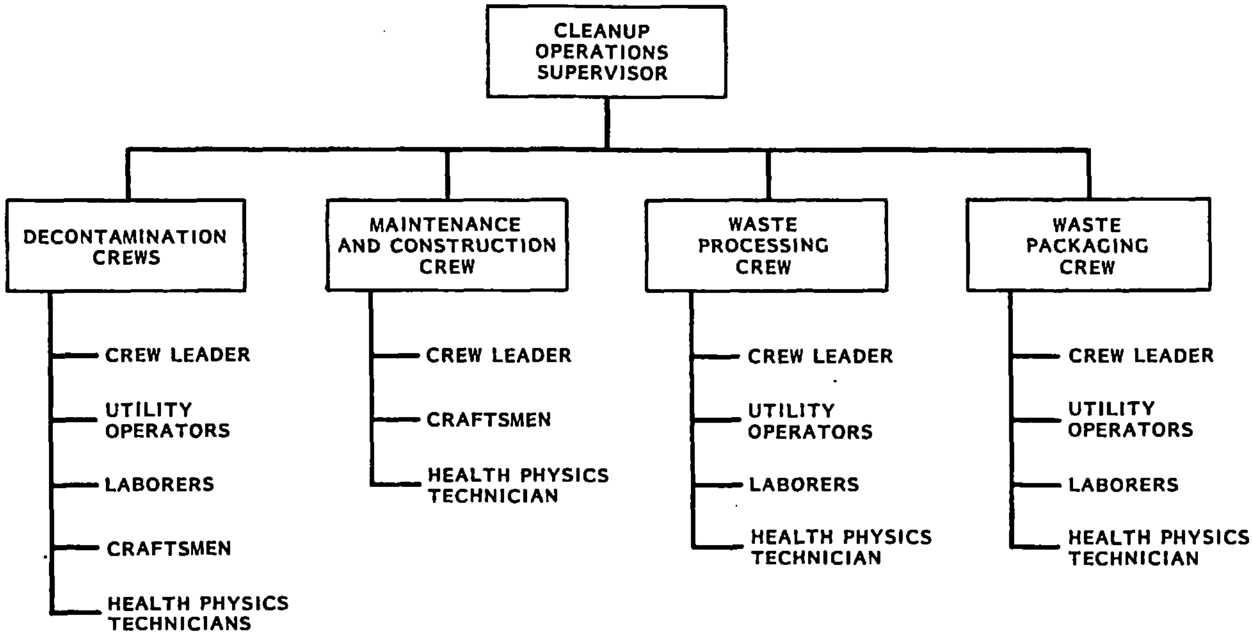


FIGURE E.3-2. Postulated Cleanup Operations Staff for Accident Cleanup in the Auxiliary and Fuel Buildings

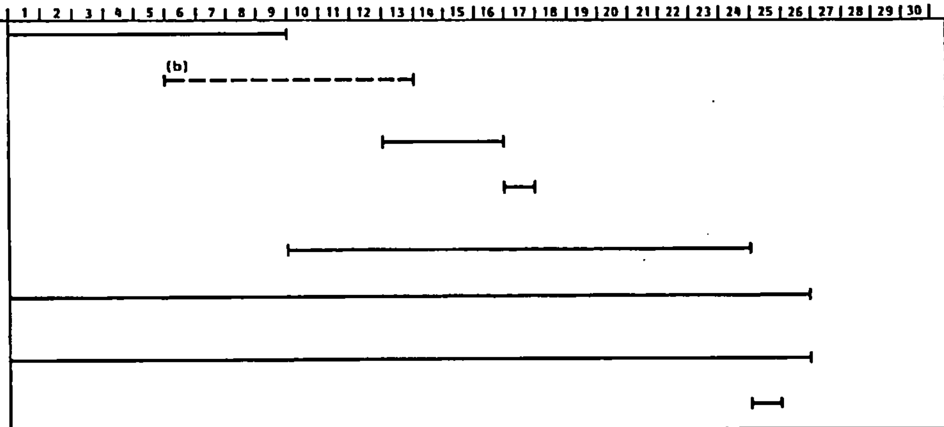
radiation dose limit[a] in short periods of time. Staff labor requirements for accident cleanup must therefore be increased to include the additional manpower needed to maintain compliance with occupational dose limits. Occupational radiation doses resulting from accident cleanup in the auxiliary and fuel buildings are discussed in Section E.3.3. Cleanup worker requirements, adjusted for occupational dose, that are used to compute labor costs for accident cleanup in the auxiliary and fuel buildings are discussed in Section E.3.4.

E.3.3 Occupational Doses for Accident Cleanup in the Auxiliary and Fuel Buildings

Estimated occupational radiation doses to cleanup workers during accident cleanup in the auxiliary and fuel buildings following the scenario 3 accident are given in Table E.3-2. The radiation doses shown in the table are external doses from gamma radiation. Workers are assumed to use respiration devices as necessary to protect against the inhalation of radioactive particulates. Cleanup workers include crew leaders, decontamination workers who operate

(a) The occupational radiation dose is assumed to be limited to 5 rem/year.[7]

TIME (MONTHS) AFTER START OF ACCIDENT CLEANUP OF AUXILIARY AND FUEL BUILDINGS



CLEANUP TASK (SHIFTS PER DAY/DURATION IN MONTHS)		MAN DAYS PER SHIFT[a]: CREW LEADER	UTILITY OPERATOR	LABORER	CRAFTSMAN	HEALTH PHYSICS TECHNICIAN
DECONTAMINATE FUEL BUILDING	(2/9)	2	4	4	2	2
REMOVE ACCUMULATED FUEL FROM SPENT FUEL POOL	((b)/10)					
INSTALL AND TEST DEMINERALIZER SYSTEM	(2/4)	1	2	2	4	1
FLUSH CVCS TANKS AND PROCESS CONTAMINATED LIQUIDS	(3/1)	1	2	2		1
DECONTAMINATE AUXILIARY BUILDING	(2[c]/15)	2	4	4	2	2
PROCESS AND PACKAGE WASTES FROM CLEANUP OPERATIONS	(2/26)	1	2	2		1
CONSTRUCTION AND MAINTENANCE SUPPORT	(2/26)	1			8	1
SURVEY AUXILIARY AND FUEL BUILDINGS	(2/1)	1		2		2

MAN MONTHS PER WORKING MONTH[e]

LABOR CATEGORY[d]	TOTAL MAN MONTHS	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30
CREW LEADER	204	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	10	8	8	8	8	8	8	8	6	4				
UTILITY OPERATOR	300	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	16	12	12	12	12	12	12	12	4	4				
LABORER	304	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	16	12	12	12	12	12	12	12	8	4				
CRAFTSMAN	534	20	20	20	20	20	20	20	20	20	20	20	20	26	26	26	26	18	20	20	20	20	20	20	20	16	16				
HEALTH PHYSICS TECHNICIAN	206	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	10	8	8	8	8	8	8	8	8	4				

(a) ASSUME 20 MAN DAYS PER MAN MONTH.
(b) THIS TASK PERFORMED BY PACKAGING CREW.
(c) ONE SHIFT PER DAY DURING MONTHS 13 THROUGH 17.
(d) SHIFT SUPERVISORS AND CRAFT SUPERVISORS NOT INCLUDED IN TASK WISE ASSESSMENT.
(e) MANPOWER REQUIREMENTS SHOWN ARE ROUNDED TO THE NEAREST 0.5 MAN MONTH. REQUIREMENTS ARE BASED ON NECESSARY LABOR TO COMPLETE TASKS AND DO NOT INCLUDE EXTRA MANPOWER NEEDED TO COMPLY WITH OCCUPATIONAL DOSE LIMITS; SEE SECTION E.3.3.

FIGURE E.3-1. Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Auxiliary and Fuel Buildings exposure rates at the conclusion of accident cleanup are postulated to be different for the three reference accident scenarios.

E.4.1.1 Processing of Contaminated Liquids

Contaminated liquids that must be processed during accident cleanup include containment building sump 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, the curies of radioactivity removed from these liquids, and estimated minimum processing time requirements are given in Section E.4.2.

Processed containment building sump water is reused for accident cleanup operations such as the decontamination of building surfaces by high-pressure hose wash and the flushing of the reactor coolant system. Reuse of this water (which still contains the tritium that was present prior to processing) reduces the total volume of water requiring interim onsite storage and ultimate disposal. Reuse also prevents the water level in the reactor building basement from continuing to rise during building decontamination operations. Reuse does not reduce the total volume of water to be processed since the same water may be processed more than once.

In this study, it is assumed that the processed water, if it is not needed for reuse, can be discharged to the river under controlled conditions. Prior to discharge of the water, processing would have reduced the contamination to values below the limits discussed in Section 5.3 of Chapter 5. This study recognizes that following an accident there is a potential that restrictions might be imposed against the discharge of the processed water. Other alternatives for disposal of the water are discussed in Section 5.3, but are not treated in detail in this study. However, a discussion of their relative costs is included in the discussion of cost sensitivity in Section 11.6 of Chapter 11.

A general discussion of options for the treatment of contaminated liquids is presented in Appendix D. In this report, 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 liquids since it was designed for processing water with significantly lower concentrations of radioactivity (less than 0.1 Ci/m^3) than exist in accident water or in some water-based decontamination solutions. The processing of contaminated water from the reference accidents requires the design and installation of a new demineralizer system which is described in the following paragraphs.

Chemical decontamination solutions that contain relatively high concentrations of chemicals and detergents are not suitable for processing by ion exchange methods because the chemicals and detergents in these liquids would cause decomposition of the ion exchange media. The processing of these liquids by evaporation is also described in this subsection.

Demineralizer System for the Treatment of Contaminated Water. The postulated filter/demineralizer system for the treatment of contaminated water is patterned after the Submerged Demineralizer System (SDS) used to process accident water at TMI-2.[6] The system is designed to operate under water for shielding and cooling and is installed in the spent fuel pool in the fuel building during preparations for accident cleanup. Approximately 4 months are estimated to be required to install and test the system after the detailed design is completed and necessary components are procured. It is assumed that the post-accident condition of the fuel building allows the use of the spent fuel pool for installation of the system, and the use of the CVCS holdup tanks (245-m^3 capacity each tank) and monitor tanks (90-m^3 capacity each tank) as feed and monitor tanks for the system. If the fuel building is contaminated as a result of the accident, accident cleanup of this building as described in Section E.3 would be required prior to installation and operation of the filter/demineralizer system.

A process diagram of the system is shown in Figure E.4-1. A submersible/centrifugal pump with a pumping capacity of 0.12 m^3/min is installed in the containment building basement and piping connections are made to the

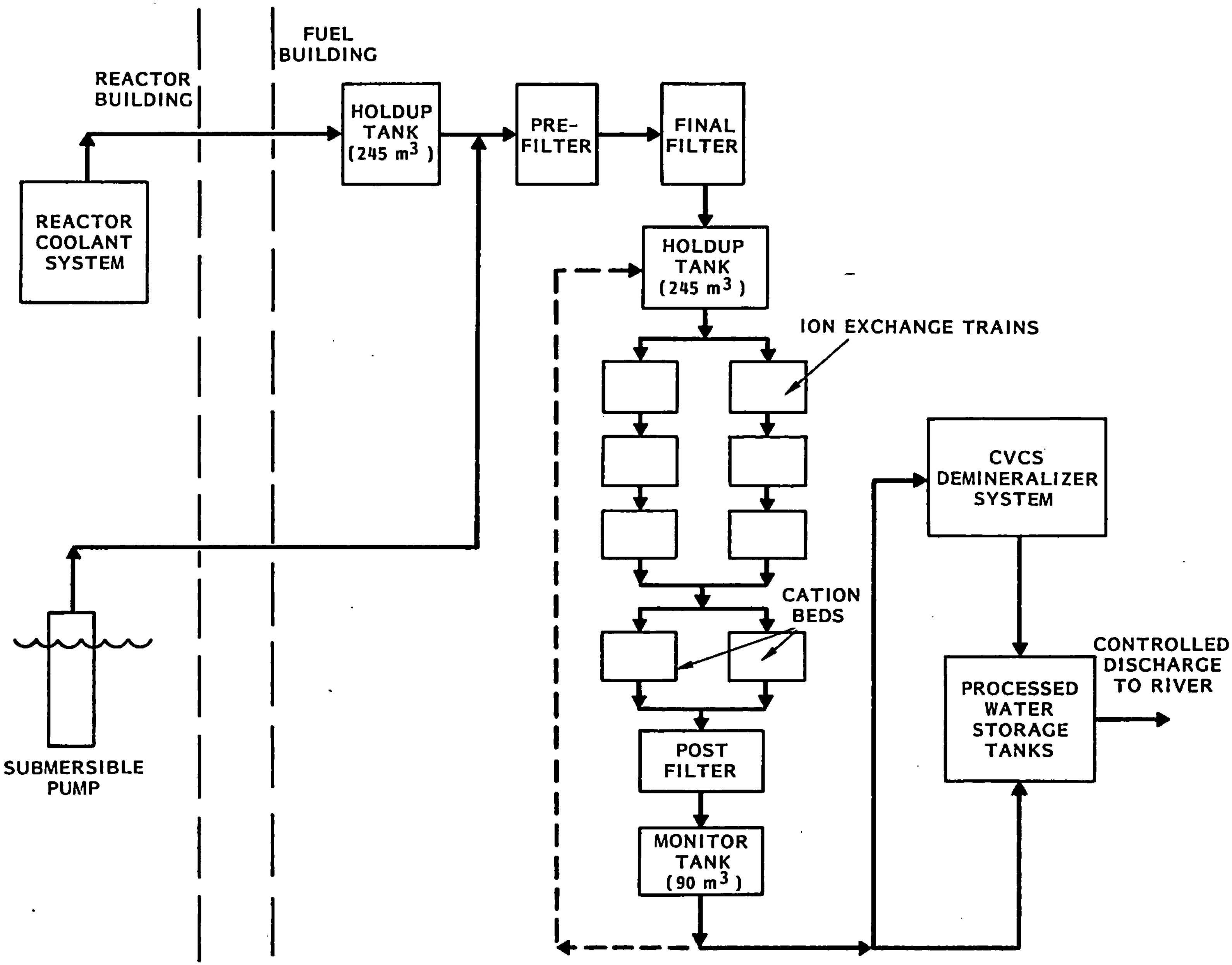


FIGURE E.4-1. Process Diagram of Filter/Demineralizer System for Contaminated Water Treatment filter/demineralizer system. The process train consists of a prefilter, final filter, holdup tank, two parallel trains of three ion exchange vessels each, two downstream ion exchange vessels in parallel, and a post filter.

Initial filtration is performed by two filters, a prefilter and a final filter. The prefilter is a roughing filter designed to remove suspended solids greater than 125 microns in size, and the final filter is designed to remove particles greater than 10 microns in size that pass through the prefilter. The filters are disposable cartridges in stainless steel tanks about 0.6 m in diameter and 1.4 m long with a volume of 0.3 m^3. Following initial filtration, the liquid is transferred to a batching tank.

The next step involves processing through two parallel trains of three ion exchange vessels each. These vessels are also 0.6 m in diameter by 1.4 m long and are constructed of stainless steel. Each vessel contains about 0.3 m^3 of zeolite ion exchange media. The design flow rate through each train is 0.02 m^3/min and the design flow through both trains is 0.04 m^3/min. Processing through these two trains is designed to remove over 99% of the cesium and strontium in the water. An ion exchange vessel is replaced when the radioactivity loading reaches approximately 60,000 Ci.

Further removal of cesium and strontium, plus removal of other ions, results from processing through two downstream parallel cation vessels, each containing 0.25 m^3 of organic resins. These vessels are essentially the same size as the zeolite vessels and are constructed of stainless steel.

The final unit in the process train is a post filter. The post filter is a stainless steel cylinder approximately 0.3 m in diameter and 1 m long designed to remove particles greater than 0.45 microns diameter.

The design objective of this demineralizer system is a radionuclide concentration less than 0.0001 Ci/m^3 in the processed water. If additional polishing of the liquid effluent from the system is required to achieve this objective, the evaporator feed ion exchangers in the CVCS demineralizer system are used. (The CVCS is described in Section B.2.)

Processed water is stored in the 1000-m^3 capacity storage tanks constructed onsite during preparations for accident cleanup. The processed water is either reused for building decontamination and reprocessed, or is assumed in this study to be discharged to the river under controlled conditions. Prior to discharge, water containing boron is processed by evaporation to remove the boron. The clean radioactive waste evaporator located in the auxiliary building is postulated to be used for this operation. This waste evaporator is described in Section B.2.

As filters or ion exchange media are expended, the vessel containing the material is removed and replaced. The vessel is flushed with processed water, removed from the system, dewatered, capped, placed in storage racks in the spent fuel pool, and attached to a vent header on the gaseous waste treatment system to relieve any potential buildup of non-condensible gas.

Treatment of Chemical Decontamination Solutions. 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 rented from a commercial supplier and is 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 postulated to be 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.

E.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.

Prior to the initial decontamination of containment building surfaces and equipment, teams of workers make short, carefully planned entries into the containment to inspect for damage, make radiation surveys, and install the submersible pump and other equipment needed to pump contaminated water from the containment building basement to the demineralizer system installed in the fuel building.

A general discussion of procedures for the physical cleaning of surfaces and equipment is given in Appendix D. For initial decontamination of the containment building, the following sequence of operations is postulated:

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."

Remote spray and hose-wash decontamination operations inside the building are carefully coordinated with sump water processing operations to maintain an approximately constant water level in the building basement until the hosing of surfaces is completed. Processed accident water is used for washing operations. Water from these operations is collected in the building basement.

There are three principal reasons for maintaining an approximately constant water level in the building basement during initial surface decontamination operations:

1. Decontamination workers can be unnecessarily exposed to high levels of gamma radiation from contaminated basement surfaces if the water shielding these surfaces is removed prior to their decontamination.
2. Removal of the basement water much in advance of the decontamination of basement surfaces results in a hardening of the radioactive "crud" deposits on these surfaces which makes decontamination more difficult.
3. An increase in the basement water level is undesirable because it causes the additional contamination of surfaces and equipment and may result in some additional equipment becoming inoperable because of water damage.

After high-pressure hosing of containment building surfaces above the basement level is completed, the water remaining in the building basement is processed through the demineralizer system installed in the fuel storage pool. As the water level in the basement is lowered, basement surfaces are washed with high-pressure hoses to remove surface contamination and "crud" deposits.

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 surface decontamination operations in the containment building begin. The processed water is then returned to the building basement to provide shielding from the surface contamination that exists on basement surfaces. Water used in remote spray and high pressure hose wash operations at the operating floor level is processed as it is generated to maintain an approximately constant water level in the building basement during these operations.

Additional details of initial decontamination operations in the PWR containment building are given in the following paragraphs.

Containment Building Spray System. If the building spray system is operable, the initial decontamination activity for cleanup of the containment building after the scenario 2 and scenario 3 accidents is a remote wash of surfaces and equipment using the containment building spray system. This system can be operated from controls located outside the building. The containment spray system is similar to the sprinkler systems used in public buildings, except that the flow of water is much greater. The primary advantage of the remote wash is that it provides an initial reduction in radiation level with very little worker exposure. The disadvantages are that a large volume of contaminated water is generated and not all parts of the building are decontaminated. A decontamination factor of 2 to 5 is estimated for the remote wash using the building spray system.

The containment spray system is assumed to be operated for short "bursts" of 2 minutes duration. Each use of the spray system generates 42 m^3 of water. Entry into the containment is made following each use of the spray system to monitor the effectiveness of this operation. Four 2-minute spray bursts over a 2-week period are assumed for remote decontamination following the scenario 2 accident. Six 2-minute spray bursts over a 2-week period are assumed for remote decontamination following the scenario 3 accident. Remote decontamination of the containment using the building spray system is not postulated to be necessary following the scenario 1 accident.

Removal of Nonessential Items. The removal from the containment building of small, nonessential items and debris serves to reduce the general background radiation level and also clears away materials that can impede the progress of the accident cleanup effort. Nonessential items include contaminated tools, loose equipment, barrels, boxes, staging, cables, hoses, wood pallets, etc. For decontamination following the scenario 3 accident, damaged pipes, cable conduits, and other damaged equipment and fixtures that interfere with decontamination operations are cut into sections and removed. Items that are removed from the containment building are wrapped in plastic and packaged as low-specific-activity waste for disposal at a shallow-land burial ground.

Semiremote Decontamination. Semiremote decontamination involves the use of equipment that permits the worker to stay some distance from the radiation source. High-pressure hose wash is postulated as the method for semiremote decontamination of the PWR containment building.

As a decontamination method, hose wash offers several advantages in terms of flow rate control, flow pattern, and directional properties. These factors are especially advantageous for the decontamination of hard-to-reach areas. However, because of low impact forces, if the surface being cleaned is covered with oil or grease, ordinary hose wash is ineffective. High-pressure water blasting equipment is commercially available that operates at pressures up to 70 MPa with water delivery rates of the order of 0.1 m^3/min. (A water delivery rate of 0.05 m^3/min is postulated for the high-pressure hose wash equipment in this study.) High-pressure water hoses are effective in removing oil and grease deposits. Depending on conditions and equipment, hose-wash decontamination factors range from 2 to 100.

After completion of the high-pressure hose washdown of building surfaces and equipment, a radiation survey is performed to assess the effectiveness of this decontamination procedure. The survey includes sample removals for laboratory analysis as well as air, water, and area radiation surveys.

Refurbish or Replace Essential Support Systems. Moderate contamination of containment building surfaces and equipment is postulated following the scenario 1 accident. Severe contamination of building surfaces and equipment including the ventilation system is postulated for the scenario 2 and scenario 3 accidents. In addition, physical damage to ventilation system components, electrical systems and other essential support systems is postulated for the scenario 3 accident.

To provide adequate ventilation for cleanup workers, contaminated and damaged ventilation system components must be replaced or decontaminated and repaired. Contaminated filters are replaced. Because of the difficulty of cleaning contaminated ductwork, it is assumed that ventilation ductwork is replaced with new ductwork as required to maintain adequate ventilation inside the containment building. Building fans and cooling units may be decontaminated or replaced. The four control rod drive (CRD) ventilation fans situated on top of the missile shield are removed and packaged for disposal. This is necessary to permit access to the reactor pressure vessel head so that the reactor can be defueled.

During accident cleanup following the scenario 3 accident, electrical cables, control panels, motors, relays, and switches are replaced as needed to maintain essential electrical services. Some replacement of electrical components is also required during accident cleanup following the scenario 2 accident. Because the polar crane is required for defueling operations, it is decontaminated and refurbished.

Hands-On Decontamination. Hands-on decontamination is minimized by first using remote and semiremote decontamination techniques. During accident cleanup, hands-on efforts are limited to wiping and scouring that must be performed to reduce radiation exposure to workers who will be engaged in reactor defueling and in reactor coolant system decontamination operations. The major hands-on decontamination effort during accident cleanup involves cleaning paths from the personnel airlock and the equipment hatch to areas around the reactor vessel head and the fuel transfer canal where defueling operations are performed.

Decontamination of the polar crane is accomplished by using the crane beams as a staging platform.

Decontamination of floors is accomplished by scrubbing with brushes or industrial floor scrubbers and a commercial decontamination agent, and then wet vacuuming or mopping to remove the resulting solution. The wash solution is stored in 0.21-m^3 drums until it is solidified for disposal. A final reagent/rinse mopping completes the effort.

Shielding of "Hot Spots". Shielding may be required to protect workers during accident cleanup operations, or it may be required to reduce radiation exposure from "hot spots" when initial decontamination of an area is completed. A low density shielding material, such as wood or plastic, can be used as a shield for low-energy beta radiation. A thin layer of aluminum or steel can be used as a shield for high-energy beta radiation. To achieve a reduction in gamma radiation, lead blankets, lead sheet, lead brick, or high density concrete blocks may be interposed between the gamma source and the work area. Containers filled with water may also serve as temporary shielding materials. Shielding materials are packaged or covered with plastic or a strippable coating to prevent their contamination.

Shielding can be used over gratings and open stairwells at the operating floor level to reduce worker exposure to gamma radiation from basement contamination.

E.4.1.3 Defueling the Reactor

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

For the scenario 1 accident, core damage is limited to slight damage to some fuel elements as a result of fuel swelling and cladding rupture. About 10% of the fuel rods are assumed to be affected. There is no damage to reactor pressure vessel internal components or the reactor vessel.

For the scenario 2 accident, core damage is assumed to include cladding rupture of about 50% of the fuel rods and cracking and crumbling of some fuel pellets. Fragmented fuel is distributed throughout the core. Some of the central fuel assemblies are bound or fused together at the spacer grid elevations and cannot be individually removed from the core. A few of the fuel assemblies are damaged to the extent that they cannot be removed by lifting the top end fitting. There is minimal damage to vessel internals and no damage to the reactor vessel.

Extensive damage to the fuel and to the reactor vessel and vessel internals is postulated for the scenario 3 accident. All of the fuel is assumed to experience cladding failure. Fuel damage includes cracking, crumbling, and melting of fuel pellets, warping of fuel assemblies, and melting and fusing together of stainless steel parts on adjacent assemblies. Fuel and cladding fragments are distributed throughout the core and the primary coolant system. Vessel internals are warped and cracked. The reactor vessel head bolts are jammed so that they cannot be removed by normal procedures.

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.

Removal of the core baffle and lower core support structure is only considered part of accident cleanup for the scenario 3 accident where cutting of the baffle is necessary to provide access for the removal of damaged fuel elements and removal of the lower core support structure is necessary to completely remove fuel fragments and cladding debris from the reactor vessel. For the SAFSTOR decommissioning alternative these internal components are left inside the reactor vessel unless damage to the fuel or to the reactor vessel necessitates their removal.

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

- assemble equipment
- install work platforms
- install temporary radiation shielding
- remove the insulation from the reactor vessel head
- install underwater lights
- decontaminate external surfaces of the reactor pressure vessel (RPV) head

- disconnect electrical cables and cooling water lines
- check whether the control rod drive shafts can be disconnected from the rod cluster assemblies
- clean up primary system water
- prepare the refueling cavity for flooding.

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 prior to defueling the reactor is accomplished through a "feed and bleed" process whereby water is removed from the system, processed by the demineralizer equipment, 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. The next steps include flooding of the refueling canal, removal of the reactor vessel head, and operation of the refueling canal cleanup system.

In addition to the usual equipment needed for RPV head and fuel element removal, special cutting and grappling tools are required for the removal of damaged components. For the scenario 2 and scenario 3 accidents, the exact nature of some special equipment cannot be known until the RPV head is removed and the reactor core is inspected. Equipment needed for inspection of the reactor includes TV cameras, periscopes, and underwater lighting.

To aid in defueling the reactor following the scenario 3 accident, a full-scale mockup of a section of the reactor pressure vessel and reactor core is constructed. This mockup is used for evaluating procedures for the removal of damaged fuel and internal components and for training workers in the use of special tools and equipment needed for defueling operations.

<u>Removal of RPV Head and Inspection of the Core</u>. For removal of the RPV head and inspection of the core, the following steps are postulated:

- remove the RPV head closure nuts and bolts

- uncouple the control rod drive shafts from the control rod cluster assemblies
- fill the refueling cavity with water
- remove the reactor pressure vessel head
- inspect the reactor core.

Under normal conditions, the removal of the RPV head closure nuts and bolts involves loosening the closure nuts from the studs after cleaning corrosion products from the threads and lubricating the threads. Removal is accomplished by applying a tensioning force to the stud and loosening and removing the nuts. For stud removal following the scenario 3 accident, loosening requires splitting or stripping jammed nuts and cutting off or machining out some of the difficult studs.

For the scenario 1 and scenario 2 accidents, the control rod cluster assemblies are disconnected from the control rod drives using normal procedures. For the scenario 3 accident, some control rods are assumed to be stuck in the reactor core or plenum grid, thus complicating their disconnection from the control rod drives. Disconnection of these control rods requires cutting the control rod drive shafts and/or lead screws.

After the RPV head closure bolts are removed and the drive shafts are uncoupled from the rod cluster assemblies, the refueling cavity is filled with water in preparation for removal of the RPV head. Processed accident water that has been stored in the 1000-m^3 storage tanks may be borated and used to fill the refueling cavity.

Under best case conditions (assumed for the scenario 1 and scenario 2 accidents) the RPV head can be lifted using the normal head lifting fixture and polar crane. After the head is in the refueling cavity, the inside of the head is decontaminated by flushing. For the scenario 3 accident, the control rod drives have been removed by cutting them off. Rigging is attached to the vessel head and secured to the polar crane. Jacking equipment is installed and the head is jacked up until it is separated from the reactor pressure vessel. The polar crane is then used to lift the pressure vessel head. (Decontamination and refurbishment of the polar crane is postulated to be part of the initial decontamination operations following the scenario 3 accident, as described in Section E.4.1.2.)

After removal of the RPV head, the inside 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. Several months may be required for the design, construction, and testing of the special tools and equipment needed for damaged fuel removal, thus delaying the start of this operation.

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. Before lifting, a periscope and a television camera are used to inspect for core debris or damage to the assembly. Core debris is removed by using water suction vacuum equipment, grapples, tongs, and water flushing techniques. Dummy control rod followers are used as necessary to hold down control rod assemblies during support assembly removal.

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 and package the pieces in canisters for interim onsite storage.

Removal of Fuel Assemblies. Prior to the removal of the fuel assemblies, any fuel or structural debris on the top surfaces of these assemblies is removed. Small pieces of fuel or structural debris are removed by hydraulic vacuuming. Larger pieces are removed with the aid of grapples or tongs. Material that has become fused to the tops of fuel assemblies is knocked or scraped off.

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 by means of a handling device that attaches to the top end fitting. (Equipment and procedures for the normal removal of undamaged PWR fuel assemblies are described in Section B.1.4.) Assemblies that have experienced structural damage that might cause them to break apart as they are lifted from the top are extracted from the core by the use of a special handling device that provides support to the bottom and sides of the assembly. Fuel assemblies are stored in fuel racks in the spent fuel storage pool. Damaged assemblies are packaged in canisters prior to being transferred from the refueling cavity to the spent fuel storage pool.

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. Special equipment is required to remove these assemblies. Several assemblies might be removed as a unit, or equipment might be used to cut them apart at the points where they are fused together.

Examples of conceptual handling devices for supporting damaged fuel assemblies on the bottom and the sides, and a conceptual procedure for packaging these assemblies are illustrated in Figures E.4-2, E.4-3, and E.4-4. These conceptual devices are adapted from descriptions given in Reference 9.

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. For this scenario, additional procedures and special equipment are required to open a full-length cavity on the periphery of the core in order 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

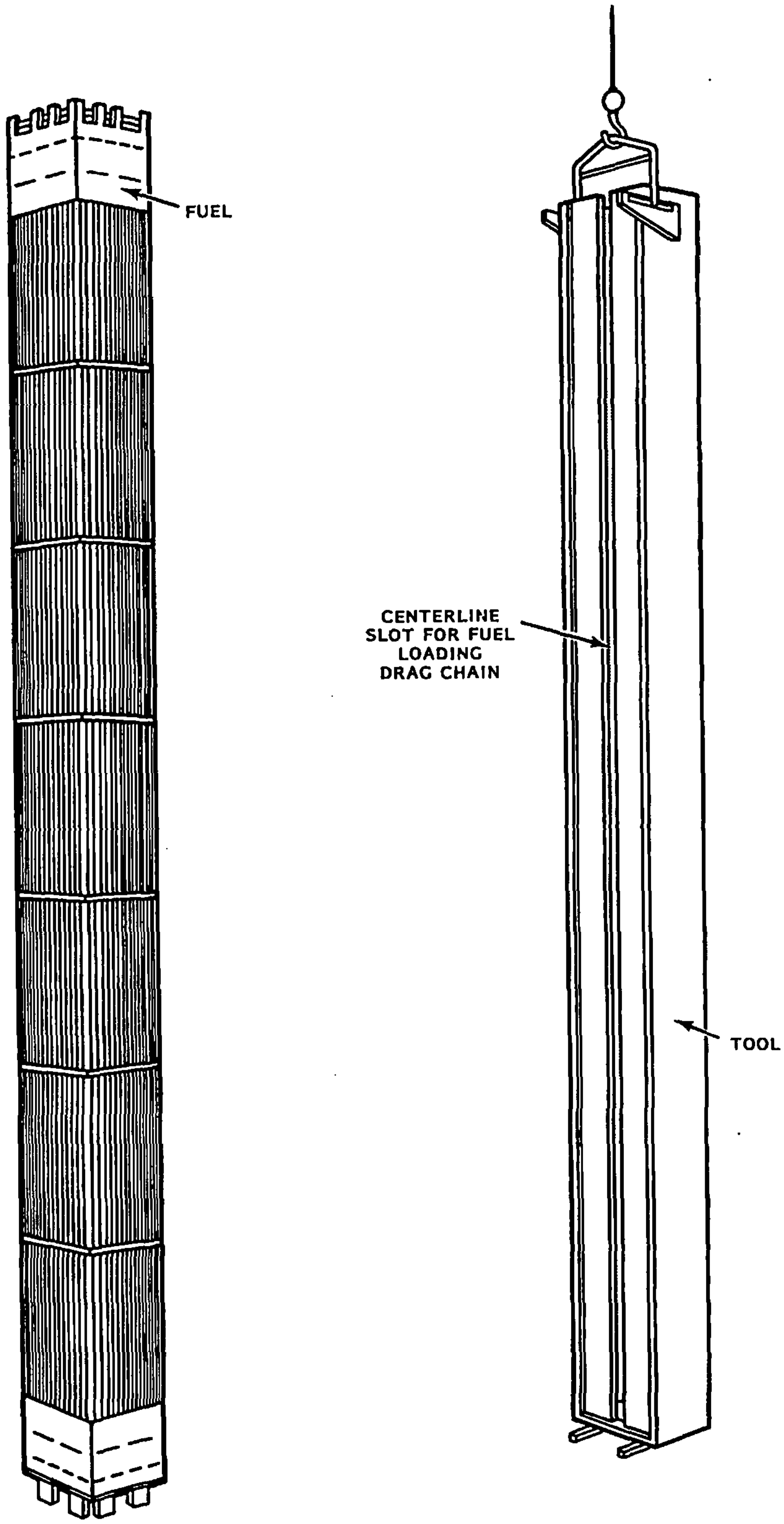


FIGURE E.4-2. Special Tool for Bottom Removal of Damaged Fuel (Adapted from Reference 9.)

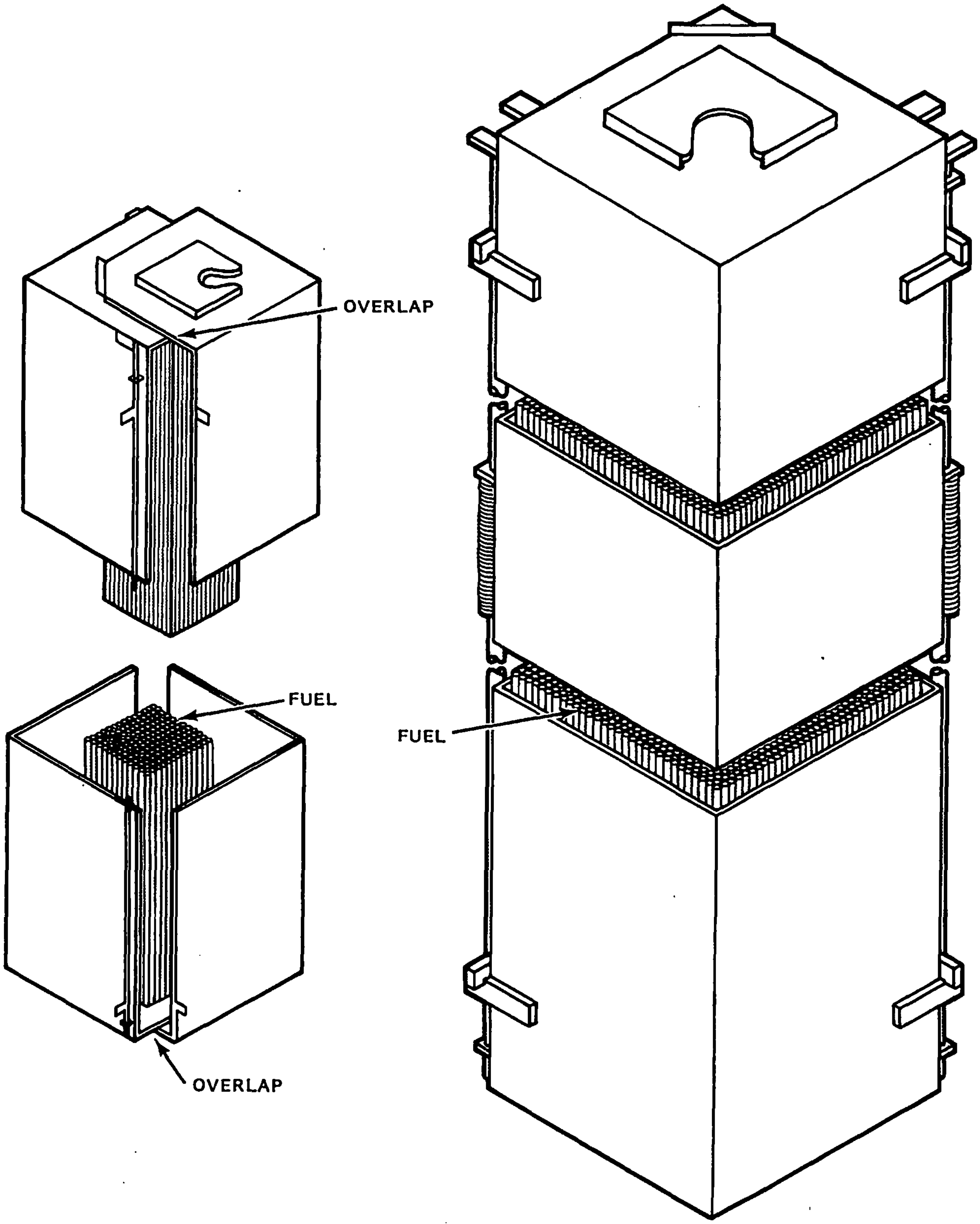


FIGURE E.4-3. Handling Shroud Concept for Removal of Damaged Fuel (Adapted from Reference 9.)

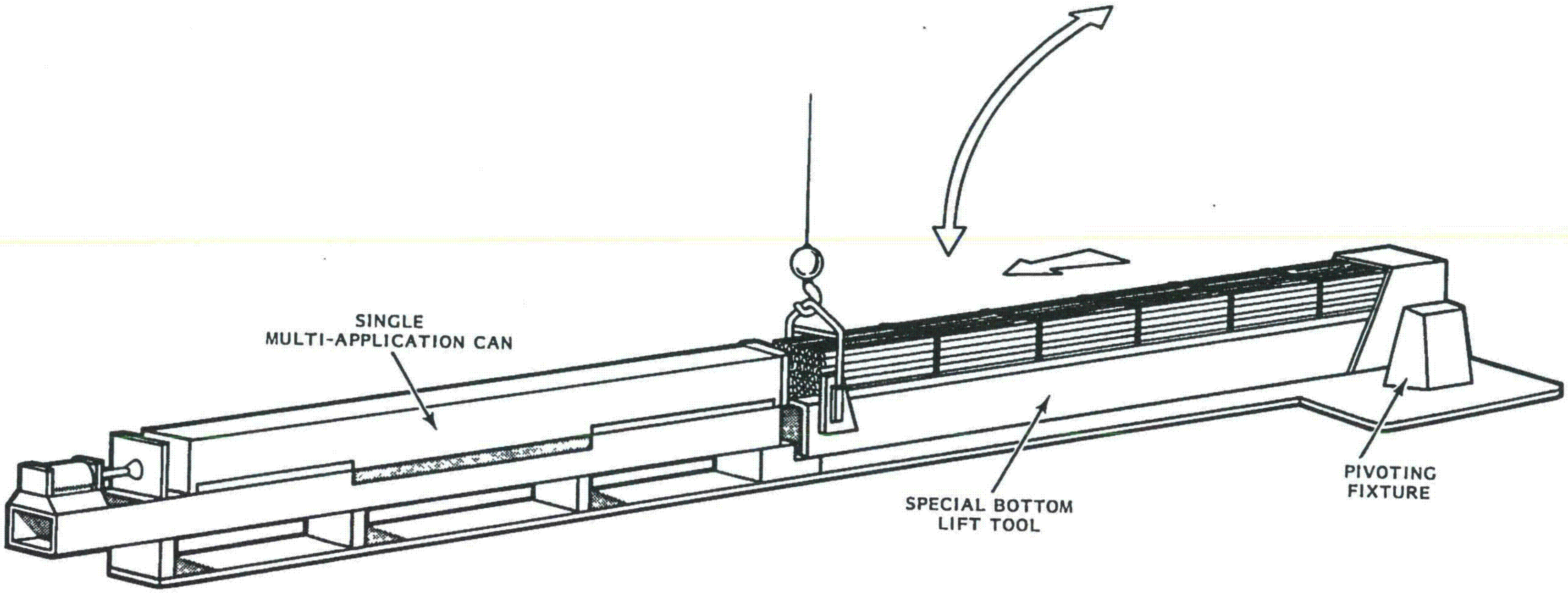


FIGURE E.4-4. Pivoting Fixture for Packaging Damaged Fuel Assemblies (Adapted from Reference 9.)

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.

All of the fuel assemblies removed from the core following the scenario 2 and scenario 3 accidents require packaging in canisters prior to their storage in the spent fuel pool. Canning of these assemblies is performed in the refueling cavity.

E.4.1.4 Cleanup of the Primary Coolant System

Before decontamination of the primary coolant system begins, reactor defueling is completed and the RPV head is reinstalled on the reactor pressure vessel. The water is drained from the refueling cavity and the fuel transfer canal, and both of these components are decontaminated by flushing. 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. Primary coolant system decontamination includes the removal of fuel debris (distributed throughout the system during the 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 system components, an oxalic-peroxide-gluconic (OPG) solution[a] is used. Relatively short circulation times (approximately 4 hours at a temperature of 80°C) are required. However, several system volumes may be needed to achieve the desired decontamination. For this study, one system volume is assumed to be required for fuel dissolution following the scenario 2 accident and two system volumes are assumed to be required for fuel dissolution following the scenario 3 accident.

For the removal of fission product plateout, the chelating agent ethylenediaminetetraacetic acid (EDTA) is used in combination with citric and

(a) The chemical equation for OPG solution is $Na_2C_2O_4$. It consists of 0.025 M $H_2C_2O_4$, 0.5 M H_2O_2, 0.013 M glucconic acid, and 0.045 M sodium gluconate at a pH of 4.5.

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.) The use of EDTA/oxalic/citric acid solutions for the removal of plateout from internal surfaces of coolant system components is discussed in Appendix D. While extended circulation times are required (approximately 1 week), only one system volume of the solution is needed. (Two system flushes with EDTA solution are postulated for cleanup following the scenario 3 accident.) Because of the incompatibility of the OPG and EDTA/ oxalic/citric acid solutions, a system flush with processed water is interposed between the two decontamination steps. A final system flush with processed water completes this decontamination procedure.

The reactor coolant system pumps are assumed to be operable and are used for circulation of the decontamination and flush solutions following the scenario 1 and scenario 2 accidents. Extensive repairs to pump motors are assumed to be necessary prior to the use of these pumps following the scenario 3 accident. During circulation of the solutions, drain valves are opened to flush particulates to cartridge-type, inline filters. Flushing and circulation of the decontamination solutions are largely remote operations.

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. The solutions are circulated by sequencing the opening of various drain valves. Upon completion of a decontamination sequence, the solution is drained from the primary coolant system to holdup tanks in the fuel building for processing through the evaporator/solidification system previously installed in the auxiliary building.

E.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 for the three accident scenarios are given in Table E.4-2.

Trash consists of compactible and noncompactible solid material, some of which is also combustible. The compactible and combustible solids consist of disposable clothing, rags, plastic covers, laydown pads, and miscellaneous trash. The noncompactible solids consist of tools, hoses, safety goggles, miscellaneous construction materials, and other small items of equipment used by decontamination personnel. The compactible trash is processed through a compactor to reduce the volume by a factor of 5, and is packaged in 0.21-m^3 steel drums. Approximately 75% of the compactible trash is also combustible. Incineration of this material is assumed to reduce the volume by a factor of 100, but immobilization of the resultant ash increases the volume by a factor of 2, resulting in an effective volume reduction factor of 50. Noncompactible and noncombustible trash is packaged in 1.2 m by 1.2 m by 2.4 m wooden LSA boxes with a capacity of 3 m^3 each.

A relatively small volume of contaminated equipment is packaged for disposal during accident cleanup. Equipment requiring disposal includes

Waste Type	Treatment Option(a)	Packa Type)	ge ce ion b)	Untreated Volume (m^3)	Treated Volume (m^3)	Number of Packages	Average Radioactivity per Package (Ci)	Average Surface Radiation Level (b) (R/hr)
				Scenario 3 Accident				
Sludge	Immobilization in cement	0.21-m^3 steel drum	.5	4.8	8.0	40	25	20
Process Solids								
Containment Bldg Sump and Wash Water								
Filter Cartridges		0.3-m^3 stainless s		7.5	7.5	25	100	80
Zeolite Liners		0.3-m^3 stainless s		15.0	15.0	50	50 000	40 000
Organic Resin Liners		0.3-m^3 stainless s		15.0	15.0	50	250	200
RCS Water								
Filter Cartridges		0.3-m^3 stainless s		0.6	0.6	2	60	48
Zeolite Liners		0.3-m^3 stainless s		1.2	1.2	4	32 000	26 000
Organic Resin Liners		0.3-m^3 stainless s		1.2	1.2	4	150	120
RCS Flush Water								
Filter Cartridges		0.3-m^3 stainless s		2.4	2.4	8	50	40
Zeolite Liners		0.3-m^3 stainless s		3.6	3.6	12	32 000	26 000
Organic Resin Liners		0.3-m^3 stainless s		2.4	2.4	8	250	200
RCS Decontamination Solutions								
Filter Cartridges	Dewater	0.21-m^3 steel drum		0.8	1.7	8	250	200
Evaporator Bottoms	Immobilization in cement	2.85-m^3 steel line		76	126	63	3 125	2 500
Organic Resins	Immobilization in cement	0.21-m^3 steel drum		200	360	1 700	1.2	1
Chemical Decontamination Solution	Immobilization in vinyl ester styrene	0.21-m^3 steel drum	.045	300	750	3 750	0.133	0.100
Trash								
Compactible, Combustible	Combustion with immobilization of ash	0.21-m^3 steel drum						
Compactible, Noncombustible	Compaction	0.21-m^3 steel drum	.280	11 881	250	1 190	0.35	0.280
Noncompactible		3.5-m^3 plywood box	.028	3 844	807	3 845	0.035	0.028
			.084	1 747	2 040	583	0.105	0.084
Contaminated Equipment								
LSA Materials	Disassembly and sectioning	plywood box(c)	.16	170	170	45	0.20	0.16
High-Activity Materials	Disassembly and sectioning	2.85-m^3 steel lin		40	40	16	10	8
Irradiated Hardware								
LSA Materials	Disassembly and sectioning	3.5-m^3 plywood box	8	24	24	8	1	0.8
High-Activity Materials	Disassembly and sectioning	2.85-m^3 steel lin		152	152	54	75 000	60 000
Fuel Assemblies								
Accumulated Spent Fuel		Bare assembly				258		
Intact Assemblies from Defueling		Bare assembly						
Damaged Assemblies from Defueling	Overpacking in steel canister	0.3-m^3 stainless s				193		
Fuel Core Debris		0.3-m^3 stainless s		3.6	3.6	12		

(a) All waste shipments are by truck except fuel assembly and fuel core debris shi
(b) Based on 0.8 R/hr per curie.
(c) Four boxes are 12 m^3 ea for CRDM cooling fans. The remaining boxes are 3.5 m^3
(d) Seven drums per cask. Two casks per truck shipment.
(e) One liner per cask. Two casks per truck shipment.
(f) Cask has dimensions of 1.63 m OD by 2.34 m high. One liner per cask. One cask
(g) A sole use van is assumed to transport 120 steel drums or six 3.5-m^3 plywood b
(h) Two plywood boxes per cask. One cask per truck shipment.
(i) The IF-300 rail cask can accommodate 7 bare fuel assemblies or 4 canistered fu

\BLE E.4-2. Packaging and Disposal Requirements for Radioactive Wastes from PWR Accident Cleanup ventilation filters, ductwork, conduit, and small items that are removed from containment during initial decontamination operations. Major equipment items such as steam generators, pumps, tanks, motors, and heat exchangers are disassembled and packaged for disposal during the decommissioning phase that follows accident cleanup. Waste disposal requirements for these major equipment items are described in Appendix H.

The major item of irradiated hardware removed from containment during accident cleanup is the upper core support assembly. A remotely operated arc saw is used to section this assembly after it is removed from the reactor vessel and placed in the refueling cavity. Segments are packaged in steel boxes and stored onsite in a shielded interim storage facility. Other irradiated hardware items that are sectioned and packaged for storage or disposal include the control rod drive mechanisms and control rod lead screws. The control rods are removed with the fuel and stored in the spent fuel storage pool.

As indicated in Section E.4.1.1, contaminated accident water and water-based decontamination solutions are treated by filtration and ion exchange processes. Packaging and disposal requirements for the process solid wastes from the treatment of contaminated water are given in Table E.4-2. The treated effluent from processing of this water is not listed in these tables. Processed water is assumed to be stored onsite in the 1000-m^3 water storage tanks constructed for this purpose and is postulated in this study to be discharged to the river in a controlled fashion to limit the concentration of radioactivity in the river to values consistent with the requirements of 10 CFR Part 20 and the EPA's Primary Drinking Water Standards.[8]

E.4.2 Schedules and Cleanup Worker Requirements for Accident Cleanup in the Containment Building

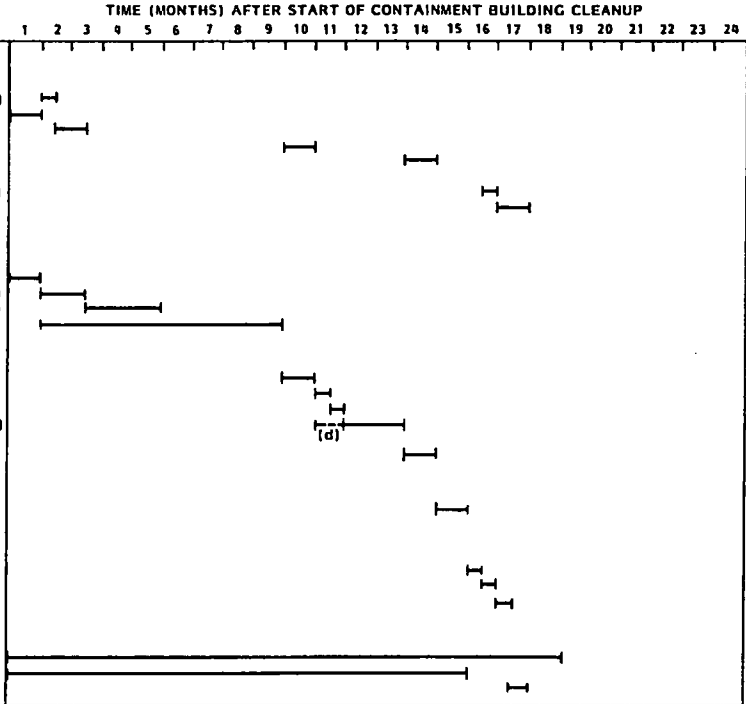
Task schedules and sequences and cleanup worker requirements for accident cleanup in the containment building following the three postulated accidents are shown in Figures E.4-5 through E.4-7. Work schedules and cleanup worker estimates vary with accident severity and are based on the cleanup procedures described in Section E.4.1. Accident cleanup in the containment building is estimated to require approximately 1.5 years following the scenario 1 accident, approximately 2.8 years following the scenario 2 accident, and approximately 5.0 years following the scenario 3 accident. Time requirements for accident cleanup are measured from the start of water processing and building decontamination operations and do not include the estimated 1 to 3 years required for preparations for accident cleanup as discussed in Section E.2.

Accident cleanup is performed by a cleanup staff defined in this report as those workers with assignments that take them inside the containment building or in other radiation areas where work is performed that relates to containment building cleanup. The cleanup staff includes all personnel shown in Figure E.4-8 (see Section E.4.4) that report to the cleanup superintendent plus health physics technicians assigned to provide radiation monitoring support and craftsmen that provide construction, equipment maintentance, and equipment disassembly support to the cleanup workers.

Cleanup crews include decontamination crews, waste processing crews, and waste packaging and shipping crews. Decontamination crews perform the accident cleanup tasks described in Section E.4.1. Crew sizes and estimated time requirements for the completion of these cleanup tasks are shown in Figures E.4-5, E.4-6, and E.4-7. Waste processing and waste packaging crews are responsible for the treatment and conditioning of the radioactive wastes that result from accident cleanup operations and for packaging these wastes for interim onsite storage or for shipment to an offsite storage or disposal location. The waste processing and waste packaging crews also assist the decontamination crews as required. Health physics technicians are added to the cleanup crews to provide radiation monitoring support.

Craftsmen perform the following functions inside the containment building during accident cleanup:

- maintain and repair systems and equipment needed to keep the reactor in a safe shutdown condition
- maintain and repair systems and equipment needed for cleanup operations



CLEANUP TASK	(SHIFTS PER DAY/DURATION IN MONTHS)	MAN DAYS PER SHIFT[a]: CREW LEADER	UTILITY OPERATOR	LABORER	CRAFTSMAN	HEALTH PHYSICS TECHNICIAN
PROCESSING OF CONTAMINATED LIQUIDS						
REMOTE WASHDOWN SPRAY WATER	(-/)					
CONTAINMENT BUILDING SUMP WATER	(3/0.5)		2	1		
OPERATING AREA HOSE WASH WATER	(3/1)		2	1		
BASEMENT AREA HOSE WASH WATER	(3/1)		2	1		
PRIMARY COOLANT SYSTEM WATER	(3/1)		2	1		
REFUELING CAVITY WATER	(3/1)		2	1		
OPG DECONTAMINATION SOLUTION	(-/-)					
EDTA DECONTAMINATION SOLUTION	(3/0.5)		2	1		
PRIMARY COOLANT SYSTEM FLUSH WATER	(3/1)		2	1		
INITIAL DECONTAMINATION OF CONTAINMENT BUILDING						
REMOTE WASHDOWN USING BUILDING SPRAY SYSTEM	(-/-)					
HIGH PRESSURE HOSE WASH OF OPERATING AREA	(2/1)	1	5	3	2	1
HIGH PRESSURE HOSE WASH OF BASEMENT AREA	(1/1.5)	1	5	3	2	1
DECONTAMINATE AND REPAIR SUPPORT SYSTEMS	(2/2.5)	1	4	2	4	1
HANDS-ON DECONTAMINATION (INCL. FLOORS)	(2/8)	1/2(b)	2/6(b)	2/4(b)		1/2(b)
DEFUELING OF THE REACTOR						
MOBILIZATION FOR DEFUELING	(2/1)	1	3	2	2	1
RPS HEAD REMOVAL AND CORE INSPECTION	(2/0.5)	1	3	2	2	1
REMOVE UPPER CORE SUPPORT STRUCTURE	(2/0.5)	1	3	2	2	1
REMOVE FUEL	((C)/2)	1	6	2	2	2
REMOVE CORE DEBRIS	(-/-)					
DECONTAMINATE REFUELING CAVITY	(2/1)	1	3	2	2	1
CLEANUP OF THE PRIMARY COOLANT SYSTEM						
PREPARATIONS FOR COOLANT SYSTEM CLEANUP	(2/1)	1	6	3	2	1
MIX, INJECT AND CIRCULATE OPG SOLUTION	(-/-)					
DRAIN OPG SOLUTION	(-/-)					
CIRCULATE AND DRAIN RINSE SOLUTION	(-/-)					
MIX, INJECT AND CIRCULATE EDTA SOLUTION	(3/0.5)	1	2	1	1	1
DRAIN EDTA SOLUTION	(3/0.5)	1	2	1	1	1
CIRCULATE AND DRAIN RINSE SOLUTION	(3/0.5)	1	2	1	1	1
SUPPORT OPERATIONS						
WASTE PROCESSING AND PACKAGING	(2/18)	1	2	2		1
CONSTRUCTION AND MAINTENANCE	(1/15)	1			6	1
FINAL RADIATION SURVEY	(2/0.5)	1		2		4

LABOR CATEGORY	TOTAL MAN MONTHS	MAN MONTHS PER WORKING MONTH[e]: 1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24
CREW LEADER	120.5	5	6	6.5	7	7	7	7	7	7	5	11	11	11	5	5	6	5	2						
UTILITY OPERATOR	401.5	22	21	18.5	16	16	16	16	16	16	18	52	52	52	18	16	16	16	4						
LABORER	234.5	14	15	13.5	12	12	12	12	12	12	12	20	20	20	12	10	10	12	4						
CRAFTSMAN	195.0	10	8	11	14	14	6	6	6	6	10	22	22	22	10	10	10	8							
HEALTH PHYSICS TECHNICIAN	147.5	5	6	6.5	7	7	7	7	7	7	5	19	19	19	5	5	7	9							

(a) ASSUME 20 MAN DAYS PER MAN MONTH.

(b) DECONTAMINATION CREW CONSISTS OF 1 CREW LEADER, 2 UTILITY OPERATORS, 2 LABORERS, AND 1 HEALTH PHYSICS TECHNICIAN PER SHIFT DURING MONTHS 2 THROUGH 5; 2 CREW LEADERS, 6 UTILITY OPERATORS, 4 LABORERS, AND 2 HEALTH PHYSICS TECHNICIANS PER SHIFT DURING MONTHS 6 THROUGH 9.

(c) ONCE STARTED, REMOVAL OF FUEL CONTINUES UNUNTERRUPTED ON AN AROUND THE CLOCK BASIS UNTIL COMPLETED. THIS REQUIRES SIX SHIFTS PER DAY.

(d) CREW TRAINING FOR FUEL REMOVAL.

(e) MANPOWER REQUIREMENTS ARE SHOWN ROUNDED TO THE NEAREST 0.5 MAN MONTH. REQUIREMENTS ARE BASED ON NECESSARY LABOR TO COMPLETE TASKS AND DO NOT INCLUDE EXTRA MANPOWER NEEDED TO COMPLY WITH OCCUPATIONAL DOSE LIMITS; SEE SECTION E.4.3.

FIGURE E.4-5. Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Containment Building Following the Scenario 1 Accident

CLEANUP TASK (SHIFTS PER DAY/DURATION IN MONTHS)	REW LEADER	UTILITY OPERATOR	LABORER	CRAFTSMAN	HEALTH PHYSICS TECHNICIAN
			MAN DAYS PER SHIFT(a)		
PROCESSING OF CONTAMINATED LIQUIDS					
REMOTE WASHDOWN SPRAY WATER (3/		2	1		
CONTAINMENT BUILDING SUMP WATER (3/		2	1		
OPERATING AREA HOSE WASH WATER (3/		2	1		
BASEMENT AREA HOSE WASH WATER (3/		2	1		
PRIMARY COOLANT SYSTEM WATER (3/		2	1		
REFUELING CAVITY WATER (3/		2	1		
OPG DECONTAMINATION SOLUTION (3/		2	1		
EDTA DECONTAMINATION SOLUTION (3/		2	1		
PRIMARY COOLANT SYSTEM FLUSH WATER (3/		2	1		
INITIAL DECONTAMINATION OF CONTAINMENT BUILDI					
REMOTE WASHDOWN USING BUILDING SPRAY SYSTEM (		4	2	2	4
HIGH PRESSURE HOSE WASH OF OPERATING AREA (2/		5	3	2	1
HIGH PRESSURE HOSE WASH OF BASEMENT AREA (1/		5	3	2	1
DECONTAMINATE AND REPAIR SUPPORT SYSTEMS (2/		4	2	4	1
HANDS-ON DECONTAMINATION (INCL. FLOORS) (2/	b)	2/6(b)	2/4(b)		1/2(b)
DEFUELING OF THE REACTOR					
MOBILIZATION FOR DEFUELING (2/		3	2	2	1
RPV HEAD REMOVAL AND CORE INSPECTION (2/		3	2	2	1
REMOVE UPPER CORE SUPPORT STRUCTURE (2/		3	2	2	1
REMOVE FUEL ([C		6	2	2	2
REMOVE CORE DEBRIS (2/		6	2	2	2
DECONTAMINATE REFUELING CAVITY (2/		3	2	2	1
CLEANUP OF THE PRIMARY COOLANT SYSTEM					
PREPARATIONS FOR COOLANT SYSTEM CLEANUP (2/		6	3	4	1
MIX, INJECT AND CIRCULATE OPG SOLUTION (3/		2	1	1	1
DRAIN OPG SOLUTION (3/		2	1	1	1
CIRCULATE AND DRAIN RINSE SOLUTION (3/		2	1	1	1
MIX, INJECT AND CIRCULATE EDTA SOLUTION (3/		2	1	1	1
DRAIN EDTA SOLUTION (3/		2	1	1	1
CIRCULATE AND DRAIN RINSE SOLUTION (3/		2	1	1	1
SUPPORT OPERATIONS					
WASTE PROCESSING AND PACKAGING (2/		2	2		1
CONSTRUCTION AND MAINTENANCE (2/				6	1
FINAL RADIATION SURVEY (2/			2		4

LABOR CATEGORY	TOTAL MAN-MONTHS
CREW LEADER	250
UTILITY OPERATOR	740
LABORER	430
CRAFTSMAN	510
HEALTH PHYSICS TECHNICIAN	302

(a) ASSUME 20 MAN-DAYS PER MAN-MONTH.

(b) DECONTAMINATION CREW CONSISTS OF 1 CF
10; 2 CREW LEADERS, 6 UTILITY OPERATOR

(c) ONCE STARTED, REMOVAL OF FUEL CONTINI

(d) CREW TRAINING FOR FUEL REMOVAL.

(e) MANPOWER REQUIREMENTS ARE SHOWN ROUI
INCLUDE EXTRA MANPOWER NEEDED TO COM

GURE E.4-6. Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Containment Building Following the Scenario 2 Accident

CLEANUP TASK (SHIFTS PER DAY/DURATION IN MONTHS)		MAN DAYS PER SHIFT[a]: CREW LEADER	UTILITY OPERATOR	LABORER	CRAFTSMAN	HEALTH PHYSICS TECHNICIAN
PROCESSING OF CONTAMINATED LIQUIDS						
REMOTE WASHDOWN SPRAY WATER	(3/1)		2	1		
CONTAINMENT BUILDING SUMP WATER	(3/6)		2	1		
OPERATING AREA HOSE WASH WATER	(3/3)		2	1		
BASEMENT AREA HOSE WASH WATER	(3/2)		2	1		
PRIMARY COOLANT SYSTEM WATER	(3/2)		2	1		
REFUELING CAVITY WATER	(3/1)		2	1		
OPG DECONTAMINATION SOLUTION	(3/1.5)		2	1		
EDTA DECONTAMINATION SOLUTION	(3/1.5)		2	1		
PRIMARY COOLANT SYSTEM FLUSH WATER	(3/2)		2	1		
INITIAL DECONTAMINATION OF CONTAINMENT BUILDING						
REMOTE WASHDOWN USING BUILDING SPRAY SYSTEM	(1/1)	1	4	2	2	4
HIGH PRESSURE HOSE WASH OF OPERATING AREA	(2/3)	1	5	3	2	1
HIGH PRESSURE HOSE WASH OF BASEMENT AREA	(1/5)	1	5	3	2	1
DECONTAMINATE AND REPAIR SUPPORT SYSTEMS	(2/8)	1	4	2	4	1
HANDS-ON DECONTAMINATION (INCL. FLOORS)	(2/19)	1/2[b]	2/6[b]	2/4[b]		1/2[b]
DEFUELING OF THE REACTOR						
MOBILIZATION FOR DEFUELING	(2/2)	1	3	2	3	1
RPV HEAD REMOVAL AND CORE INSPECTION	(2/6)	1	3	2	3	1
REMOVE UPPER CORE SUPPORT STRUCTURE	(2/2)	1	3	2	3	1
REMOVE FUEL	((C)/12)	1	6	2	3	2
REMOVE CORE DEBRIS	(3/2)	1	4	2	3	1
DECONTAMINATE REFUELING CAVITY	(2/1)	1	3	2	2	1
CLEANUP OF THE PRIMARY COOLANT SYSTEM						
PREPARATION FOR COOLANT SYSTEM CLEANUP	(2/4)	1	6	3	4	1
MIX, INJECT AND CIRCULATE OPG SOLUTION	(3/1)	1	2	1	2	1
DRAIN OPG SOLUTION	(3/1)	1	2	1	2	1
CIRCULATE AND DRAIN RINSE SOLUTION	(3/0.5)	1	2	1	2	1
MIX, INJECT AND CIRCULATE EDTA SOLUTION	(3/1)	1	2	1	2	1
DRAIN EDTA SOLUTION	(3/1)	1	2	1	2	1
CIRCULATE AND DRAIN RINSE SOLUTION	(3/0.5)	1	2	1	2	1
SUPPORT OPERATIONS						
WASTE PROCESSING AND PACKAGING	(2/60)	1	2	2		1
CONSTRUCTION AND MAINTENANCE	(2/53)	1			8	1
FINAL RADIATION SURVEY	(2/2)	1		2		4

LABOR CATEGORY	TOTAL MAN-MONTHS	1	2	55	56	57	58	59	60	61	62
CREW LEADER	478	5	4	6	6	6	6	4	4		
UTILITY OPERATOR	1481	16	12	20	20	20	20	8	4		
LABORER	819	10	8	12	12	12	12	10	8		
CRAFTSMAN	1384	18	16	8	8	8	8				
HEALTH PHYSICS TECHNICIAN	571	8	4	6	6	6	6	10	10		

(a) ASSUME 20 MAN-DAYS PER MAN-MONTH.

(b) DECONTAMINATION CREW CONSISTS OF 1 CREW LEADER, 2 U
2 CREW LEADERS, 6 UTILITY OPERATORS, 4 LABORERS, AND

(c) ONCE STARTED, REMOVAL OF FUEL CONTINUES UNINTERRUP

(d) CREW TRAINING FOR FUEL REMOVAL

(e) MANPOWER REQUIREMENTS ARE SHOWN ROUNDED TO THE NE
INCLUDE EXTRA MANPOWER NEEDED TO COMPLY WITH OCCUF

FIGURE E.4-7. Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Containment Building Following the Scenario 3 Accident

- construct and install additional systems and equipment such as platforms, scaffolding, special tools, ventilation and contamination control systems, electrical systems, pumps, and motors, etc., needed for cleanup operations.

Craftsmen are added to decontamination crews as required to disassemble contaminated systems and equipment.

The following bases and assumptions are used in estimating time and manpower requirements for accident cleanup in the containment building:

1. In general, the time required for performance of a particular cleanup task is determined by estimating the time requirement for efficient performance of the task and doubling this time to account for inefficiencies associated with work in high radiation areas. Additional assumptions related to time requirements for particular tasks are given in the following paragraphs.
2. Decontamination workers are assumed to spend only 4 hours inside the containment building during an 8-hour shift. The remaining time is assumed to be required to put on and remove protective clothing, rehearse cleanup procedures, etc.
3. Most accident cleanup tasks are performed on a 2-shift, 5-day-week basis. Exceptions to this general rule are noted below.
4. High pressure hose-wash decontamination operations inside the containment are coordinated with sump water processing operations as described in Section E.4.1.2. The high-pressure hose wash of building surfaces proceeds at a rate corresponding to wash water processing, to maintain a constant water level in the building basement. Processing of contaminated water to reduce the water level in the basement begins after the high-pressure hose wash of building surfaces above the basement is completed. As the water level in the basement is lowered, newly exposed building and equipment surfaces are washed with high-pressure hoses to remove radioactive deposits adhering to these surfaces.

5. The processing of contaminated liquids and decontamination of the primary coolant system are continuous operations that proceed around the clock, 7-days-per-week.

6. The design processing rate for the demineralizer system used to process contaminated water is 57.6 m^3/day (0.04 m^3/min). The actual processing rate for this system is assumed to be approximately 1/3 of the design processing rate, or 20 m^3/day. This allows for delays due to equipment breakdown and for downtime required for changeout of resin liners. Similarly, the actual processing rate for the evaporator system used to process chemical decontamination solutions is assumed to be 30 m^3/day which is approximately 1/3 of the design processing rate of 86.4 m^3/day (0.06 m^3/min). Minimum time requirements for processing the contaminated liquids from accident cleanup are shown in Table E.4-3. In addition to the contaminated liquids shown in the table, approximately 1500 m^3 of decontaminated water must be processed by evaporation to remove the boron prior to controlled discharge to the river.

7. Defueling the reactor is a continuous operation. Once initiated, removal of the damaged fuel from the reactor core and transfer of the fuel to the fuel storage basin is performed by work crews operating on a 6-shift-per-day, 7-day-week basis. Defueling requires a large complement of trained personnel (a total of 56 crew leaders and utility operators) not normally available at the accident-damaged reactor station. Additional personnel are assumed to be temporarily assigned from other reactor stations in order to complete the defueling operation. Personnel on temporary assignment are paid a per diem in addition to their regular salaries. This per diem is included in the labor costs for accident cleanup discussed in Appendix F.

8. To estimate time requirements for defueling operations, the following assumptions are made:

TABLE E.4-3. Minimum Time Requirements for Processing Contaminated Liquids from Accident Cleanup in the Containment Building[(a,b)]

Source of Contaminated Liquid	Processing Option	Scenario 1 Accident Volume (m^3)	Scenario 1 Accident Minimum Processing Time (days)	Scenario 2 Accident Volume (m^3)	Scenario 2 Accident Minimum Processing Time (days)	Scenario 3 Accident Volume (m^3)	Scenario 3 Accident Minimum Processing Time (days)
Reactor Building Sump Water	Demineralizer	200	10	1000	50	1600	80
Remote Washdown Spray Water	Demineralizer	--	--	168	9	252	13
Operating Area Hose Wash	Demineralizer	420	21	720	36	1200	60
Basement Area Hose Wash	Demineralizer	300	15	480	24	900	45
Primary Coolant System Water	Demineralizer	380	19	380	19	380	19
Refueling Cavity Water	Demineralizer	500	25	500	25	500	25
OPG Decontamination Solution	Evaporator	--	--	380	13	760	26
EDTA Decontamination Solution	Evaporator	380	13	380	13	760	26
Primary Coolant System Flush	Demineralizer	380	19	760	38	760	38

(a) Processing times are computed on the basis of assumed average processing rates of 20 m^3/day for the demineralizer system and 30m^3/day for the evaporator system.
(b) Processing is assumed to proceed around the clock on a continuous 24-hour-per-day, 7-day-week basis.

a) Removal and storage of an undamaged fuel assembly requires 2 hours including time for inspection with periscopes and underwater TV cameras prior to removal.

b) Removal and storage of a fuel assembly that has experienced cladding failure requires 8 hours including time to inspect the assembly prior to removal and to overpack the assembly in a stainless steel canister prior to storage.

c) Removal, overpacking, and storage of a fuel assembly that is damaged as a result of fuel melting requires 20 to 40 hours depending on the extent of the damage.

d) Time estimates for fuel removal based on the above assumptions are doubled to allow for unavoidable inefficiencies associated with work in high radiation areas.

9. To defuel the reactor following the scenario 3 accident, it is necessary to cut some of the core baffle plates as described in Section F.4.1.3. A plasma arc torch is used for this operation

which requires 20 hours of cutting time, including time for positioning the saw prior to each cut and relocating the saw out of the way before each segment is removed.

10. Plant operations related to maintenance of the reactor in a safe shutdown condition, plant security, and radiation protection activities are performed on a 24-hour-per-day, 7-day-week basis.

11. Training time for staff involved in cleanup operations (especially defueling operations) is included in time and manpower estimates for completion of the various cleanup tasks.

12. Cleanup following an accident provides unique opportunities for research in areas related to accident consequences (contamination dispersal mechanisms, fuel core evaluation, waste management requirements, etc.). However, in this report no scheduling allowances are made for research and development activities except those related to the design, fabrication, and testing of the special tools and equipment required for decontamination and defueling operations.

These assumptions provide the bases for the time schedules and cleanup worker requirements shown in Figures E.4-5, E.4-6, and E.4-7. Cleanup worker requirements shown in these figures include only the labor required to actually complete the cleanup tasks and to provide radiation monitoring and craft support to decommissioning workers and do not include the extra labor needed to maintain compliance with occupational radiation dose limits.[7] The occupational doses received by these workers are discussed in Section E.4.3. Cleanup worker requirements, adjusted to comply with occupational radiation dose limitations, are discussed in Section E.4.4 where total staff requirements for accident cleanup in the containment building are also discussed.

E.4.3 Occupational Doses for Accident Cleanup in the Containment Building

Estimated occupational radiation doses to cleanup workers during accident cleanup in the containment building following the three reference accidents are shown in Tables E.4-4, E.4-5, and E.4-6. The occupational doses shown in the tables are external doses from gamma radiation. Workers are assumed to use respiration devices as necessary to protect against the inhalation of radioactive particulates. Cleanup workers are those workers defined in Section E.4.2 as having work assignments in the containment building or in other radiation areas where work is performed (e.g., the processing and packaging the wastes) that is related to containment building cleanup.

Dose calculations are based on time and manpower requirements shown in the task schedules for accident cleanup in the containment building (Figures E.4-5, E.4-6, and E.4-7). Exposure hours are estimated on the basis that workers engaged in decontamination operations and in the installation and repair of systems needed for accident cleanup spend an average of 4 hours inside the containment building during an 8-hour shift. Workers who operate the demineralizer and evaporator systems, monitor the reactor coolant system cleanup operations, or are engaged in waste packaging activities spend an average of 6 hours in a radiation area during an 8-hour shift.

Total estimated occupational radiation doses to cleanup workers for accident cleanup in the containment building are 667 man-rem following the scenario 1 accident, 2923 man-rem following the scenario 2 accident, and 10,131 man-rem following the scenario 3 accident.

E.4.4 Staff Requirements for Accident Cleanup in the Containment Building

Cleanup worker requirements for accident cleanup in the containment building following the three reference accidents can be obtained from the task schedules for containment cleanup shown in Figures E.4-5, E.4-6, and E.4-7. Cleanup worker requirements are estimated to be 92 man-years following the scenario 1 accident, 186 man-years following the scenario 2 accident, and 395 man-years following the scenario 3 accident. These requirements include only the labor to actually complete the designated cleanup tasks and do not include the additional labor needed to maintain compliance with occupational radiation dose limits or the management and support staff required during cleanup operations.

TABLE E.4-4. Estimated Occupational Radiation Doses for Accident Cleanup in the Containment Building Following the Scenario 1 Accident

Cleanup Task	Average Dose Rate (rem/hr)	Crew Leader		Utility Operator		Laborer		Craftsman		Health Physics Technician		Task Totals	
		Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man/hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)
Processing of Contaminated Liquids													
Remote Washdown Spray Water	--												
Containment Building Sump Water	0.005			480	2.40	240	1.20					720	3.60
Operating Area Hose Wash Water	0.005			960	4.80	480	2.40					1 440	7.20
Basement Area Hose Wash Water	0.005			960	4.80	480	2.40					1 440	7.20
Primary Coolant System Water	0.005			960	4.80	480	2.40					1 440	7.20
Refueling Cavity Water	0.005			960	4.80	480	2.40					1 440	7.20
OPG Decontamination Solution	--												
EDTA Decontamination Solution	0.005			480	2.40	240	1.20					720	3.60
Primary Coolant System Flush Water				960	4.80	480	2.40					1 440	7.20
Subtotals				5 760	28.80	2 880	14.40					8 640	43.20
Initial Decontamination of Containment Building													
Remote Washdown Using Building Spray System	--												
High Pressure Hose Wash of Operating Area	0.015	160	2.40	800	12.00	480	7.20	320	4.80	160	2.40	1 920	28.80
High Pressure Hose Wash of Basement Area	0.025	120	3.00	600	15.00	360	9.00	240	6.00	120	3.00	1 440	36.00
Decontaminate & Repair Support Systems	0.008	400	3.20	1 600	12.80	800	6.40	1 600	12.80	400	3.20	4 800	38.40
Hands-on Decontamination (incl. Floors)	0.008	1 920	15.36	5 120	40.96	3 840	30.72			1 920	15.36	12 800	102.40
Subtotals		2 600	23.96	8 120	80.76	5 480	53.32	2 160	23.60	2 600	23.96	20 960	205.60
Defueling of the Reactor													
Mobilization for Defueling	0.005	160	0.80	480	2.40	320	1.60	320	1.60	160	0.80	1 440	7.20
RPV Head Removal & Core Inspection	0.010	80	0.80	160	1.60	160	1.60	160	1.60	80	0.80	640	6.40
Remove Upper Core Support Structure	0.010	80	0.80	160	1.60	160	1.60	160	1.60	80	0.80	640	6.40
Remove Fuel	0.010	1 280	12.80	7 680	76.80	2 560	25.60	2 560	25.60	2 560	25.60	16 640	166.40
Remove Core Debris	--												
Decontaminate Refueling Cavity	0.005	160	0.80	960	4.80	480	2.40	320	1.60	160	0.80	2 080	10.40
Subtotals		1 760	16.00	9 440	87.20	3 680	32.80	3 520	32.00	3 040	28.80	21 440	196.80
Cleanup of the Primary Coolant System													
Preparations for Coolant System Cleanup	0.005	160	0.80	960	4.80	480	2.40	320	1.60	160	0.80	2 080	10.40
Mix, Inject & Circulate OPG Solution	--												
Drain OPG Solution	--												
Circulate & Drain Rinse Solution	--												
Mix, Inject & Circulate EDTA Solution	0.005	240	1.20	480	2.40	240	1.20	240	1.20	240	1.20	1 440	7.20
Drain EDTA Solution	0.005	240	1.20	480	2.40	240	1.20	240	1.20	240	1.20	1 440	7.20
Circulate & Drain Rinse Solution	0.005	240	1.20	480	2.40	240	1.20	240	1.20	240	1.20	1 440	7.20
Subtotals		880	4.40	2 400	12.00	1 200	6.00	1 040	5.20	880	4.40	6 400	32.00
Support Operations													
Waste Processing & Packaging	0.005	3600	18.00	7200	36.00	7200	36.00			3 600	18.00	21 600	108.00
Construction & Maintenance	0.008	1200	9.60							1 200	9.60	9 600	76.80
Final Radiation Survey	0.005	120	0.60			240	1.20	7 200	57.60	480	2.40	840	4.20
Subtotals		4 920	28.20	7 200	36.00	7 440	37.20	7 200	57.60	5 280	30.00	32 040	189.00
Totals		10 160	72.56	32 920	244.76	20 680	143.72	13 920	118.40	11 800	87.16	89 480	666.60

TABLE E.4-5. Estimated Occupational Radiation Doses for Accident Cleanup in the Containment Building Following the Scenario 2 Accident

Cleanup Task	Average Dose Rate (rem/hr)	Crew Leader Exposure (man-hr)	Crew Leader Dose (man-rem)	Utility Operator Exposure (man-hr)	Utility Operator Dose (man-rem)	Laborer Exposure (man-hr)	Laborer Dose (man-rem)	Craftsman Exposure (man-hr)	Craftsman Dose (man-rem)	Health Physics Technician Exposure (man/hr)	Health Physics Technician Dose (man-rem)	Task Totals Exposure (man-hr)	Task Totals Dose (man-rem)
Processing of Contaminated Liquids													
Remote Washdown Spray Water	0.008			480	3.84	240	1.92					720	5.76
Containment Building Sump Water	0.008			3 840	30.72	1 920	15.36					5 760	46.08
Operating Area Hose Wash Water	0.008			1 440	11.52	720	5.76					2 160	17.28
Basement Area Hose Wash Water	0.008			960	7.68	480	3.84					1 440	11.52
Primary Coolant System Water	0.008			1 440	11.52	720	5.76					2 160	17.28
Refueling Cavity Water	0.008			960	7.68	480	3.84					1 440	11.52
OPG Decontamination Solution	0.008			480	3.84	240	1.92					720	5.76
EDTA Decontamination Solution	0.008			480	3.84	240	1.92					720	5.76
Primary Coolant System Flush Water	0.008			1 920	15.36	960	7.68					2 880	23.04
Subtotals				12 000	96.00	6 000	48.00					18 000	144.00
Initial Decontamination of Containment Building													
Remote Washdown Using Building Spray System	0.100	6	0.60	24	2.40	12	1.20	12	1.20	24	2.40	78	7.80
High Pressure Hose Wash of Operating Area	0.030	240	7.20	1 200	36.00	720	21.60	480	14.40	240	7.20	2 880	86.40
High Pressure Hose Wash of Basement Area	0.050	240	12.00	1 200	60.00	720	36.00	480	24.00	240	12.00	2 880	144.00
Decontaminate & Repair Support Systems	0.020	640	12.80	2 560	51.20	1 280	25.60	2 560	51.20	640	12.80	7 680	153.60
Hands-on Decontamination (incl. Floors)	0.020	2720	54.40	7 040	140.80	5 440	108.80			2 720	54.40	17 920	358.40
Subtotals		3846	87.00	12 024	290.40	8 172	193.20	3 532	90.80	3 864	88.80	31 438	750.20
Defueling of the Reactor													
Mobilize for Defueling	0.010	240	2.40	720	7.20	480	4.80	480	4.80	240	2.40	2 160	21.60
RPV Head Removal & Core Inspection	0.020	480	9.60	1 440	28.80	960	19.20	960	19.20	480	9.60	4 320	86.40
Remove Upper Core Support Structure	0.020	80	1.60	240	4.80	160	3.20	160	3.20	80	1.60	720	14.40
Remove Fuel	0.020	3200	64.00	19 200	384.00	6 400	128.00	6 400	128.00	6 400	128.00	41 600	832.00
Remove Core Debris	0.020	160	3.20	960	19.20	320	6.40	320	6.40	320	6.40	2 080	41.60
Decontaminate Refueling Cavity	0.010	160	1.60	480	4.80	320	3.20	320	3.20	160	1.60	1 440	14.40
Subtotals		4 320	82.40	23 040	448.80	8 640	164.80	8 640	164.80	7 680	149.60	52 320	1010.40
Cleanup of the Primary Coolant System													
Preparations for Coolant System Cleanup	0.015	240	3.60	1 440	21.60	720	10.80	960	14.40	240	3.60	3 600	54.00
Mix, Inject & Circulate OPG Solution	0.010	240	2.40	480	4.80	240	2.40	240	2.40	240	2.40	1 440	14.40
Drain OPG Solution	0.010	240	2.40	480	4.80	240	2.40	240	2.40	240	2.40	1 440	14.40
Circulate & Drain Rinse Solution	0.010	240	2.40	480	4.80	240	2.40	240	2.40	240	2.40	1 440	14.40
Mix, Inject & Circulate EDTA Solution	0.010	240	2.40	480	4.80	240	2.40	240	2.40	240	2.40	1 440	14.40
Drain EDTA Solution	0.010	240	2.40	480	4.80	240	2.40	240	2.40	240	2.40	1 440	14.40
Circulate & Drain Rinse Solution	0.010	240	2.40	480	4.80	240	2.40	240	2.40	240	2.40	1 440	14.40
Subtotals		1 680	18.00	4 320	50.40	2 160	25.20	2 400	28.80	1 680	18.00	12 240	140.40
Support Operations													
Waste Processing & Packaging	0.008	6 600	52.80	13 200	105.60	13 200	105.60			6 600	52.80	39 600	316.80
Construction & Maintenance	0.015	4 560	68.40					27 360	410.40	4 560	68.40	36 480	547.20
Final Radiation Survey	0.012	160	1.92			320	3.84			640	7.68	1 120	13.44
Subtotals		11 320	123.12	13 200	105.60	13 520	109.44	27 360	410.40	11 800	128.88	77 200	877.44
Totals		21 166	310.52	64 584	991.20	38 492	540.64	41 932	694.80	25 024	385.28	191 198	2922.44

TABLE E.4-6. Estimated Occupational Radiation Doses for Accident Cleanup in the Containment Building Following the Scenario 3 Accident

Cleanup Task	Average Dose Rate (rem/hr)	Crew Leader Exposure (man-hr)	Crew Leader Dose (man-rem)	Utility Operator Exposure (man-hr)	Utility Operator Dose (man-rem)	Laborer Exposure (man-hr)	Laborer Dose (man-rem)	Craftsman Exposure (man-hr)	Craftsman Dose (man-rem)	Health Physics Technician Exposure (man/hr)	Health Physics Technician Dose (man-rem)	Task Totals Exposure (man-hr)	Task Totals Dose (man-rem)
Processing of Contaminated Liquids													
Remote Washdown Spray Water	0.010			960	9.60	480	4.80					1 440	14.40
Containment Building Sump Water	0.010			5 760	57.60	2 880	28.80					8 640	86.40
Operating Area Hose Wash Water	0.010			2 880	28.80	1 440	14.40					4 320	43.20
Basement Area Hose Wash Water	0.010			1 920	19.20	960	9.60					2 880	28.80
Primary Coolant System Water	0.010			1 920	19.20	960	9.60					2 880	28.80
Refueling Cavity Water	0.010			960	9.60	480	4.80					1 440	14.40
OPG Decontamination Solution	0.010			1 440	14.40	720	7.20					2 160	21.60
EDTA Decontamination Solution	0.010			1 440	14.40	720	7.20					2 160	21.60
Primary Coolant System Flush Water	0.010			1 920	19.20	960	9.60					2 880	28.80
Subtotals				19 200	197.00	9 600	96.00					28 800	288.00
Initial Decontamination of Containment Building													
Remote Washdown Using Building Spray System	0.300	10	3.00	40	12.00	20	6.00	20	6.00	40	12.00	130	39.00
High Pressure Hose Wash of Operating Area	0.050	480	24.00	2 400	120.00	1 440	72.00	960	48.00	480	24.00	5 760	288.00
High Pressure Hose Wash of Basement Area	0.075	400	30.00	2 000	150.00	1 200	90.00	800	60.00	400	30.00	4 800	360.00
Decontaminate & Repair Support Systems	0.030	1 280	38.40	5 120	153.60	2 560	76.80	5 120	153.60	1 280	38.40	15 360	460.80
Hands-on Decontamination (incl. Floors)	0.030	4 000	120.00	9 920	297.60	8 000	240.00			4 000	120.00	25 920	777.60
Subtotals		6 170	215.40	19 480	733.20	13 220	484.80	6 900	267.60	6 200	224.40	51 970	1 925.40
Defueling of the Reactor													
Mobilization for Defueling	0.020	320	6.40	960	19.20	640	12.80	960	19.20	320	6.40	3 200	64.00
RPV Head Removal & Core Inspection	0.030	960	28.80	2 880	86.40	1 920	57.60	2 880	86.40	960	28.80	9 600	288.00
Remove Upper Core Support Structure	0.030	320	9.60	960	28.80	640	19.20	960	28.80	320	9.60	3 200	96.00
Remove Fuel	0.030	7 680	230.40	46 080	1 382.40	15 360	460.80	23 040	691.20	15 360	460.80	107 520	3 225.60
Remove Core Debris	0.030	640	19.20	2 560	76.80	1 280	38.40	1 920	57.60	640	19.20	7 040	211.20
Decontaminate Refueling Cavity	0.020	160	3.20	480	9.20	320	6.40	320	6.40	160	3.20	1 440	28.40
Subtotals		10 080	297.60	53 920	1 602.80	20 160	595.20	30 080	889.60	17 760	528.00	132 000	3 913.20
Cleanup of the Primary Coolant System													
Preparations for Coolant System Cleanup	0.030	640	19.20	3 840	115.20	1 920	57.60	2 560	76.80	640	19.20	9 600	288.00
Mix Inject & Circulate OPG Solution	0.015	480	7.20	960	14.40	480	7.20	960	14.40	480	7.20	3 360	50.40
Drain OPG Solution	0.015	480	7.20	960	14.40	480	7.20	960	14.40	480	7.20	3 360	50.40
Circulate & Drain Rinse Solution	0.015	240	3.60	480	7.20	240	3.60	480	7.20	240	3.60	1 680	25.20
Mix Inject & Circulate EDTA Solution	0.015	480	7.20	960	14.40	480	7.20	960	14.40	480	7.20	3 360	50.40
Drain EDTA Solution	0.015	480	7.20	960	14.40	480	7.20	960	14.40	480	7.20	3 360	50.40
Circulate & Drain Rinse Solution	0.015	240	3.60	480	7.20	240	3.60	480	7.20	240	3.60	1 680	25.20
Subtotals		3 040	55.20	8 640	187.20	4 320	93.60	7 360	148.80	3 040	55.20	26 400	540.00
Support Operations													
Waste Processing & Packaging	0.012	12 000	144.00	24 000	288.00	24 000	288.00			12 000	144.00	72 000	864.00
Construction & Maintenance	0.030	8 480	254.40					67 840	2 035.20	8 480	254.40	84 800	2 5444.00
Final Radiation Survey	0.025	320	8.00			640	16.00			1 280	32.00	2 240	56.00
Subtotals		20 800	406.40	24 000	288.00	24 640	304.00	67 840	2 035.20	21 760	430.40	159 040	3 464.00
Totals		40 090	974.60	125 240	3 003.20	71 940	1 573.60	112 180	3 341.20	48 760	1 238.00	398 210	10 130.60

Adjustments in cleanup worker requirements to comply with occupational radiation dose limits are shown in Tables E.4-7, E.4-8, and E.4-9 for the three reference accidents. Adjusted cleanup worker requirements are 138 man-years following the scenario 1 accident, 596 man-years following the scenario 2 accident, and 2039 man-years following the scenario 3 accident.

The postulated staff organization for accident cleanup in the containment building is shown in Figure E.4-8. This staff organization includes a plant operations branch and several support branches (e.g., engineering, health and safety, security, contracts and accounting, and quality assurance) as well as the cleanup staff.

Total utility staff labor requirements for accident cleanup in the containment building following the three reference accidents are shown in Table E.4-10. The requirements presented include the management and support staff as well as the cleanup workers but do not include contractor personnel. Management and support staff man-years are directly related to the time required for accident cleanup. Cleanup staff man-years shown in the table are the requirements adjusted to comply with occupational dose limitations. The staff labor man-years shown in Table E.4-10 are used in Appendix F to compute labor costs for accident cleanup in the containment building.

TABLE E.4-7. Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Containment Building Following the Scenario 1 Accident

Worker Category	Estimated Worker Requirements[a] (man-yr)	Estimated Occupational Dose: Total[b] (man-rem)	Estimated Occupational Dose: Individual Average (man-rem/man-yr)	Adjustment Factor[c]	Adjusted Worker Requirements (man-yr)
Crew Leader	10.1	72.6	7.2	1.5	15.2
Utility Operator	33.5	244.8	7.3	1.5	50.3
Laborer	19.6	143.7	7.4	1.5	29.4
Craftsman	16.3	118.4	7.3	1.5	24.5
Health Physics Technician	12.3	87.2	7.1	1.5	18.5
Totals	91.8	666.7			137.9

(a) Based on Figure E.4-5.
(b) Based on Table E.4-4.
(c) Increase in worker requirements necessary to reduce average individual dose to ≤ 5 man-rem/man-year.

TABLE E.4-8. Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Containment Building Following the Scenario 2 Accident

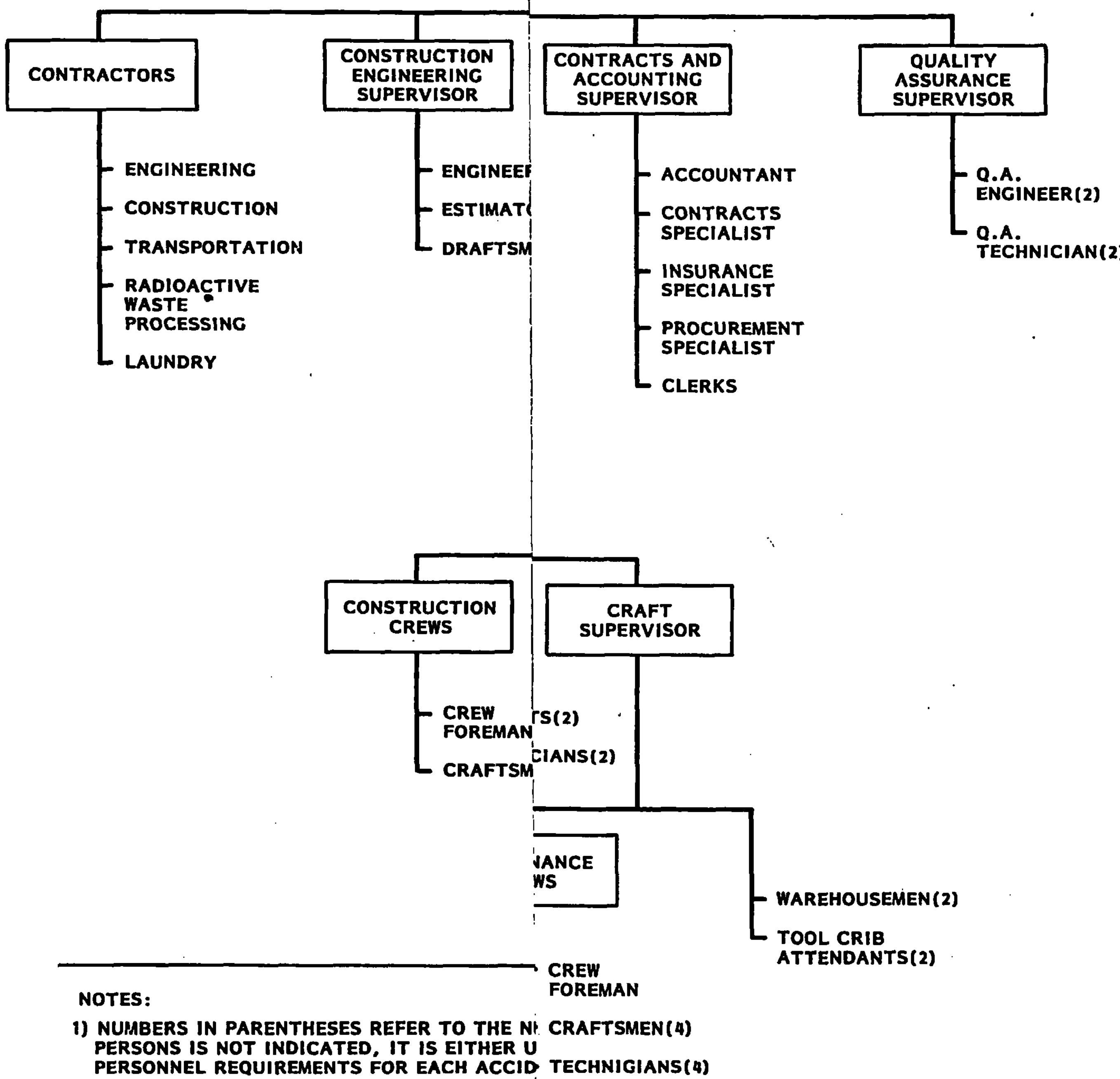
Worker Category	Estimated Worker Requirements[a] (man-yr)	Estimated Occupational Dose: Total[b] (man-rem)	Estimated Occupational Dose: Individual Average (man-rem/man-yr)	Adjustment Factor[c]	Adjusted Worker Requirements (man-yr)
Crew Leader	20.8	310.5	15.0	3.0	62.4
Utility Operator	61.7	991.2	16.1	3.3	203.6
Laborer	35.8	540.6	15.1	3.1	111.0
Craftsman	42.5	694.8	16.4	3.3	140.3
Health Physics Technician	25.2	385.3	15.3	3.1	78.2
Totals	186.0	2922.4			595.5

(a) Based on Figure E.4-6.
(b) Based on Table E.4-5.
(c) Increase in worker requirements necessary to reduce average individual dose to ≤ 5 man-rem/man-year.

TABLE E.4-9. Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Containment Building Following the Scenario 3 Accident

Worker Category	Estimated Worker Requirements[a] (man-yr)	Estimated Occupational Dose: Total[b] (man-rem)	Estimated Occupational Dose: Individual Average (man-rem/man-yr)	Adjustment Factor[c]	Adjusted Worker Requirements (man-yr)
Crew Leader	39.9	974.6	24.5	4.9	195.5
Utility Operator	123.5	3003.2	24.4	4.9	605.2
Laborer	68.3	1573.6	23.1	4.7	321.0
Craftsman	115.4	3341.2	29.0	5.8	669.4
Health Physics Technician	47.6	1238.0	26.0	5.2	247.6
Totals	394.7	10 130.6			2 038.7

(a) Based on Figure E.4-7.
(b) Based on Table E.4-6.
(c) Increase in worker requirements necessary to reduce average individual dose to ≤ 5 man-rem/man-year.



NOTES:

1) NUMBERS IN PARENTHESES REFER TO THE N PERSONS IS NOT INDICATED, IT IS EITHER U PERSONNEL REQUIREMENTS FOR EACH ACCID

2) THE SECURITY FORCE, THE HEALTH PHYSICS PLANT MAINTENANCE STAFF FUNCTION ON A

FIGURE E.4-8. Postulated Staff Organization for Accident Cleanup in the Containment Building

TABLE E.4-10. Estimated Utility Staff Labor Requirements for Accident Cleanup in the Containment Building Following the Postulated PWR Accidents

Position	Scenario 1 Accident man-years/year	Scenario 1 Accident Total man-years	Scenario 2 Accident man-years/year	Scenario 2 Accident Total man-years	Scenario 3 Accident man-years/year	Scenario 3 Accident Total man-years
Plant Superintendent	1.0	1.5	1.0	2.8	1.0	5.0
Assistant Plant Superintendent	1.0	1.5	1.0	2.8	1.0	5.0
Consultants	3.0	4.5	6.0	16.8	10.0	50.0
Secretaries & Word Processors	8.0	12.0	10.0	28.0	12.0	60.0
Site Support Staff						
Health and Safety Supervisor	1.0	1.5	1.0	2.8	1.0	5.0
Health Physicist	1.0	1.5	1.0	2.8	1.0	5.0
Senior Health Physics Technician	8.0	12.0	8.0	22.4	12.0	60.0
Health Physics Technician[a]	8.0	12.0	8.0	22.4	12.0	60.0
Protective Equipment Attendant	4.0	6.0	8.0	22.4	12.0	60.0
Industrial Safety Specialist	1.0	1.5	1.0	2.8	1.0	5.0
Industrial Safety Technician	2.0	3.0	2.0	5.6	2.0	10.0
Security Supervisor	1.0	1.5	1.0	2.8	1.0	5.0
Security Shift Supervisor	4.0	6.0	4.0	11.2	4.0	20.0
Security Patrolman	48.0	72.0	48.0	134.4	48.0	240.0
Contracts & Accounting Supervisor	1.0	1.5	1.0	2.8	1.0	5.0
Accountant	1.0	1.5	1.0	2.8	2.0	10.0
Contracts Specialist	1.0	1.5	1.0	2.8	1.0	5.0
Insurance Specialist	1.0	1.5	1.0	2.8	2.0	10.0
Procurement Specialist	1.0	1.5	1.0	2.8	1.0	5.0
Clerk	2.0	3.0	4.0	11.2	6.0	30.0
Quality Assurance Supervisor	1.0	1.5	1.0	2.8	1.0	5.0
Quality Assurance Engineer	2.0	3.0	2.0	5.6	2.0	10.0
Quality Assurance Technician	2.0	3.0	2.0	5.6	2.0	10.0
Construction Engineering Supervisor	1.0	1.5	1.0	2.8	1.0	5.0
Engineer	6.0	9.0	8.0	22.4	12.0	60.0
Estimator	1.0	1.5	2.0	5.6	4.0	20.0
Draftsman	2.0	3.0	4.0	11.2	6.0	30.0
Subtotals		150.0		310.8		675.0
Plant Operations Staff						
Plant Operations Supervisor	1.0	1.5	1.0	2.8	1.0	5.0
Plant Chemist	1.0	1.5	1.0	2.8	1.0	5.0
Chemist	2.0	3.0	2.0	5.6	2.0	10.0
Reactor Operations Engineer	1.0	1.5	1.0	2.8	1.0	5.0
Engineer	2.0	3.0	2.0	5.6	2.0	10.0
Reactor Operations Shift Supervisor	4.0	6.0	4.0	11.2	4.0	20.0
Senior Reactor Operator	8.0	12.0	8.0	22.4	8.0	40.0
Reactor Operator	16.0	24.0	16.0	44.8	16.0	80.0
Utility Operator	16.0	24.0	16.0	44.8	16.0	80.0
Technicians	16.0	24.0	20.0	56.0	24.0	120.0
Craft Supervisor	1.0	1.5	1.0	2.8	1.0	5.0
Crew Foreman	4.0	6.0	4.0	11.2	4.0	20.0
Craftsman[b]	8.0	12.0	12.0	33.6	12.0	60.0
Warehouseman	4.0	6.0	8.0	22.4	8.0	40.0
Tool Crib Attendant	4.0	6.0	8.0	22.4	8.0	40.0
Subtotals		132.0		291.2		540.0
Accident Cleanup Staff						
Cleanup Superintendent	1.0	1.5	1.0	2.8	1.0	5.0
Radioactive Shipment Specialist	1.0	1.5	1.0	2.8	1.0	5.0
Clerk	1.0	1.5	1.0	2.8	2.0	10.0
Shift Supervisor	4.0	6.0	4.0	11.2	4.0	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.

REFERENCES

1. Final Programmatic Environmental Impact Statement Related to 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.

2. 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.

3. 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.

4. 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.

5. Environmental Assessment - Use of Epicor II at Three Mile Island, Unit 2, NUREG-0591, U.S. Nuclear Regulatory Commission, Washington, D.C. October 1979.

6. 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.

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. U.S. Code of Federal Regulations, Title 40, Part 141, "National Interim Primary Drinking Water Regulations," (40 CFR 141), U.S. Government Printing Office, Washington, D.C., 1980.

9. G. A. Townes, Canister Design Considerations for Packaging of TMI Unit 2 Damaged Fuel and Debris, GEND-011, Allied-General Nuclear Services, Barnwell, South Carolina, September 1981.

APPENDIX F

DETAILS OF COSTS OF ACCIDENT CLEANUP AT A REFERENCE PWR

Details of the costs of accident cleanup at the reference PWR are presented in this appendix. Costs are based on the technical requirements, manpower needs, and cleanup schedules presented in Appendix E, and are given 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 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 presented in Appendix H. Unit cost information used as bases for these cost estimates is given in Appendix I. Some key assumptions, used as bases for these cost estimates, are presented in Section 4.2 of Volume 1.

Details of the costs of preparations for accident cleanup are presented in Section F.1; details of the costs of accident cleanup in the auxiliary and fuel buildings in Section F.2; and details of the costs of accident cleanup in the containment building in Section F.3.

A brief analysis of the sensitivity of cost estimates to various factors is presented in Section 11.6 of Volume 1. Factors that can affect the cost estimates include potential delays in accident cleanup, uncertainties in plant condition, alternative waste processing and waste disposal situations, and other factors.

F.1 DETAILS OF COSTS OF PREPARATIONS FOR ACCIDENT CLEANUP

The estimated costs of preparations for accident cleanup are presented in this section. These costs are summarized in Table F.1-1. Preparations for cleanup following the scenario 1 accident are estimated to require 1.5 years

TABLE F.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 demineralizer system and for fuel from defueling the reactor following the accident.

and to cost approximately $33.7 million. Preparations for cleanup following the scenario 2 accident are estimated to require 2.5 years and to cost approximately $67.2 million. Preparations for cleanup following the scenario 3 accident are estimated to require 3 years and to cost approximately $98.0 million.

Costs of preparations for accident cleanup include the costs of maintaining the reactor in a safe shutdown condition as well as the costs of completing the activities described in Section E.2 of Appendix E. Approximately 50% of these costs are utility staff labor costs. Contractor costs for providing the engineering support to prepare documentation for regulatory agencies, to prepare work plans and work schedules, and to design the special tools and equipment needed for cleanup contribute an additional 10 to 20% to the total preparations costs. The costs of the energy (electricity) needed to maintain the plant in a safe cold-shutdown condition contribute about 20% to the total preparations costs. The total costs of planning and preparation are expected to vary approximately linearly with the time required to complete these activities following a particular accident.

During preparations for accident cleanup, the spent fuel that has accumulated in the spent fuel pool is postulated to be shipped offsite to provide space in the pool for the demineralizer system used to process contaminated water as well as space for the fuel removed from the reactor during the defueling operation that takes place during accident cleanup. This accumulated fuel is assumed to be transported to an ISFSI for interim storage. The costs of transportation and 10-year storage of this fuel are estimated to total about $13.6 million. This cost is assumed to be an operating cost rather than an accident cleanup cost (since the accumulated fuel would eventually have been shipped offsite to make space available in the spent fuel pool for later refueling operations). The cost of removal of the fuel is shown as a line item at the bottom of Table F.1-1 for completeness.

Details of the costs of preparations for accident cleanup are presented in the following subsections.

F.1.1 Costs of Utility Staff Labor

The annual costs of utility staff labor during preparations for accident cleanup are shown in detail in Table F.1-2. These costs are based on the utility staff labor requirements described in Section E.2 of Appendix E. Costs are included for site support and plant operations staff as well as for cleanup planning staff. The annual costs of utility staff labor are multiplied by the number of years assumed for completion of the planning and preparations phase to obtain the staff labor costs shown in Table F.1-1. Total staff labor costs for preparations for accident cleanup are estimated to be about $12.8 million following the scenario 1 accident, about $24.4 million following the scenario 2 accident, and about $35.0 million following the scenario 3 accident.

Contractor labor costs to provide engineering support during preparations for accident cleanup are not shown in Table F.1-2. These costs are included with specialty contractor costs discussed in Section F.1.6.

F.1.2 Cost of Waste Management

The cost of waste management is expected to be small during preparations for accident cleanup. Wastes generated during this period consist mostly of compactible and combustible solids (disposable clothing, rags, plastic covers, laydown pads, and miscellaneous trash) as well as some filters and ion exchange materials. The generation rate for these wastes during preparations for accident cleanup is expected to be similar to the generation rate during normal reactor operations.

During preparations for cleanup following the scenario 1 accident, some fuel racks are postulated to be removed from the spent fuel pool to provide space for the demineralizer system used to process accident water. All of the fuel racks are removed from the spent fuel pool during preparations for cleanup following the scenario 2 and scenario 3 accidents to provide space for the demineralizer system and for new fuel racks that can accommodate canistered fuel. The costs of packaging, transportation, and disposal of the old fuel racks at a shallow-land burial ground are given in Table F.1-3. These costs are included as part of the waste management costs for preparations for accident cleanup.

TABLE F.1-2. Estimated Annual Costs of Utility Staff Labor for Preparations for Accident Cleanup at the Reference PWR

Staff Position	Annual Cost per Person[a] ($ thousands)	Scenario 1 Accident: Staff Labor[b] (man years/year)	Scenario 1 Accident: Labor Cost per Year[c] ($ thousands)	Scenario 2 Accident: Staff Labor[b] (man years/year)	Scenario 2 Accident: Labor Cost per Year[c] ($ thousands)	Scenario 3 Accident: Staff Labor[b] (man years/year)	Scenario 3 Accident: Labor Cost per Year[c] ($ thousands)
Plant Superintendent	89.4	1.0	89.4	1.0	89.4	1.0	89.4
Assistant Plant Superintendent	76.2	1.0	76.2	1.0	76.2	1.0	76.2
Consultants	100.0	3.0	300.0	6.0	600.0	10.0	1 000.0
Secretaries and Word Processors	24.4	8.0	195.2	10.0	244.0	12.0	292.8
Site Support Staff							
Health and Safety Supervisor	60.5	1.0	60.5	1.0	60.5	1.0	60.5
Health Physicist	47.3	1.0	47.3	1.0	47.3	1.0	47.3
Senior Health Physics Technician	39.5	8.0	316.0	8.0	316.0	12.0	474.0
Health Physics Technician	30.1	16.0	481.6	16.0	481.6	24.0	722.4
Protective Equipment Attendant	27.8	4.0	111.2	8.0	222.4	8.0	222.4
Industrial Safety Specialist	52.6	1.0	52.6	1.0	52.6	1.0	52.6
Industrial Safety Technician	30.1	2.0	60.2	2.0	60.2	2.0	60.2
Security Supervisor	55.9	1.0	55.9	1.0	55.9	1.0	55.9
Security Shift Supervisor	36.8	4.0	147.2	4.0	147.2	4.0	147.2
Security Patrolman	25.6	48.0	1 228.8	48.0	1 228.8	48.0	1 228.8
Contracts and Accounting Supervisor	47.1	1.0	47.1	1.0	47.1	1.0	47.1
Accountant	39.3	1.0	39.3	1.0	39.3	2.0	78.6
Contracts Specialist	39.3	1.0	39.3	1.0	39.3	1.0	39.3
Insurance Specialist	39.3	1.0	39.3	1.0	39.3	1.0	39.3
Procurement Specialist	39.3	1.0	39.3	1.0	39.3	1.0	39.3
Clerk	24.4	2.0	48.8	4.0	97.6	4.0	97.6
Quality Assurance Supervisor	52.6	1.0	52.6	1.0	52.6	1.0	52.6
Quality Assurance Engineer	47.3	1.0	47.3	2.0	94.6	2.0	94.6
Quality Assurance Technician	27.8	1.0	27.8	2.0	55.6	2.0	55.6
Subtotals		96.0	2 942.1	104.0	3 177.2	117.0	3 615.3
Plant Operations Staff							
Plant Operations Supervisor	61.2	1.0	61.2	1.0	61.2	1.0	61.2
Plant Chemist	52.4	1.0	52.4	1.0	52.4	1.0	52.4
Chemist	46.9	2.0	93.8	2.0	93.8	2.0	93.8
Reactor Operations Engineer	52.4	1.0	52.4	1.0	52.4	1.0	52.4
Engineer	46.9	2.0	93.8	2.0	93.8	2.0	93.8
Reactor Operations Shift Supervisor	52.4	4.0	209.6	4.0	209.6	4.0	209.6
Senior Reactor Operator	46.9	8.0	375.2	8.0	375.2	8.0	375.2
Reactor Operator	34.8	16.0	556.8	16.0	556.8	16.0	556.8
Utility Operator	32.5	16.0	520.0	16.0	520.0	16.0	520.0
Technician	30.9	12.0	370.8	12.0	370.8	12.0	370.8
Craft Supervisor	47.3	1.0	47.3	1.0	47.3	1.0	47.3
Crew Foreman	44.8	4.0	179.2	4.0	179.2	4.0	179.2
Craftsman	32.5	32.0	1 040.0	32.0	1 040.0	32.0	1 040.0
Warehouseman	27.8	4.0	111.2	4.0	111.2	4.0	111.2
Tool Crib Attendant	27.8	4.0	111.2	4.0	111.2	4.0	111.2
Subtotals		108.0	3 874.9	108.0	3 874.9	108.0	3 874.9
Cleanup Planning Staff							
Cleanup Planning Supervisor	61.2	1.0	61.2	1.0	61.2	1.0	61.2
Engineering Supervisor	52.4	1.0	52.4	1.0	52.4	2.0	104.8
Engineer	46.9	6.0	281.4	8.0	375.2	12.0	562.8
Estimator	46.9	1.0	46.9	2.0	93.8	4.0	187.6
Draftsman	30.0	2.0	60.0	4.0	120.0	6.0	180.0
Crew Leader	44.8	1.0	44.8	2.0	89.6	2.0	89.6
Utility Operator	32.5	4.0	130.0	8.0	260.0	16.0	520.0
Craftsman	32.5	8.0	260.0	12.0	390.0	16.0	520.0
Laborer	31.1	4.0	124.4	8.0	248.8	16.0	497.6
Subtotals		28.0	1 061.1	46.0	1 691.0	75.0	2 723.6
Totals		245.0	8 538.9	276.0	9 752.7	324.0	11 672.2

(a) From Table I.1-1 of Appendix I.
(b) From Table E.2-2 of Appendix E.
(c) Number of figures shown is for computational accuracy and does not imply precision to the nearest hundred dollars.

TABLE F.1-3. Waste Management Costs for the Disposal of Spent Fuel Racks

Item	Value
Burial Volume (m^3)	390
Estimated Radioactivity Content (Ci)	3.5
Type of Disposable Container	Plywood Box
Number of Disposable Containers	14
Number of Waste Shipments	5
Disposable Container Cost ($)	33 600
Transportation Cost ($)	11 810
Shallow-Land Burial Costs	
Disposal Charge ($)	119 810
State Surcharge ($)	4 130
Handling Surcharge ($)	2 080
Total Waste Management Costs ($)	171 [illegible]0

Any post-accident cleanup required in the auxiliary and fuel buildings during preparations for accident cleanup can significantly increase the amount of radioactive waste generated and the costs of waste management. The costs of waste management for cleanup in the auxiliary and fuel buildings are discussed in Section F.2.2.

F.1.3 Cost of Energy

Electricity is the primary energy cost item associated with providing essential systems and services that must remain in place to keep the damaged reactor in a safe shutdown condition and to provide necessary support services during accident cleanup operations. 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 during preparations for accident cleanup. Energy costs during preparations for accident cleanup are estimated to be about $7.2 million following the scenario 1 accident, about $12.0 million following the scenario 2 accident, and about $14.4 million following the scenario 3 accident.

F.1.4 Costs of Special Equipment and Facilities

Special equipment and facility items that are needed for accident cleanup in the containment building include:

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

These items are postulated to be designed, fabricated, and installed during preparations for cleanup, and their costs are shown in Table F.1-1. Costs include contractor labor, material, and overhead costs for design and construction.

The demineralizer system is described in Section E.4.1.1 of Appendix E. The cost of this sytem is estimated to be about $1 million based on the costs of system components[(2)] and an allowance for engineering costs.

Fuel racks for canistered fuel are installed in the spent fuel pool prior to defueling the reactor following the scenario 2 and scenario 3 accidents. The cost of these fuel racks is estimated to be $310,000 based on published costs[(3)] of fuel storage racks at a PWR power station.

One 1,000-m^3 steel tank for interim storage of processed water is needed following the scenario 1 accident; two tanks are needed following the scenario 2 accident; and three tanks are needed following the scenario 3 accident. The installed cost per tank, including concrete foundation, piping, and controls is estimated to be about $135,000 based on published construction cost data.[(2)]

Facilities for the interim onsite storage of radioactive wastes include a warehouse for storage of drums and boxes of low-activity wastes and a shielded storage facility for interim storage of high-activity wastes (filters, ion exchange materials, and evaporator bottoms). These storage facilities are described in Section E.2.1.4 of Appendix E. Construction costs are estimated based on the sizes of the required facilities and on published construction cost data.[(2)]

A reactor vessel mockup is used to test equipment for the removal of damaged fuel and to train cleanup personnel in the use of this equipment. Defueling following the scenario 3 accident is postulated to require the use of a more elaborate mockup than is needed for defueling following the scenario 2 accident. Cost estimates of the mockup are based on the reported costs of a facility for training nuclear plant personnel in refueling and maintenance operations.(4)

F.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.

F.1.6 Costs of Specialty Contractors

Major specialty contractor costs include the costs of engineering support, environmental monitoring, and laundry costs for protective clothing. Engineering support costs are estimated by postulating a support staff size and using a charge-out rate of $100,000 per man-year. Engineering support costs are estimated to be about $3 million following the scenario 1 accident, about $10 million following the scenario 2 accident, and about $18 million following the scenario 3 accident.

Environmental monitoring costs are based on the environmental sampling program described in Section D.8 of Appendix D. The costs of sample analyses are taken from Reference 5. Laundry costs are estimated on the basis of a cost of $5 per set of clothing laundered.

F.1.7 Costs of Nuclear Insurance and License Fees

Estimated costs of nuclear liability insurance and of license fees required during preparations for accident cleanup are shown in Table F.1-4. These costs are estimated to total about $2.2 million following the scenario 1 accident, about $3.7 million following the scenario 2 accident, and about $4.5 million following the scenario 3 accident.

TABLE F.1-4. Estimated Costs of Nuclear Liability Insurance and License Fees During Preparations for Accident Cleanup

Category	Unit Cost ($)	Total Cost ($)[a] Preparations for Cleanup Following Scenario 1 Accident	Preparations for Cleanup Following Scenario 2 Accident	Preparations for Cleanup Following Scenario 3 Accident
Property Damage Insurance	1 000 000/yr	1 500 000	2 500 000	3 000 000
Nuclear Liability Insurance	400 000/yr	600 000	1 000 000	1 200 000
License Fees[b]--Facility License Amendment (Class V)	25 800	25 800	25 800	25 800
Routine Health, Safety, and Environmental Inspections	75 700/yr	113 550	189 200	227 100
Routine Safeguards Inspections	11 800/yr	17 700	29 500	35 400
Total Costs		2 257 050	3 744 500	4 488 300

(a) Costs are in early-1981 dollars.
(b) From 10 CFR 170.

F.1.8 Costs of Disposal of Accumulated Spent Fuel

The costs of disposal of the accumulated spent fuel that is removed from the fuel storage basin during preparations for cleanup are shown in Table F.1-5. These costs include cask rental charges, transportation charges, and 10-year storage charges. Transport of the fuel is assumed to be by rail

TABLE F.1-5. Estimated Cost of Disposal of Accumulated Spent Fuel During Preparations for Accident Cleanup[a,b]

Item	Value
Number of Fuel Assemblies[c]	258
Number of Shipments[d]	37
Cask Rental Charges[e] ($)	2 220 000
Transportation Costs[f] ($)	675 250
ISFSI Costs[g] ($)	10 707 000
Total Cost of Disposal ($)	13 602 250

(a) Numbers of significant figures shown are for computational accuracy only.
(b) Costs are in early 1981-dollars.
(c) Assumes 1 1/3 fuel cores.
(d) Shipment is by rail in IF-300 casks.
(e) Based on information from Table I.2-2 of Appendix I. Assumes 15 days for a round trip rail shipment.
(f) Based on information in Section I.3.2 of Appendix I.
(g) Based on information in Section I.4.2 of Appendix I. Costs are given for 10-year storage.

in IF-300 casks to an ISFSI located 1600 km from the reactor site. These costs are shown for informational purposes only since disposal of the accumulated spent fuel is assumed to be an operational cost.

F.2 DETAILS OF COSTS OF ACCIDENT CLEANUP IN THE AUXILIARY AND FUEL BUILDINGS

The estimated costs of accident cleanup in the auxiliary and fuel buildings following the scenario 2 and scenario 3 accidents are presented in this section. (Accident cleanup in the auxiliary and fuel buildings is not postulated following the scenario 1 accident.) These costs are summarized in Table F.2-1. Accident cleanup in the auxiliary and fuel buildings is

TABLE F.2-1. Summary of Estimated Costs of Accident Cleanup in the Auxiliary and Fuel Buildings at the Reference PWR[a]

Cost Category	Estimated Costs[b] ($ millions)		of Total
Cleanup Worker Labor		11.252	72.2
Waste Management		1.292	8.3
Special Tools & 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 accident cleanup in the containment building. Management and support staff costs and incidental costs are included in the costs of preparations for accident cleanup (see Table F.1-1).
(b) Costs are in early-1981 dollars. Number of significant figures is for computational accuracy only.

estimated to require 2.2 years, to cost approximately $19.5 million, and is postulated to take place during preparations for cleanup in the containment building.

Costs shown in Table F.2-1 include cleanup worker labor costs, waste management costs, costs of equipment and supplies, and specialty contractor costs specifically related to 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 such as energy costs, environmental surveillance costs, and insurance costs are included with the costs of preparations for cleanup shown in Table F.1-1.

Details of the costs of accident cleanup in the auxiliary and fuel buildings following the scenario 2 and scenario 3 accidents are presented in the following subsections.

F.2.1 Costs of Cleanup Worker Labor

Estimated cleanup worker labor costs for accident cleanup in the auxiliary and fuel buildings are shown in Table F.2-2 to be about $11.2 million. These costs are based on the cleanup worker requirements described in Section E.3 of Appendix E.

TABLE F.2-2. Estimated Cleanup Worker Labor Costs for Accident Cleanup in the Auxiliary and Fuel Buildings

Worker Category	Annual Cost per Person[a] ($ thousands)	Worker Requirement[b] (man-years)	Labor Cost[c] ($ thousands)
Cleanup Operations Supervisor	61.2	2.2	134.6
Crew Leader	44.8	42.5	1 904.0
Utility Operator	32.5	65.0	2 112.5
Laborer	31.1	65.8	2 046.4
Craftsman	32.5	115.7	3 760.2
Health Physics Technician	30.1	43.0	1 294.3
Totals		334.2	11 252.0

(a) From Table I.1-1 of Appendix I.
(b) From Table E.3-2 of Appendix E.
(c) Number of figures shown is for computational accuracy and does not imply precision to the nearest hundred dollars.

Contractor labor costs to provide engineering support during accident cleanup in the auxiliary and fuel buildings are not shown in Table F.2-2. These costs are included with specialty contractor costs (see Section F.2.5).

F.2.2 Costs of Waste Management

Based on the waste management disposal assumptions discussed in Appendix E, estimated costs of radioactive waste management for accident cleanup in the auxiliary and fuel buildings are shown in detail in Table F.2-3. These costs are based on the waste management requirements given in Table E.3-1 of Appendix E. Except for the high-activity wastes (filter cartridges and ion exchange materials) from processing contaminated water, all wastes from accident cleanup in the auxiliary and fuel buildings are assumed to be transported by truck to a shallow-land burial ground located 1600 km from the reactor site. The high-activity wastes from the processing of CVCS liquids are placed in interim storage at the site and are ultimately transported in shielded containers to a federal repository. The federal repository is also assumed to be located 1600 km from the reactor site.

F.2.3 Costs of Special Tools and Equipment

The estimated costs of the special tools and equipment anticipated to be used during accident cleanup in the auxiliary and fuel buildings are shown in Table F.2-4 to be about $0.3 million.

F.2.4 Costs of Miscellaneous Supplies

Expendable supplies include decontamination chemicals, ion exchange resins, glass fiber and HEPA filters, cartridge type filters, disposable protective clothing, assorted cleaning agents, rags, mops, plastic bags and sheeting, and expendable tools. The estimated costs of these miscellaneous supplies are shown in Table F.2-5 to be about $1.4 million.

F.2.5 Costs of Specialty Contractors

Major specialty contractor costs include the costs of engineering support and laundry services. These costs are estimated to be about $1.3 million for cleanup in the auxiliary and fuel buildings following the scenario 3 accident.

TABLE F.2-3. Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Auxiliary and Fuel Buildings at the Reference PWR[a]

Waste Category	Burial Volume (m^3)	Estimated Radioactivity Content (Ci)	Disposable Container Requirements: Type	Disposable Container Requirements: Number	Number of Shipments	Packaging and Transport Costs: Disposable Container Cost[b] ($)	Packaging and Transport Costs: Cask Rental Charge[c] ($)	Packaging and Transport Costs: Transport Costs[d,e] ($)	Disposal Site Costs[e,f] – Shallow Land Burial Ground Costs ($): Disposal Charge	Shallow Land Burial Ground Costs ($): State Surcharge	Shallow Land Burial Ground Costs ($): Handling Surcharge	Shallow Land Burial Ground Costs ($): Liner Surcharge	Shallow Land Burial Ground Costs ($): Curie Surcharge	Federal Repository Costs ($)	Total Waste Management Costs ($)
Sludge	3	42	0.21-m^3 steel drum	14	1	420	2 700	4 110	1 380	30	1 220				9 860
Process Solids															
Filter Cartridges	0.3	20	0.3-m^3 steel liner	1	0.5		1 350	2 055						2 500	5 905
Zeolite Liners	0.9	20 000	0.3-m^3 steel liner	3	2		4 050	8 210						7 500	19 760
Organic Resin Liners	0.3	100	0.3-m^3 steel liner	1	0.5		1 350	2 055						2 500	5 905
Chemical Decontamination Solutions	375	100	0.21-m^3 steel drum	1 875	16	56 250		37 790	115 200	3 980	11 720				224 940
Trash															
Compactible, Combustible	101	168	0.21-m^3 steel drum	480	4	14 400		9 450	33 880	1 070	2 930				61 730
Compactible, Noncombustible	326	54	0.21-m^3 steel drum	1 550	13	46 500		30 710	100 150	3 460	9 520				190 340
Noncompactible	822	25	3.5-m^3 plywood box	235	39	94 000		92 120	252 520	8 710	16 210				463 560
Contaminated Equipment															
LSA Materials	185	14	3.5-m^3 plywood box	53	9	21 200		21 260	56 830	1 960	3 740				104 990
High-Activity Materials	57	100	2.85-m^3 steel liner	20	20	40 000	36 000	82 100	17 510	600	20 120	8 220	–		196 330
Subtotals															
Wastes Sent to Shallow-Land Burial	1 869	503		4 227	102	272 770	38 700	277 540	577 470	19 810	65 460	8 200	0		1 259 970
Wastes Sent to Federal Repository Repository	1.5	20 120		5	3		6 750	12 320					–	12 500	31 570
Totals	1 871	20 623		4 232	105	272 770	45 450	289 860	577 470	19 810	65 460	8 220	0	12 500	1 291 540

(a) Numbers of significant figures shown are for computational accuracy only.
(b) Based on information from Table I.2-1 of Appendix I.
(c) Based on information from Table I.2-2 of Appendix I. Assumes 6 days for round-trip truck shipment.
(d) Based on information from Table I.3-4 of Appendix I. Includes overweight charges and second driver costs where applicable.
(e) Charges are computed on the assumption that all shipments for a given waste category are identical. Actually, charges for individual shipments will vary depending on the specific physical and radiological characteristics of individual shipments. The averaging technique used is believed to be appropriate for computing total charges.
(f) Based on information from Table I.4-1 of Appendix I.

TABLE F.2-4 Estimated Costs of Special Tools and Equipment for Accident Cleanup in the Auxiliary and Fuel Buildings

Tool or Equipment Item	Estimated Unit Cost[a] ($ thousands)	Number Required	Total Cost ($ thousands)
Portable plasma arc torch	20	1	20
Portable oxyacetylene torch	1	2	2
Guillotine pipe saw	4	2	8
Reciprocating hack saw	1	2	2
Underwater lights and viewing aids		AR[b]	5
Underwater tools (e.g., wrenches, tongs)		AR[b]	15
Submersible pump with disposable filters	2	1	2
High-pressure water jet	20	2	40
Scaffolding and safety nets		AR	10
Power-operated mobile man lift	40	1	40
9100-kg mobile hydraulic crane	28	1	28
9100-kg forklift	28	2	56
Rigging materials (e.g., chokers, grapplers, winchers)		AR	5
Vacuum cleaner (HEPA filtered)	4	2	8
Portable ventilation enclosure	10	4	40
Supplied-air plastic suit	0.2	20	4
Waste compactor[c]			
Incinerator[c]			
Total Cost			285

(a) From Table I.5-1 of Appendix I. Costs are in early-1981 dollars.
(b) AR: as required.
(c) Cost of this item is included in equipment costs for accident cleanup in the containment building.

TABLE F.2-5. Estimated Costs of Miscellaneous Supplies for Accident Cleanup in the Auxiliary and Fuel Buildings

Item	Quantity	Total Cost[a] ($ thousands)
Anticontamination clothing	15 500 sets[b]	775
Cleaning supplies	see note (c)	300
Expendable tools	see note (d)	200
Ion exchange resins	10 m^3	50
Glass-fiber and HEPA filters	Unspecified	50
Mechanical supplies and hardware	Unspecified	20
Electrical components and cables	Unspecified	10
Respirator face pieces	50 ea	5
Ion-exchange and filter liners	5 ea	25
Total cost		1 435

(a) Costs are in early 1981 dollars and are rounded to the nearest $1000.
(b) Estimated at two clothing changes per shift per cleanup worker. One set of clothing can be laundered and used four times.
(c) Estimated at $150,000 per year.
(d) Estimated at $100,000 per year.

F.3 DETAILS OF COSTS OF ACCIDENT CLEANUP IN THE CONTAINMENT BUILDING

The estimated costs of accident cleanup in the containment building of the reference PWR are presented in this section, and are summarized in Table F.3-1. Accident cleanup in the containment building is estimated to require 1.5 years and to cost approximately $71.5 million following the scenario 1 accident. It is estimated to require 2.8 years and to cost approximately $137.2 million following the scenario 2 accident. It is estimated to require 5.0 years and to cost approximately $287.0 million following the scenario 3 accident.

TABLE F.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: Estimated Costs[a] ($ millions)		Percent of Total	Accident Cleanup Following Scenario 2 Accident: Estimated Costs[a] ($ millions)		Percent of Total	Accident Cleanup Following Scenario 3 Accident: 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.

Labor costs and waste management costs are the major cost items for accident cleanup in the containment building. Staff labor costs account for about 30 to 55% of the total cost of accident cleanup, depending on accident scenario. Contractor costs for engineering support contribute an additional 3 to 6% to the total cost of accident cleanup. Waste management costs account for about 15 to 45% of accident cleanup costs. The major waste management cost is the cost of disposal of the nuclear fuel from defueling the reactor following an accident. Energy costs account for about 12% of the total cost of accident cleanup in the containment building of the reference PWR.

Details of the costs of accident cleanup in the containment building are presented in the following subsections.

F.3.1 Costs of Utility Staff Labor

Estimated utility staff labor costs for accident cleanup in the containment building are shown in Table F.3-2. These costs are based on the utility staff labor requirements described in Section E.4 of Appendix E and specifically on the tasks outlined in Figures E.4-5, E.4-6, and E.4-7. Costs are included for site support and plant operations staff as well as for the staff actually involved in cleanup operations inside the containment building.

For the scenario 1 accident, the accident cleanup staff accounts for only about 34% of total staff labor costs, with the site support and plant operations staffs accounting for the remainder of staff labor costs. For the scenario 2 accident, the accident cleanup staff accounts for about 49% of total staff labor costs, and for the scenario 3 accident, the accident cleanup staff accounts for over 60% of these costs. The increase in the fraction of the staff labor cost attributed to cleanup staff labor with increasing accident severity is due primarily to 1) the increase in the labor requirement for defueling the reactor and 2) the additional cleanup manpower required to comply with occupational dose limitations.

An additional staff labor cost not shown in Table F.3-2 but included in the total labor costs shown in Table F.3-1 is the living allowance paid to crew leaders and utility operators brought from other plants to assist in reactor defueling operations. As explained in Section E.4.2 of Appendix E, a

TABLE F.3-2. Estimated Costs of Utility Staff Labor for Accident Cleanup in the Containment Building at the Reference PWR

Staff Position	Annual Cost per Person[a] ($ thousands)	Scenario 1 Accident		Scenario 2 Accident		Scenario 3 Accident	
		Labor Requirement[b] (man-years)	Labor Cost[c] ($ thousands)	Labor Requirement[b] (man-years)	Labor Cost[c] ($ thousands)	Labor Requirement[b] (man-years)	Labor Cost[c] ($ thousands)
Plant Superintendent	89.4	1.5	134.1	2.8	250.3	5.0	447.0
Assistant Plant Superintendent	76.2	1.5	114.3	2.8	213.4	5.0	381.0
Consultants	100.0	4.5	450.0	16.8	1 680.0	50.0	5 000.0
Secretaries and Word Processors	24.4	12.0	292.8	28.0	683.2	60.0	1 464.0
Site Support Staff							
Health and Safety Supervisor	60.5	1.5	90.8	2.8	169.4	5.0	302.5
Health Physicist	47.3	1.5	71.0	2.8	132.4	5.0	236.5
Senior Health Physics Technician	39.5	12.0	474.0	22.4	884.8	60.0	2 370.0
Health Physics Technician	30.1	12.0	361.2	22.4	674.2	60.0	1 806.0
Protective Equipment Attendant	27.8	6.0	166.8	22.4	622.7	60.0	1 668.0
Industrial Safety Specialist	52.6	1.5	78.9	2.8	147.3	5.0	263.0
Industrial Safety Technician	30.1	3.0	90.3	5.6	168.6	10.0	301.0
Security Supervisor	55.9	1.5	83.9	2.8	156.5	5.0	279.5
Security Shift Supervisor	36.8	6.0	220.8	11.2	412.2	20.0	736.0
Security Patrolman	25.6	72.0	1 843.2	134.4	3 440.6	240.0	6 144.0
Contracts and Accounting Supervisor	47.1	1.5	70.7	2.8	131.9	5.0	235.5
Accountant	39.3	1.5	59.0	2.8	110.0	10.0	393.0
Contracts Specialist	39.3	1.5	59.0	2.8	110.0	5.0	196.5
Insurance Specialist	39.3	1.5	59.0	2.8	110.0	10.0	393.0
Procurement Specialist	39.3	1.5	59.0	2.8	110.0	5.0	196.5
Clerk	24.4	3.0	73.2	11.2	273.3	30.0	732.0
Quality Assurance Supervisor	52.6	1.5	78.9	2.8	147.3	5.0	263.0
Quality Assurance Engineer	47.3	3.0	141.9	5.6	264.9	10.0	473.0
Quality Assurance Technician	27.8	3.0	83.4	5.6	155.7	10.0	278.0
Construction Engineering Supervisor	61.2	1.5	91.8	2.8	171.4	5.0	306.0
Engineer	52.4	9.0	471.6	22.4	1 173.8	60.0	3 144.0
Estimator	46.9	1.5	70.4	5.6	262.6	20.0	938.0
Draftsman	30.0	3.0	90.0	11.2	336.0	30.0	900.0
Subtotals		150.0	4 888.8	310.8	10 165.6	675.0	22 555.0
Plant Operations Staff							
Plant Operations Supervisor	61.2	1.5	91.8	2.8	171.4	5.0	306.0
Plant Chemist	52.4	1.5	78.6	2.8	146.7	5.0	262.0
Chemist	46.9	3.0	140.7	5.6	262.6	10.0	469.0
Reactor Operations Engineer	52.4	1.5	78.6	2.8	146.7	5.0	262.0
Engineer	46.9	3.0	140.7	5.6	262.6	10.0	469.0
Reactor Operations Shift Supervisor	52.4	6.0	314.4	11.2	586.9	20.0	1 048.0
Senior Reactor Operator	46.9	12.0	562.8	22.4	1 050.6	40.0	1 876.0
Reactor Operator	34.8	24.0	835.2	44.8	1 559.0	80.0	2 784.0
Utility Operator	32.5	24.0	780.0	44.8	1 456.0	80.0	2 600.0
Technician	30.9	24.0	741.6	56.0	1 730.4	120.0	3 708.0
Craft Supervisor	47.3	1.5	71.0	2.8	132.4	5.0	236.5
Crew Foreman	44.8	6.0	268.8	11.2	501.8	20.0	896.0
Craftsman	32.5	12.0	390.0	33.6	1 092.0	60.0	1 950.0
Warehouseman	27.8	6.0	166.8	22.4	622.7	40.0	1 112.0
Tool Crib Attendant	27.8	6.0	166.8	22.4	622.7	40.0	1 112.0
Subtotals		132.0	4 827.8	291.2	10 344.5	540.0	19 090.5
Accident Cleanup Staff							
Cleanup Superintendent	61.2	1.5	91.8	2.8	171.4	5.0	306.0
Radioactive Shipment Specialist	39.3	1.5	59.0	2.8	110.0	5.0	196.5
Clerk	24.4	1.5	36.6	2.8	68.3	10.0	244.0
Shift Supervisor	52.4	6.0	314.4	11.2	586.9	20.0	1 048.0
Crew Leader	44.8	15.2	681.0	62.4	2 795.5	195.5	8 758.4
Utility Operator	32.5	50.3	1 634.8	203.6	6 617.0	605.2	19 669.0
Laborer	31.1	29.4	914.3	111.0	3 452.1	321.0	9 983.1
Craftsman	32.5	24.5	796.2	140.3	4 559.8	669.4	21 755.5
Health Physics Technician	30.1	18.5	556.8	78.2	2 353.8	247.6	7 452.8
Subtotals		148.4	5 084.9	615.1	20 714.8	2 078.7	69 413.3
Totals		449.9	15 792.7	1 267.5	44 051.8	3 413.7	118 350.8

(a) From Table I.1-1 of Appendix I.
(b) From Table E.4-10 of Appendix E.
(c) Number of Figures shown is for computational accuracy and does not imply precision to the nearest hundred dollars.

large number of trained personnel not normally available at the accident-damaged reactor station are needed for defueling the reactor. Additional personnel are therefore assumed to be temporarily assigned from other reactor stations to assist in the defueling. Personnel on temporary assignment are assumed to be paid a living allowance of $2000 per month in addition to their regular salaries.

Contractor labor costs to provide engineering support during accident cleanup in the containment building are not shown in Table F.3-2. These costs are included with specialty contractor costs discussed in Section F.3.6.

F.3.2 <u>Costs of Waste Management</u>

Based on the waste management disposal assumptions discussed in Appendix E, the estimated costs of radioactive waste management for accident cleanup in the containment building are shown in detail in Tables F.3-3, F.3-4, and F.3-5 for the three accident scenarios. These costs are based on the waste management requirements given in Table E.4-2 of Appendix E and include container costs, transportation, and disposal costs. Labor costs for packaging the wastes prior to shipment are included in the utility staff labor costs shown in Table F.3-2. Labor costs for transportation and disposal are included in the total costs for these activities shown in Tables F.3-3, F.3-4, and F.3-5, since transportation and disposal are contracted operations.

High-activity wastes (filter cartridges, ion exchange resin liners, and evaporator bottoms from processing radioactive liquids) and damaged fuel assemblies are postulated in this study to be transported to a federal repository. Fuel assemblies that are not damaged 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.

Most of the costs of waste management are for packaging, transportation, and disposal of radioactive wastes shipped to a federal repository. For example, for the scenario 3 accident, although only 6% of the total volume of waste is sent to a federal repository, this waste accounts for about 82% of

TABLE F.3-3. Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Containment Building at the Reference PWR Following the Scenario 1 Accident[a,b]

Waste Category	Burial Volume (m^3)	Estimated Radioactivity Content (Ci)	Disposable Container Requirements: Type	Disposable Container Requirements: Number	Number of Shipments	Packaging and Transport Costs: Disposable Container Cost[c] ($)	Packaging and Transport Costs: Cask Rental Charge[d] ($)	Packaging and Transport Costs: Transport Costs[e,f] ($)	Disposal Site Costs[f,g]: Shallow-Land Burial Ground Costs ($): Disposal Charge	Shallow-Land Burial Ground Costs ($): State Surcharge	Shallow-Land Burial Ground Costs ($): Handling Surcharge	Shallow-Land Burial Ground Costs ($): Liner Surcharge	Shallow-Land Burial Ground Costs ($): Curie Surcharge	Federal Repository Costs ($)	Total Waste Management Costs ($)
Sludge	2	20	0.21-m^3 steel drum	8	1	240	1 350	4 110	750	20	550				7 020
Process Solids															
Filter Cartridges	1	30	0.3-m^3 steel liner	3	1.5		4 050	6 160						7 500	17 710
Zeolite Liners	2	30 500	0.3-m^3 steel liner	7	4		9 450	16 420						17 500	43 370
Organic Resin Liners	1	151	0.3-m^3 steel liner	3	1.5		4 050	6 160						7 500	17 710
Process Solids															
Filter Cartridges	1	20	0.21-m^3 steel drum	1	0	30			700	10					740
Evaporator Bottoms	46	20 000	2.85-m^3 steel liner	16	16	240 000	28 800	65 680						160 000	494 480
Organic Resins	4	20	0.21-m^3 steel drum	17	2	510	4 050	8 210	1 340	40	2 440				16 590
Chemical Decontamination Solutions	79	5	0.21-m^3 steel drum	375	3	11 250		7 090	24 270	840	2 200				45 650
Trash															
Compactible, Combustible	56	93	0.21-m^3 steel drum	265	2	7 950		4 720	18 780	590	1 460				33 500
Compactible, Noncombustible	1 180	30	0.21-m^3 steel drum	855	8	25 650		18 900	55 300	1 910	5 860				107 620
Noncompactible	455	14	3.5-m^3 plywood box	130	22	52 000		51 960	139 780	4 820	9 150				257 710
Contaminated Equipment															
LSA Materials	62	2	plywood box	8	3	2 600		7 090	19 050	660	1 640				31 040
High-Activity Materials	12	40	2.85-m^3 steel liner	4	4	8 000	7 200	16 420	3 690	130	4 020	2 380			41 840
Irradiated Hardware															
LSA Materials	28	8	3.5-m^3 plywood box	8	4	3 200	7 200	16 420	9 390	300	4 300				40 810
High-Activity Materials	52	26 000	2.85-m^3 steel liner	18	18	36 000	32 400	73 890	15 970	550	18 110	90 000	14 220		281 140
Fuel Assemblies															
Intact Assemblies	44		bare assembly	173	25		1 500 000	456 250						18 684 000	20 640 250
Damaged Assemblies	6		0.3-m^3 steel canister	20	5	120 000	300 000	91 250						2 160 000	2 671 250
Fuel Core Debris	—	—		—	—	—	—	—	—	—	—	—	—	—	—
Subtotals															
Waste Sent to Shallow Land Burial	931	26 252		1 689	67	147 430	52 200	208 810	289 020	9 870	49 730	92 380	14 220		863 660
Wastes Sent to Federal Repository	50	50 682		29	23	240 000	46 350	94 420						192 500	573 270
Reactor Fuel and Fuel Core Debris	50			193	30	120 000	1 800 000	547 500						20 844 000	23 311 500
Totals	1 031			1 911	120	507 430	1 898 550	850 730	289 020	9 870	49 730	92 380	14 220	21 036 500	24 748 430

(a) Numbers of significant figures shown are for computational accuracy only.
(b) Packaging and disposal requirements for radioactive wastes from accident cleanup in the containment building are given in Table E.4-2 of Appendix E.
(c) Based on information from Table I.2-1 of Appendix I.
(d) Based on information from Table I.2-2 of Appendix I. Assumes 6 days for round-trip truck shipment; 15 days for round-trip rail shipment.
(e) Based on information from Table I.3-4 of Appendix I. Includes overweight charges and second driver costs where applicable.
(f) Charges are computed on the assumption that all shipments for a given waste category are identical. Actually, charges for individual shipments would vary depending on the specific physical and radiological characteristics of individual shipments. The averaging technique used is believed to be appropriate for computing total charges.
(g) Based on information from Table I.4-1 of Appendix I.

TABLE F.3-4. Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Containment Building at the Reference PWR Following the Scenario 2 Accident[(a,b)]

Waste Category	Burial Volume (m^3)	Estimated Radioactivity Content (Ci)	Disposable Container Requirements: Type	Disposable Container Requirements: Number	Number of Shipments	Packaging and Transport Costs: Disposable Container Cost[(c)] ($)	Packaging and Transport Costs: Cask Rental Charge[(d)] ($)	Packaging and Transport Costs: Transport Costs[(e,f)] ($)	Disposal Site Costs[(f,g)]: Shallow-Land Burial Ground Costs ($): Disposal Charge	Shallow-Land Burial Ground Costs ($): State Surcharge	Shallow-Land Burial Ground Costs ($): Handling Surcharge	Shallow-Land Burial Ground Costs ($): Liner Surcharge	Shallow-Land Burial Ground Costs ($): Curie Surcharge	Disposal Site Costs: Federal Repository Costs ($)	Total Waste Management Costs ($)
Sludge	5	200	0.21-m^3 steel drum	25	2	750	5 400	8 210	2 710	50	2 440				19 560
Process Solids															
Filter Cartridges	3	470	0.3-m^3 steel liner	8	4		10 800	16 420						20 000	47 220
Zeolite Liners	6	432 000	0.3-m^3 steel liner	20	10		27 000	41 050						50 000	118 050
Organic Resin Liners	4	2 350	0.3-m^3 steel liner	12	6		16 200	24 630						30 000	70 830
Process Solids															
Filter Cartridges	1	250	0.21-m^3 steel drum	1	0	30			2 500	10			660		3 200
Evaporator Bottoms	92	280 000	2.85-m^3 steel liner	32	32	480 000	57 600	131 360						320 000	988 960
Organic Resins	51	288	0.21-m^3 steel drum	240	18	7 200	47 250	73 890	17 110	540	22 000				167 990
Chemical Decontamination Solutions	253	70	0.21-m^3 steel drum	1 250	11	37 500		25 980	80 790	2 790	8 060				155 120
Trash															
Compactible, Combustible	119	198	0.21-m^3 steel drum	565	5	16 950		11 810	39 920	1 260	3 660				73 600
Compactible, Noncombustible	385	64	0.21-m^3 steel drum	1 830	15	54 900		35 430	118 270	4 080	10 990				233 670
Noncompactible	970	29	3.5-m^3 plywood box	277	46	110 800		108 650	297 980	10 280	19 120				546 830
Contaminated Equipment															
LSA Materials	108	4	plywood box	21	5	10 800		11 810	33 180	1 140	2 740				59 670
High-Activity Materials	23	80	2.85-m^3 steel liner	8	8	16 000	14 400	32 840	7 060	240	8 050	4 750			83 340
Irradiated Hardware															
LSA Materials	28	8	3.5-m^3 plywood box	8	4	3 200	7 200	16 420	9 390	300	4 300				40 810
High-Activity Materials	52	76 000	2.85-m^3 steel liner	18	18	36 000	32 400	73 890	15 970	550	18 110	90 000	14 220		281 140
Fuel Assemblies															
Intact Assemblies															
Damaged Assemblies	58		0.3-m^3 steel canister	193	49	1 158 000	2 940 000	894 250						20 844 000	25 836 250
Fuel Core Debris	1	400	0.3-m^3 steel canister	4	1	24 000	60 000	18 250						100 000	202 250
Subtotals															
Wastes Sent to Shallow Land Burial	2 005	27 191		4 243	132	294 130	106 650	398 930	624 880	21 240	99 470	94 750	14 880		1 654 930
Wastes Sent to Federal Repository	105	714 820		72	52	480 000	111 600	213 460						420 000	1 225 060
Reactor Fuel and Fuel Core Debris	59	......		197	50	1 182 000	3 000 000	912 500						20 944 000	26 038 500
Totals	2 169			4 512	234	1 956 130	3 718 250	1 524 890	624 880	21 240	99 470	94 750	14 880	21 364 000	28 918 490

(a) Numbers of significant figures shown are for computational accuracy only.
(b) Packaging and disposal requirements for radioactive wastes from accident cleanup in the containment building are given in Table E.4-2 of Appendix E.
(c) Based on information from Table I.2-1 of Appendix I.
(d) Based on information from Table I.2-2 of Appendix I. Assumes 6 days for a round-trip truck shipment; 15 days for a round-trip rail shipment.
(e) Based on information from Table I.3-4 of Appendix I. Includes overweight charges and second driver costs where applicable.
(f) Charges are computed on the assumption that all shipments for a given waste category are identical. In fact, charges for individual shipments would vary depending on the specific physical and radiological characteristics of individual shipments. The averaging technique used is believed to be appropriate for computing total charges.
(g) Based on information from Table I.4-1 of Appendix I.

TABLE F.3-5. Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Containment Building at the Reference PWR Following the Scenario 3 Accident[a,b]

Waste Category	Burial Volume (m^3)	Estimated Radioactivity Content (Ci)	Disposable Container Requirements: Type	Disposable Container Requirements: Number	Number of Shipments	Packaging and Transport Costs: Disposable Container Cost[c] ($)	Packaging and Transport Costs: Cask Rental Charge[d] ($)	Packaging and Transport Costs: Transport Costs[e,f] ($)	Disposal Site Costs[f,g]: Shallow-Land Burial Ground Costs ($): Disposal Charge	Shallow-Land Burial Ground Costs ($): State Surcharge	Shallow-Land Burial Ground Costs ($): Handling Surcharge	Shallow-Land Burial Ground Costs ($): Liner Surcharge	Shallow-Land Burial Ground Costs ($): Curie Surcharge	Federal Repository Costs ($)	Total Waste Management Costs ($)
Sludges	9	1000	0.21-m^3 steel drum	40	3	1 200	8 100	12 320	6 320	100	3 670		2 070		33 780
Process Solids															
Filter Cartridges	11	3 020	0.3-m^3 steel liner	35	18		47 250	73 890						87 500	208 640
Zeolite Liners	20	3 000 000	0.3-m^3 steel liner	66	33		89 100	135 460						165 000	389 560
Organic Resin Liners	19	15 100	0.3-m^3 steel liner	62	31		83 700	127 260						155 000	365 960
Process Solids															
Filter Cartridges	2	2 000	0.21-m^3 steel drum	8	1	240	1 350	4 110	5 000	20	1 220		840		12 780
Evaporator Bottoms	180	2 000 000	2.85-m^3 steel liner	63	63	945 000	113 400	258 620						630 000	1 947 020
Organic Resins	357	2 040	0.21-m^3 steel drum	1 700	122	51 000	328 050	500 810	119 760	3 780	149 100				1 152 500
Cnemical Decontamination Solutions	788	500	0.21-m^3 steel drum	3 750	32	112 500		75 580	242 070	8 350	23 440				461 940
Trasn															
Compactible, Combustible	250	416	0.21-m^3 steel drum	1 190	10	35 700		23 620	83 860	2 650	7 320				153 150
Compactible, Noncombustible	308	135	0.21-m^3 steel drum	3 845	32	115 350		75 580	248 220	8 560	23 440				471 150
Noncompactible	2 040	61	3.5-m^3 plywood box	583	97	232 200		229 110	626 690	21 620	40 330				1 150 950
Contaminated Equipment															
LSA Materials	192	9	plywood box	45	9	20 400		21 260	58 980	2 040	4 930				107 610
High-Activity Materials	46	160	2.85-m^3 steel liner	16	16	32 000	28 800	65 680	14 130	490	16 100	9 500			166 700
Irradiated Hardware															
LSA Materials	28	8	2.5-m^3 plywood box	8	4	3 200	7 200	16 420	9 390	300	4 300				40 810
High-Activity Materials	217	4 000 000	2.15-m^3 shielded steel liner	76	76	1 140 000	136 800	311 980	66 660	2 300	76 450	380 000	410 160		2 524 350
Fuel Assemblies															
Intact Assemblies															
Damaged Assemblies	58		0.3-m^3 steel canister	193	49	1 158 000	2 940 000	894 250						20 844 000	25 836 250
Fuel Core Debris	4	1 200	0.3-m^3 steel canister	12	3	72 000	180 000	54 750						300 000	606 750
Subtotals															
Wastes Sent to Shallow Land Burial	4 737	4 006 329		11 261	402	1 744 790	510 300	1 336 470	1 481 080	50 210	350 300	389 500	413 070		6 275 720
Wastes Sent to Federal Repository	230	5 018 120		226	145	945 000	333 450	595 230						1 037 500	2 911 180
Reactor Fuel and Fuel Core Debris	62			205	52	1 230 000	3 120 000	949 000						21 144 000	26 443 000
Totals	5 029			11 692	599	3 919 790	3 963 750	2 880 700	1 481 080	50 210	350 300	389 500	413 070	22 181 500	35 629 900

(a) Numbers of significant figures shown are for computational accuracy only.
(b) Packaging and disposal requirements for radioactive wastes from accident cleanup in the containment building are given in Table E.4-2 of Appendix E.
(c) Based on information from Table I.2-1 of Appendix I.
(d) Based on information from Table I.2-2 of Appendix I. Assumes 6 days for round-trip truck shipment; 15 days for round-trip rail shipment.
(e) Based on information from Table I.3-4 of Appendix I. Includes overweight charges and second driver costs where applicable.
(f) Charges are computed on the assumption that all shipments for a given waste category are identical. In fact, charges for individual shipments would vary depending on the specific physical and radiological characteristics of individual shipments. The averaging technique used is believed to be appropriate for computing total charges.
(g) Based on information from Table I.4-1 of Appendix I.

the cost of waste management. The major cost item for wastes shipped to a federal repository is for disposal of the reactor fuel from defueling following an accident.

The volumes of radioactive waste from accident cleanup in the containment building estimated to be disposed of by shallow-land burial are 931 m^3 for the scenario 1 accident, 2005 m^3 for the scenario 2 accident, and 4737 m^3 for the scenario 3 accident. For comparison purposes, the waste capacity of a typical shallow-land burial trench is estimated to be about 8300 m^3.[5]

F.3.3 Cost of Energy

Significant quantities of electrical energy are required to operate essential systems and services and the pumps and motors needed during accident cleanup in the containment building. Costs of electrical energy are estimated to represent about 12% of the total cost of cleanup following the postulated accidents.

The following bases and assumptions are used to calculate the costs of electricity during accident cleanup in the containment building:

1) The cold shutdown plant load at the reference PWR is about 22 MW.[1]

2) Use of the RCS pumps during chemical decontamination would add about 18 MW to the base load while the pumps are running.[1]

3) Operation of the demineralizer system adds about 0.5 MW to the base load while the system is operating.

4) At an assumed efficiency of 40%, 1.8 MWh are required to evaporate 1 m^3 of water.

F.3.4 Costs of Special Tools and Equipment

The estimated costs for the special tools and equipment anticipated to be used during accident cleanup in the containment building are presented in Table F.3-6. Estimated costs of equipment for removing damaged fuel from the reactor and packaging it in canisters for storage include research and development costs as well as fabrication costs.

TABLE F.3-6. Estimated Costs of Special Tools and Equipment for Accident Cleanup in the Containment Building

Tool or Equipment Item	Estimated Unit Cost[a] ($ thousands)	Accident Cleanup Following Scenario 1 Accident: Number Required	Accident Cleanup Following Scenario 1 Accident: Total Cost ($ thousands)	Accident Cleanup Following Scenario 2 Accident: Number Required	Accident Cleanup Following Scenario 2 Accident: Total Cost ($ thousands)	Accident Cleanup Following Scenario 3 Accident: Number Required	Accident Cleanup Following Scenario 3 Accident: Total Cost ($ thousands)
Underwater manipulator	1 000	1	1 000	1	1 000	1	1 000
Underwater plasma-arc torch	20	1	20	2	40	3	60
Underwater oxyacetylene torch	5	1	5	2	10	4	20
Arc saw	120	1	120	1	120	1	120
Portable plasma-arc torch	20	1	20	2	40	4	80
Portable oxyacetylene torch	1	2	2	2	2	4	4
Guillotine pipe saw	4	2	8	4	16	4	16
Power-operated reciprocating hacksaw	1	2	2	4	4	5	5
Closed-circuit, high-resolution TV systems	50	2	100	2	100	4	200
Underwater lights and viewing aids		AR[b]	10	AR	20	AR	40
Underwater tools (e.g., impact wrenches, cutters, tongs)		AR	25	AR	50	AR	100
Submersible pump with disposable filters	2	2	4	4	8	6	12
High-pressure water jet	20	4	80	4	80	6	120
Scaffolding and safety nets		AR	10	AR	15	AR	20
Shielded vehicle with manipulator arms and interchangeable tools	120	1	120	1	120	1	120
Power-operated mobile manlift	40	1	40	2	80	2	80
9100-Kg mobile hydraulic crane	28	1	28	2	56	2	56
9100-Kg forklift	28	2	56	2	56	2	56
Rigging materials (e.g., chokers, grapples, winches)		AR	10	AR	20	AR	50
Vacuum cleaner (HEPA filtered)	4	3	12	4	16	6	24
Portable ventilation enclosure	10	6	60	10	100	16	160
Supplied-air plastic suit	0.2	30	6	50	10	100	20
Waste compactor	12	1	12	1	12	1	12
Incinerator	250	1	250	1	250	1	250
Underwater vacuum system	25	1	25	1	25	1	25
Equipment for removing damaged fuel from reactor[c]			500		3 000		10 000
Equipment for loading damaged fuel into canisters[c]			500		1 000		1 000
Total Costs			3 025		6 250		13 650

(a) From Table I.5-1 of Appendix I. Costs are in early-1981 dollars.
(b) AR: as required.
(c) Includes research and development costs as well as costs of fabrication.

F.3.5 Costs of Miscellaneous Supplies

Expendable supplies used for accident cleanup in the containment building include decontamination chemicals, protective clothing, filters and ion exchange resins, mechanical and electrical supplies, cleaning supplies, and expendable tools. The estimated costs for these miscellaneous supplies are presented in Table F.3-7.

F.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. Costs for these contractor services are estimated as described in Section F.1.6.

F.3.7 Costs of Nuclear Insurance and License Fees

Estimated costs for nuclear liability insurance and for license fees required during accident cleanup in the containment building are presented in Table F.3-8.

F.4 SUMMARY OF THE COSTS OF ACCIDENT CLEANUP AT THE REFERENCE PWR

Based on the accident cleanup activities at the reference PWR as described in Appendix E, and on the cost information for the accident cleanup activities described in Sections F.1, F.2, and F.3, the total estimated costs of accident cleanup following the reference accidents are summarized in Table F.4-1. Accident cleanup costs are estimated to be $105.2 million following the scenario 1 accident, $223.8 million following the scenario 2 accident, and $404.5 million following the scenario 3 accident. These costs include planning and preparation costs as well as the actual costs of cleanup in the auxiliary and fuel buildings and the containment building. For cost estimating purposes, planning and preparation is 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.

TABLE F.3-7. Estimated Costs of Miscellaneous Supplies for Accident Cleanup in the Containment Building

Item	Accident Cleanup Following Scenario 1 Accident		Accident Cleanup Following Scenario 2 Accident		Accident Cleanup Following Scenario 3 Accident	
	Quantity	Total Cost ($ thousands)[a]	Quantity	Total Cost ($ thousands)[a]	Quantity	Total Cost ($ thousands)[a]
Decontamination chemicals						
EDTA/oxalic/citric acid	19 000 kg	31	19 000 kg	31	38 000 kg	62
OPG solution			380 m^3 of solution	22	760 m^3 of solution	43
Respirator facepieces	100 ea	10	200 ea	20	500 ea	50
Anticontamination clothing	11 000 sets[b]	550	22 300 sets[b]	1 115	47 300 sets[b]	2 365
Cleaning supplies	see note (c)	225	see note (c)	420	see note (c)	750
Expendable tools	see note (d)	150	see note (d)	280	see note (d)	500
Ion-exchange resins	15 m^3	75	30 m^3	150	60 m^3	300
Filters	unspecified	200	unspecified	400	unspecified	600
Mechanical supplies and hardware	unspecified	50	unspecified	100	unspecified	300
Electrical components and cables	unspecified	25	unspecified	50	unspecified	200
Ion-exchange and filter liners	13 ea	65	40 ea	200	163 ea	815
Canisters for damaged fuel	20 ea	100	193 ea	965	193 ea	965
Total Costs		1 486		3 753		6 950

(a) Costs are in early-1981 dollars and are rounded to the nearest $100.
(b) Estimated at two clothing changes per shift per cleanup worker. One set of clothing can be laundered and used four times.
(c) Estimated at $150,000 per year.
(d) Estimated at $100,000 per year.

TABLE F.3-8. Estimated Costs of Nuclear Liability Insurance and License Fees During Accident Cleanup in the Containment Building

Category	Unit Cost ($)	Total Cost ($)[a] Cleanup Following Scenario 1 Accident	Cleanup Following Scenario 2 Accident	Cleanup Following Scenario 3 Accident
Property Damage Insurance	1 000 000/yr	1 500 000	3 000 000	5 000 000
Nuclear Liability Insurance	400 000/yr	600 000	1 200 000	2 000 000
License Fees[b]				
Routine Health, Safety, and Environmental Inspections	75 700/yr	113 550	227 100	378 500
Routine Safeguards Inspections	11 800/yr	17 700	35 400	59 000
Total Costs		2 231 250	4 462 500	7 437 500

(a) Costs are in early-1981 dollars.
(b) From 10 CFR 170.

TABLE F.4-1. Summary of Estimated Total Costs of Accident Cleanup Following the Reference Accidents

Cleanup Operation	Costs of Cleanup Following Scenario 1 Accident[a] ($ millions)	Costs of Cleanup Following Scenario 2 Accident[a] ($ millions)	Costs of Cleanup Following Scenario 3 Accident[a] ($ millions)
Preparations for Accident Cleanup	33.7	67.2	98.0
Accident Cleanup in Auxiliary and Fuel Buildings	--[b]	19.5[c]	19.5[c]
Accident Cleanup in Containment Building	71.5	137.1	287.0
Total Accident Cleanup Costs	105.2	223.8	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) 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.

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. R. S. Godfrey, ed., Building Construction Cost Data - 1981, 39th Edition, R. S. Means Co., Inc., Kingston, Massachusetts, 1980.

3. Capital Cost: Pressurized Water Reactor Power Plant, NUREG-0241, U.S. Nuclear Regulatory Commission, Washington, D.C., June 1977.

4. M. G. McGraw, "Nuclear 'Look-Alike' Enhances M&R Training," Electrical World, February 1981, pp. 109-110.

5. E. S. Murphy and G. M. Holter, 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.

APPENDIX G

DETAILS OF 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. As discussed in Chapter 4 of Volume 1, there are three basic decommissioning alternatives:

- DECON - the immediate removal of all radioactive material to permit license termination and unrestricted release of the property
- SAFSTOR - preparation and maintenance of the property so that risk to public safety is acceptable for a period of storage until either the facility is decontaminated or the residual radioactivity decays to an unrestricted release level
- ENTOMB - the encasement and maintenance of property in a strong and structurally long-lived material to ensure retention and isolation from the environment until the contained radioactivity decays to an unrestricted release level.

Selection of the decommissioning alternative to be used at an accident-damaged LWR is essentially independent of the accident cleanup activities that precede the decommissioning.

This appendix provides the details of post-accident decommissioning activities at the reference PWR following completion of the accident cleanup campaign. (Accident cleanup is discussed in Appendix E.) A comparison of normal and post-accident decommissioning requirements is provided in Section G.1. The details of the post-accident decommissioning of the reference PWR by the DECON, SAFSTOR, or ENTOMB alternatives are given in Sections G.2, G.3, and G.4, respectively. The costs associated with these decommissioning activities are presented in Appendix H. General information on decontamination, liquid waste treatment, and packaging and disposing of wastes is given in Appendix D.

The post-accident decommissioning analyses in this study use the results of previous analyses of PWR decommissioning following normal reactor shutdown, presented in References 1 and 2, with appropriate modifications as necessary to account for post-accident conditions. In addition, the analysis of post-accident decommissioning of TMI-2 presented in Reference 3 provides useful background information for this study. The decommissioning analyses presented in this study provide a quantitative assessment of the requirements and costs for decommissioning following a scenario 2 accident. (The three accident scenarios are described in Chapter 8 and Appendix C.) Variations in decommissioning requirements and costs that would result from the other two accident scenarios are discussed where applicable.

A basic assumption of the analyses presented in this appendix 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 that would result from an inability to dispose of the wastes offsite at the time of decommissioning is presented in Chapter 15 of Volume 1.

G.1 COMPARISON OF NORMAL VERSUS POST-ACCIDENT DECOMMISSIONING OF THE REFERENCE PWR

Under normal circumstances, decommissioning of an LWR follows the orderly shutdown of the facility at the end of its planned operating life. However, the situation at a reactor that has experienced an accident is significantly different from normal, with moderate to severe contamination of the major plant buildings, damage to the reactor core, and possible physical damage to plant equipment and services. As a result, decommissioning following an accident may differ significantly from that following normal shutdown.

It is assumed in this study that the accident cleanup activities are completed prior to the start of the actual decommissioning effort. These cleanup activities are discussed in detail in Appendix E. The principal goals of accident cleanup are:

- to provide initial decontamination of certain plant systems and of selected building surfaces and equipment so as to reduce to a practicable level the radiation doses to workers engaged in defueling the reactor and in subsequent decommissioning activities
- to safely defuel the reactor and place the fuel in a configuration that is safe from nuclear criticality and/or fuel meltdown
- to remove and process the accident water in the facility.

The tasks that must be performed to accomplish these goals are postulated to be independent of the alternative (DECON, SAFSTOR, or ENTOMB) chosen to complete the decommissioning, although the methods used to complete certain tasks may vary with the decommissioning alternative. The work required to complete each task will certainly be influenced by the severity of the accident.

In carrying out the accident cleanup activities, certain tasks that would be part of the decommissioning process following normal shutdown are completed, and significant portions of other such tasks are undertaken. Examples of tasks completed during cleanup are reactor defueling, comprehensive radiation surveys of the facility, and decontamination of the reactor coolant recirculation and purification systems. Decommissioning tasks partially completed during cleanup include removal and segmentation of reactor vessel internals, decontamination of internal surfaces in the reactor containment, and removal of spent fuel storage racks from the spent fuel pool.

The requirements of carrying out the accident cleanup activities also result in certain new tasks that must be completed during the decommissioning process. In general, these new tasks are limited to the removal of new equipment installed to process accident water and the decommissioning of the temporary onsite waste storage structures specially constructed for the management of wastes resulting from accident cleanup activities. (In the event that the accident cleanup wastes cannot be shipped offsite for disposal at the time of decommissioning, onsite storage of the wastes will be extended and decommissioning of the onsite waste storage structures will be deferred. See Chapter 15 for additional information).

A number of decommissioning tasks are independent of accident cleanup activities and, thus, are common to both post-accident and normal-shutdown decommissioning. However, the changes in the physical and radiological condition of the plant resulting from an accident lead to substantial qualitative changes in a number of these decommissioning tasks. Manpower requirements for carrying out specific tasks are related to a number of factors (e.g., the physical condition of the equipment and structures, local radiation dose rates, and the methods used to complete tasks) that may be affected by the accident and the subsequent cleanup program. Radiation doses to decommissioning workers are likely to be higher than those following normal shutdown because of the increased contamination of equipment, piping, and structural surfaces caused by the accident. The schedule and sequence of events for any particular decommissioning alternative may need to be revised to account for these changes and for the addition and deletion of specific tasks as a result of accident cleanup, as discussed previously. Furthermore, the requirements for special decommissioning tools and equipment may vary somewhat because of changes in specific tasks and because some of these tools and equipment items may be available for reuse as a result of the accident cleanup campaign that precedes the decommissioning. In summary, even tasks common to both post-accident and normal-shutdown decommissioning can be expected to differ significantly between the two situations.

Although the accident cleanup activities remove a large portion of the accident-generated contamination in the plant, accident severity will likely have some impact on the decommissioning tasks subsequent to cleanup. Radiation doses to decommissioning workers are likely to increase with accident severity because of the increased level and spread of radioactive contamination in the plant resulting from the accident. In addition, physical damage to the plant from the more severe accidents may compromise certain systems, structural features, and equipment items that are required to carry out the decommissioning tasks, thus necessitating repairs and/or substitutions and resulting in delays and additional expenses. In areas of extensive plant damage, different methods may be required to accomplish certain decommissioning tasks. It should be noted that the effects of accident severity on the level of effort required to complete the decommissioning activities are much less than the corresponding effects on the cleanup activities.

Comparisons of the applicable activities for normal versus post-accident decommissioning of the reference PWR by DECON, SAFSTOR, and ENTOMB are presented in Tables G.1-1, -2, and -3, respectively. The information pertaining to normal decommissioning is derived from References 1 and 2. The details of the post-accident decommissionings are presented in the subsequent sections of this appendix.

G.2 DECON AT THE REFERENCE PWR

In general, DECON is the decommissioning alternative used to remove from the facility, as soon as practicable following final shutdown, all materials with radioactive contamination above unrestricted release levels. For a reactor that has experienced an accident, DECON begins following accident cleanup and is postulated to be completed within about 2-3/4 years. After DECON is completed and the radioactive materials are shipped from the site, the nuclear license can be terminated and the facility and the site can be released for unrestricted use. In the reference PWR, the principal plant structures containing radioactive materials at the start of decommissioning are the containment building, the fuel building, and the auxiliary building.

Details of post-accident DECON at the reference PWR are discussed in this section, including disassembly methods, schedules and manpower requirements, and external occupational radiation doses. These details are based largely on the analysis of DECON at the reference PWR following normal shutdown, presented in Appendix G and Chapter 9 of Reference 1, because, after cleanup is completed, many of the requirements for DECON are similar whether or not the reactor has experienced an accident. Where the postulated accident results in significant changes in the DECON requirements, the differences in the requirements are identified and new information is developed to support the analysis. The analysis presented is based on the assumption that the reactor has experienced a scenario 2 accident. Variations in DECON requirements with changes in the severity of the accident are discussed.

TABLE G.1-1. Comparison of Activities for Normal Versus Post-Accident DECON at the Reference PWR

Task	Applicable to:(a) DECON Following Normal Shutdown	Post-Accident Cleanup	Post-Accident DECON
Containment Building			
1. Comprehensive Radiation Survey	X	X	
2. Initial Decontamination of Containment Building		X	
3. Move Reactor Vessel Internals to Refueling Cavity	X	P	P
4. Segment Vessel Internals and Load Containers	X	P	P
5. Chemical Decon Reactor Coolant System	X	X	
6. Segment Reactor Pressure Vessel and Load Containers	X		X
7. Segment Steam Generators (4) and Prepare for Shipment	X		X
8. Remove Reactor Coolant System Pumps (4) and Piping	X		X
9. Remove Pressurizer, Relief Tank, and Safety Injection System	X		X
10. Remove Heat Exchangers and Assorted Pumps	X		X
11. Remove Contaminated Internal Structures	X		X
12. Remove Spray Piping and Ventilation Systems	X		X
13. Decontaminate Building Internal Surfaces	X	P	X
14. Final Radiation Survey	X		X
Fuel Building			
1. Chemical Decon Chemical Volume Control System	X	P	X
2. Remove Chemical Volume Control System	X		X
3. Remove Boric Acid System	X		X
4. Remove Concentrate Holding Tank System	X		X
5. Remove Spent Fuel Storage Racks, New Fuel Storage Racks, and Fuel Transfer System	X	p(b)	X
6. Remove Accident-Water Cleanup Demineralizer System			X
7. Remove Spent Fuel Cleanup System, Liners, and Contaminated Concrete	X		X
8. Remove Closed Cooling Water System	X		X
9. Final Radiation Survey	X		X
Auxiliary Building			
1. Decontamination (Contact Work and Internal Flushes)	X	P	X
2. Remove and Package Selected Building Internals	X		X
3. Remove IX Resins and Liquid System Filters	X	X	X
4. Remove IX and Filter System Piping	X		X
5. Remove Tanks, Pumps, and Heat Exchangers	X		X
6. Remove HVAC, Fire Sprinkler, and Monorail Systems	X		X
7. Remove Electrical Equipment	X		X
8. Final Radiation Survey	X		X
Ancillaries			
1. Process Accident Water		X	
2. Defuel Reactor	X(c)	X	
3. Ship Spent Fuel Offsite	X		X
4. Packaging and Shipment of Contaminated Equipment and Debris	X		X
5. Packaging and Shipment of Combustible Wastes	X	X	X
6. Decontamination and Removal in Other Buildings as Required	X		X
7. Remove Onsite Waste Storage Structures			X

(a) Applicability indicated by X, partial applicability by P, nonapplicability by a blank; chronological order of tasks may vary from that shown.
(b) Sufficient spent fuel racks removed to provide room for special racks to handle canistered fuel.
(c) Considered to be part of normal shutdown procedures and charged to reactor operations, even though carried out after start of decommissioning.

TABLE G.1-2. Comparison of Activities for Normal Versus Post-Accident SAFSTOR at the Reference PWR

Task	Applicable to:[a] SAFSTOR Following Normal Shutdown	Post-Accident Cleanup	Post-Accident SAFSTOR
Containment Building			
1. Comprehensive Radiation Survey	X	X	
2. Initial Decontamination of Containment Building		X	
3. Move Reactor Vessel Internals to Refueling Cavity		P(b)	
4. Segment Vessel Internals and Load Containers		P(b)	
5. Chemical Decon Reactor Coolant System	X	X	
6. Decontaminate Reactor Cavity and Refueling Cavity	X	P	X
7. Decontaminate Building Internal Surfaces	X	P	X
8. Install Intrusion, Radiation, and Fire Alarms	X		X
9. Final Radiation Survey	X		X
Fuel Building			
1. Chemical Decon Chemical Volume Control System	X	P	X
2. Remove Spent Fuel Storage Racks		P(c)	
3. Remove Accident-Water Cleanup Demineralizer System			X
4. Decontamination (Contact Work and Internal Flushes)	X	P	X
5. Spent Fuel Pool Draining, Decontamination, and Cover Installation	X		X
6. Install Intrusion, Radiation, and Fire Alarms	X		X
7. Final Radiation Survey	X		X
Auxiliary Building			
1. Remove IX Resins and Liquid System Filters	X	X	X
2. Decontamination (Contact Work and Internal Flushes)	X	P	X
3. Place Building in Safe Storage	X		X
4. Install Intrusion, Radiation, and Fire Alarms	X		X
5. Final Radiation Survey	X		X
Ancillaries			
1. Process Accident Water		X	
2. Defuel Reactor	X(d)	X	
3. Ship Spent Fuel Offsite	X		X
4. Purge RCS and CVCS	X		X
5. Packaging and Shipment of Combustible Wastes	X	X	X
6. Decontamination and Removal in Other Buildings as Required	X		X
7. Place Onsite Waste Storage Structures in Safe Storage			X

(a) Applicability indicated by X, partial applicability by P, nonapplicability by a blank; chronological order of tasks may vary from that shown.
(b) Removal of some internals required to remove damaged fuel from the reactor pressure vessel.
(c) Sufficient spent fuel racks removed to provide room for special racks to handle canistered fuel.
(d) Considered to be part of normal shutdown procedures and charged to reactor operations, even though carried out after start of decommissioning.

TABLE G.1-3. Comparison of Activities for Normal Versus Post-Accident ENTOMB at the Reference PWR

Task	ENTOMB Following Normal Shutdown	Post-Accident Cleanup	Post-Accident ENTOMB
Containment Building			
1. Comprehensive Radiation Survey	X	X	
2. Initial Decontamination of Containment Building		X	
3. Move Reactor Vessel Internals to Refueling Cavity	X	P	P
4. Segment Vessel Internals and Load Containers	X	P	P
5. Chemical Decon Reactor Coolant System	X	X	
6. Remove Equipment Above Operating-Floor Level and Place in Storage Below (Including Relocation of Steam Generators and Pressurizer)	X		X
7. Cut and Seal Containment Penetrations	X		X
8. Form and Pour Entombment Barrier	X		X
9. Decontaminate Internal Surfaces Above Operating-Floor Level	X	P	X
10. Install Security and Surveillance Monitoring Equipment	X		X
11. Final Radiation Survey	X		X
Fuel Building			
1. Chemical Decon Chemical Volume Control System	X	P	X
2. Remove Chemical Volume Control System	X		X
3. Remove Boric Acid System	X		X
4. Remove Concentrate Holding Tank System	X		X
5. Remove Spent Fuel Storage Racks, New Fuel Storage Racks, and Fuel Transfer System	X	P(b)	X
6. Remove Accident-Water Cleanup Demineralizer System			X
7. Remove Spent Fuel Cleanup System, Liners, and Contaminated Concrete	X		X
8. Remove Closed Cooling Water System	X		X
9. Final Radiation Survey	X		X
Auxiliary Building			
1. Decontamination (Contact Work and Internal Flushes)	X	P	X
2. Remove Selected Building Internals and Package or Transfer to Entombment Area	X		X
3. Remove IX Resins and Liquid System Filters	X	X	X
4. Remove IX and Filter System Piping	X		X
5. Remove Tanks, Pumps, and Heat Exchangers	X		X
6. Remove HVAC, Fire Sprinkler, and Monorail Systems	X		X
7. Remove Electrical Equipment	X		X
8. Final Radiation Survey	X		X
Ancillaries			
1. Process Accident Water		X	
2. Defuel Reactor	X(c)	X	
3. Ship Spent Fuel Offsite	X		X
4. Disposal (Placement in Entombment Area or Packaging and Shipment) of Contaminated Equipment and Debris	X		X
5. Packaging and Shipment of Combustible Wastes	X	X	X
6. Decontamination and Removal in Other Buildings as Required	X		X
7. Entombment of Onsite Waste Storage Structures			X

Applicable to:(a)

(a) Applicability indicated by X, partial applicability by P, nonapplicability by a blank; chronological order of tasks may vary from that shown.
(b) Sufficient spent fuel racks removed to provide room for special racks to handle canistered fuel.
(c) Considered to be part of normal shutdown procedures and charged to reactor operations, even though carried out after start of decommissioning.

The facility description given in Appendix B provides the basic information that supports the development of the tasks, schedules, manpower loadings, and occupational radiation exposure estimates presented here. Additional details pertinent to specific DECON activities come from engineering drawings, manufacturers' data, and Reference 1.

The information in this section forms the basis for the estimated costs and safety impacts of post-accident DECON at the reference PWR which are developed in Appendices H and J, respectively, and for the comparison of post-accident and normal-shutdown DECON of the reference PWR that is presented in Chapter 17 of Volume 1.

G.2.1 Disassembly Methods

The disassembly methods used for post-accident DECON at the reference PWR are generally the same as those for DECON following normal shutdown, which are described in detail in Reference 1. The disassembly methods used for post-accident DECON are summarized here, with appropriate details included where these differ significantly from the ones used following normal shutdown.

The disassembly methods proposed for decontamination of the reference PWR following accident cleanup employ techniques that have been used successfully and are described generically in Appendix D and in Reference 1. Disassembly methods are discussed in the following subsections for each of the three buildings containing significant amounts of radioactive materials. Ancillary activities required to complete DECON are also discussed.

DECON begins in the containment building, which comprises the major effort for the decommissioning staff. The containment building contains the neutron-activated materials and the bulk of the activated corrosion products and the accident-generated fission-product contamination remaining after accident cleanup activities are completed. The work proceeds through the fuel building, where the decontamination activities take place in steps that depend on when the various systems in the building are no longer needed to support other decommissioning activities. The auxiliary building contains the bulk of the radioactive waste treatment systems and, since these systems are required to be in service during much of the DECON effort, final dismantlement of the auxiliary building takes place rather late in the overall schedule. The ancillary activities are performed on a schedule that depends on the need for the plant areas involved and on the availability of manpower to perform the activities.

G.2.1.1 Containment Building

All of the neutron-activated materials and the major portion of the radioactive contamination (both from normal operations and from the postulated accident) are located in the containment building of the reference PWR. Most of the reactor vessel internals and portions of the reactor pressure vessel and the reactor cavity concrete contain neutron-activation products. Neutron-activated components are cut into pieces that will fit steel liners for shielded shipping casks. Radioactively contaminated materials include equipment items, piping, structural members, liner plates, and concrete. Contaminated components are removed and cut up as required for packaging in 0.21-m^3 steel drums, in standard shipping boxes (1.2 m x 1.2 m x 2.4 m), or in specially made boxes.

Methods postulated for the removal of the radioactive materials from the containment building during normal decommissioning are presented in Table G.1-1 of Reference 1. In general, these same methods are postulated to be used for post-accident decommissioning. Differences from the normal disassembly procedure in the containment building are discussed in the following paragraphs.

During the postulated accident at the reference PWR, all interior spaces of the containment building are contaminated with radioactivity released from the reactor coolant system. Thus, following the accident, 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. In order to carry out the primary objectives of accident cleanup (i.e., removal of the accident water and defueling of the reactor), much of this excess contamination is removed during cleanup to reduce radiation exposure rates to workers to practicable levels. However, these cleanup activities require worker access to only certain portions of the containment building and, therefore, some areas of the building still contain substantial accident-generated contamination following the cleanup campaign. Because DECON requires access to all portions of the containment building, the major access routes used by the DECON workers must be cleaned up or shielded to keep radiation doses to workers within reasonable limits. Furthermore, hot spots (areas with exceptionally high contamination levels) outside of the access routes that can materially affect worker doses are also postulated to be cleaned up or shielded. This task is undertaken at the start of DECON to obtain the maximum dose-reduction benefits. The methods used during post-accident DECON are the same as those used during normal decommissioning or during accident cleanup and include hose washing, vacuuming, sweeping, damp mopping, and scrubbing. Because the amount of contamination varies with accident severity, the level of effort required for this task increases with increasing accident severity.

It is postulated in this study that a portion of the reactor vessel internals must be removed during the accident cleanup campaign to facilitate the complete removal of the spent reactor fuel and associated debris. Following both the scenario 1 and 2 accidents, only the upper core support structure must be removed for defueling. However, following the relatively severe fuel damage resulting from the scenario 3 accident, the core shroud and lower core support structure must also be removed to completely defuel the reactor and remove the fuel debris from the reactor. For all of the accident scenarios, it is postulated that the internals removed during cleanup are also segmented and packaged for disposal at that time. The remainder of the vessel internals (i.e., those that do not interfere with fuel removal) are postulated to be removed and segmented during the decommissioning activities that follow cleanup. The methods used for internals removal and segmentation are the same as those employed for normal decommissioning; however, some additional difficulties may be encountered because of accident-caused damage to the internals and higher radiation exposure rates in the work area. Therefore, the removal activities may proceed at a somewhat slower pace than following normal reactor shutdown.

Chemical decontamination of the reactor coolant system is postulated in Reference 1 for DECON following normal shutdown to reduce both worker doses and shielding requirements for the packaged components after removal and segmentation. In this study, chemical decontamination of the reactor coolant system is postulated to occur during accident cleanup, and no further decontamination is anticipated to be required during DECON.

As stated previously, decontamination of internal surfaces in the containment building is postulated to be initiated during accident cleanup and to be carried to a point that reduces to reasonable levels the radiation doses to the cleanup workers. However, the bulk of this decontamination work, particularly the removal of contaminated structural material, is still carried out during the actual DECON process. The majority of the effort involves the removal of contaminated concrete on floors and walls. The methods used during post-accident DECON (i.e., spalling of concrete, disassembly or cutting of metal components, etc.) are the same as those used during normal decommissioning. However, because of contamination caused by the accident, the level of effort required and the amount of radioactive waste produced are greater than during normal decommissioning, increasing with accident severity.

G.2.1.2 Fuel Building

The accident scenarios postulated for this study result in relatively limited impacts to the fuel building. There is some additional fission-product contamination of the chemical volume control system (CVCS) resulting from fuel damage during the accident, as well as the potential for greater contamination of building surfaces than would be present following normal shutdown of the reactor. These accident-caused effects are postulated to be cleaned up during the accident cleanup campaign to allow hands-on operation and maintenance of systems within the building that are required during the cleanup activities, without excessive radiation doses to the workers involved (see Appendix E). Therefore, changes in post-accident DECON activities in the fuel building are not caused by accident effects. Rather, they result from the use of fuel building facilities during the accident cleanup campaign. Differences from DECON activities in the fuel building following normal shutdown are discussed in the following paragraphs. The tasks required and the schedule for performing them are not anticipated to vary with accident severity, although the occupational radiation doses to the workers are expected to increase somewhat with increased accident severity because of increased contamination levels in certain areas of the building (e.g., the spent fuel pool and fuel transfer tunnel).

Although the chemical volume control system (CVCS) is decontaminated together with the reactor coolant system during accident cleanup, portions of the system are then postulated to be used to store and process accident water. Therefore, some chemical decontamination of the CVCS is required during DECON. This decontamination involves a somewhat lower level of effort than during normal decommissioning because of the system decontamination during cleanup and because only certain portions of the system are used for the accident water processing and thus require additional decontamination.

In order to safely handle the reactor fuel damaged during the accident, some of the original spent fuel racks in the spent fuel storage pool are removed during cleanup and replaced with new, specially fabricated racks that can accommodate the canisters of damaged fuel. The extent of this fuel rack replacement is determined by the severity of the accident, which relates to the amount of fuel damage in the reactor core. During the DECON activities that follow accident cleanup, the spent fuel racks are removed, segmented, and packaged for offsite shipment as radioactive waste. Although some of the original racks have been replaced with others of a different type, no significant impacts on the manpower requirements and the schedule for these activities are anticipated.

It is also postulated that, during the accident cleanup campaign, the demineralizer system used to process the accident water is installed and operated in the spent fuel pool. Therefore, during the DECON activities that follow cleanup, this demineralizer system must be removed, segmented, and packaged for shipment offsite. This is a new requirement for post-accident DECON. Furthermore, this use of the spent fuel pool is anticipated to result in greater than normal contamination levels in the pool. The manpower requirements and the schedule for cleanup and removal of the pool liner and the surrounding concrete are not anticipated to be significantly affected by the increased contamination, but the occupational radiation doses to the workers involved are increased somewhat, depending on the severity of the postulated accident.

G.2.1.3 Auxiliary Building

Any accident-caused impacts to the auxiliary building are postulated to be mitigated during the accident cleanup campaign (see Appendix E). Therefore, the post-accident DECON requirements in the auxiliary building are anticipated to be the same as those for DECON following normal reactor shutdown. The schedule for the auxiliary building is adjusted somewhat to coordinate properly with the revised schedules for the other buildings and to allow for efficient overall completion of all of the DECON tasks.

G.2.1.4 Ancillaries

The ancillary activities for post-accident DECON at the reference PWR are as follows:

- decontamination of other site buildings as required (e.g., the condensate-demineralizer building, the control building, and the turbine building)
- the packaging and shipment of the radioactive wastes generated during DECON (i.e., the activated and contaminated materials removed from the plant and the combustible wastes generated in carrying out the DECON tasks)
- the shipment of the spent fuel from the final reactor core to an offsite repository[a]
- removal of onsite waste storage structures postulated to handle the wastes generated during accident cleanup (including the offsite shipment of the accident-cleanup wastes[a] stored in these structures).

These activities are discussed in the following paragraphs.

The site buildings other than the major ones discussed previously are postulated to contain only very minor amounts of radioactivity at the time of

(a) The costs of packaging, shipping, and disposing of these wastes are included in the cost of accident cleanup, even though the removal of these wastes from the site is not anticipated to be completed during the cleanup campaign and time is allotted during the DECON schedule to complete these activities. The costs of decommissioning the onsite waste storage structures are included in the cost of DECON.

decommissioning, regardless of whether the reactor is shut down normally or an accident has occurred. Therefore, decontamination of these other site buildings is anticipated to be unchanged for post-accident DECON.

The packaging and shipment of radioactive wastes generated during DECON is assumed to be handled by standing crews that are postulated to be available over the entire duration of the tasks that generate the waste (i.e., until DECON activities in the auxiliary building are completed). The amount of waste handled by these crews and, consequently, the radiation doses to the workers involved are anticipated to be greater than those for DECON following normal shutdown and to increase with accident severity. Because the duration of the DECON effort varies only slightly with accident severity, the manpower requirements for this task are almost unaffected by accident severity, except that radiation dose limitations are anticipated to require the addition of manpower as accident severity increases.

Shipment of spent fuel from an operating reactor is a relatively routine procedure and, thus, this task poses no special difficulties for post-accident DECON. The shipping task is assumed to be performed by the waste handling crews described in the previous paragraph. However, because it is postulated that the spent fuel removed after either a scenario 2 or scenario 3 accident is placed in canisters (see Appendix E) and, as a result, fewer fuel assemblies can be placed in a shipping cask, this task is anticipated to take 40-60% longer to complete following either of these postulated accidents, assuming that the same number of shipping casks are available. About 10% of the spent fuel is anticipated to be placed in canisters following a scenario 1 accident, increasing the time required for shipment only slightly from that during normal decommissioning. Doses to workers engaged in the spent fuel shipment are anticipated to increase with accident severity because of increased area dose rates in the vicinity of the spent fuel pool where this task is carried out.

To effectively handle the radioactive waste materials that are generated during the accident cleanup campaign, it is postulated that new onsite structures are built to temporarily store selected wastes. This temporary onsite waste storage allows for a relatively steady offsite shipment of accident cleanup wastes that are generated sporadically because of the nature of the cleanup schedule. In addition, certain highly radioactive wastes generated during accident cleanup require shielding, which the storage structures are designed to provide. These wastes are assumed to be removed from the site and shipped to appropriate repositories before the completion of the DECON activities; the waste shipping is performed by the waste handling crews described previously. Some radioactive contamination of the onsite storage structures is anticipated due to package failures, smearable contamination on package surfaces, and the like. Therefore, these structures require structural decontamination before DECON is completed. The methods used for decontamination of the temporary onsite waste storage structures are the same as those employed in the major plant buildings. Smearable contamination is removed by water-jet cleaning, chemical agents, or manual scrubbing. Contaminated structural components and surfaces are removed, segmented, and packaged for offsite shipment; structural members, conduit, and the like are cut away using conventional methods, while contaminated concrete is removed using the concrete spalling technique described in References 1 and 3. Because the amount of accident cleanup waste requiring onsite storage increases with accident severity, the level of effort required to ship this waste offsite and to decontaminate the waste storage structures also increases with accident severity.

G.2.2 Schedules and Decommissioning Worker Requirements

Development of the schedules and decommissioning worker requirements for DECON requires several steps to arrive at reasonably optimum results. First, the sequence in which the various systems must be decontaminated and removed is determined. Next, the task time requirements and the numbers and types of decommissioning workers required to accomplish each task in the allotted time are estimated. The job sequences are then arranged to require a relatively constant-sized work force.

The primary decommissioning activities are postulated to be performed on a two-shift, 5-day-week basis. However, selected support activities (i.e., system decontamination and radwaste system operation) and security functions are carried out on three shifts, around-the-clock, 7 days per week. In addition, the main control room is manned full time for operation of essential systems and services.

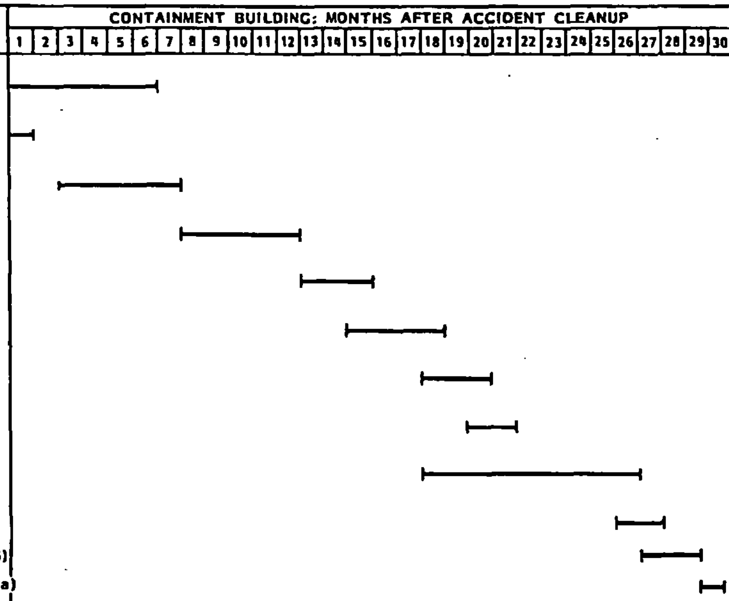
The schedules and decommissioning worker requirements for post-accident DECON at the reference PWR following accident cleanup are presented in this section. The information presented here is based largely on that for DECON following normal shutdown, presented in Appendix G of Reference 1. Schedules and requirements for the three major plant buildings are discussed in the first three subsections. A fourth subsection presents the overall task schedule for post-accident DECON.

G.2.2.1 Containment Building

The majority of the effort involved in DECON at the reference PWR is expended in the containment building. Based on the information describing normal DECON at the reference plant, Section G.2.2 of Reference 1, and on the discussion of disassembly methods presented previously in Section G.2.1.1 of this appendix, the post-accident DECON task schedule and sequence and associated decommissioning worker requirements in the containment building are presented in Figure G.2-1. The schedule shown assumes that the PWR has experienced the postulated scenario 2 (moderate) accident. DECON activities in the containment building are estimated to be accomplished in about 2.5 years; this schedule is several months shorter than DECON following normal shutdown because the reactor defueling, part of the critical path of activities, is accomplished during the accident cleanup campaign, allowing the remainder of the tasks to be undertaken earlier in the schedule.

Accident severity is judged to have only minimal effect on the duration of the DECON activities in the PWR containment building. Although some tasks on the critical path are lengthened by increasing accident severity, this is balanced against the corresponding shortening of Tasks 2 and 3 (reactor vessel internals removal and segmentation) that results from the need to remove more of the internals during accident cleanup to defuel the reactor, thus leaving less of this work for DECON. Manpower requirements are anticipated to vary

TASK (SHIFTS PER DAY/DURATION IN MONTHS)		MANPOWER PER SHIFT: CREW LEADERS	UTILITY OPERATORS	LABORERS	CRAFTSMEN
CLEAN UP AND SHIELD ACCESS ROUTES AND HOT SPOTS	(2/6)	1	2	2	2
MOVE REACTOR VESSEL INTERNALS TO REFUELING CAVITY	(2/1)	0	1	5	4
SEGMENT VESSEL INTERNALS AND LOAD CONTAINERS	(2/6)	1	4	4	4
SEGMENT PRESSURE VESSEL AND LOAD CONTAINERS	(2/5)	1	2	4	4
SEGMENT STEAM GENERATORS AND PREPARE FOR SHIPMENT	(2/3)	1	2	4	4
REMOVE REACTOR COOLANT SYSTEM PUMPS AND PIPING	(2/4)	1	1	4	4
REMOVE PRESSURIZER, RELIEF TANK, AND SAFETY INJECTION SYSTEM	(2/3)				
REMOVE HEAT EXCHANGERS AND CONTAINMENT AND CAVITY PUMPS	(2/2)	1	2	7	4
REMOVE CONTAMINATED INTERNAL STRUCTURES	(2/9)				
REMOVE SPRAY PIPING AND VENTILATION SYSTEMS	(2/2)	0	0	2	2
DECONTAMINATE INTERNAL SURFACES	(2/2.5)	1	2	7	1
FINAL RADIATION SURVEY	(2/1)[a]	0	0	0	0



MANPOWER LABOR GRADE[c]	TOTAL MAN-MONTHS	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30
		MAN MONTHS PER WORKING MONTH[b]																													
CREW LEADERS	71	2	4	4	4	4	4	2	2	2	2	2	2	2	2	4	2	2	4	2	2	2	2	2	2	2	2	2	2	1	0
UTILITY OPERATORS	160	6	12	12	12	12	12	8	4	4	4	4	4	4	4	6	2	2	6	4	4	4	4	4	4	4	4	4	4	2	0
LABORERS	347	14	12	12	12	12	12	8	8	8	8	8	8	8	8	16	8	8	22	14	14	14	14	14	14	14	18	18	14	7	0
CRAFTSMEN	261	12	12	12	12	12	12	8	8	8	8	8	8	8	8	16	8	8	16	8	8	8	8	8	8	8	12	6	2	1	0

(a) ONLY HEALTH PHYSICS PERSONNEL REQUIRED.

(b) MANPOWER REQUIREMENTS SHOWN ARE BASED ON LABOR REQUIRED TO COMPLETE TASKS AND DO NOT INCLUDE EXTRA MANPOWER NEEDED TO COMPLY WITH OCCUPATIONAL RADIATION DOSE LIMITS. SEE SECTION G.2.3.

(c) SHIFT ENGINEERS, HEALTH PHYSICS PERSONNEL, AND CRAFT SUPERVISORS NOT INCLUDED IN THIS TASK-WISE ASSESSMENT.

FIGURE G.2-1. DECON Task Schedule and Decommissioning Worker Requirements in the PWR Containment Building Following a Scenario 2 Accident somewhat from the values shown with accident severity, although this variance is judged to be within the range of about ±10% for the scenario 1 and scenario 3 accidents considered in this study.

G.2.2.2 Fuel Building

The post-accident DECON task schedule and sequence and associated decommissioning worker requirements in the fuel building are presented in Figure G.2-2, based on the corresponding information for DECON following normal shutdown contained in Section G.2.3 of Reference 1 and on the previous discussion of the fuel building in Secion G.2.1.2 of this appendix. No variability with accident scenario is anticipated, except that the timing of the final task is adjusted as appropriate to allow completion of the activities in the fuel building one month prior to the completion of activities in the containment building, the same as for DECON following normal shutdown.

G.2.2.3 Auxiliary Building

Because the requirements and task sequence in the auxiliary building during post-accident DECON are the same as those for DECON following normal shutdown, as shown in Figure G.2-4 of Reference 1, no detailed schedule is presented here for the auxiliary building. The timing of the final task in the auxiliary building is assumed to be adjusted, similar to that for the fuel building, so that completion of the activities in the auxiliary building occurs one month after the completion of the activities in the containment building. The decommissioning worker requirements for DECON activities in the auxiliary building are summarized in Table G.2-1, as are the decommissioning worker requirements for the other major DECON activities.

G.2.2.4 Overall Schedule and Decommissioning Worker Requirements

The overall schedule and sequence for DECON at the reference PWR, following a scenario 2 accident and the subsequent accident cleanup campaign, is shown in Figure G.2-3. The overall schedule includes the ancillary activities required to complete DECON, as described previously in Section G.2.1.4 of this appendix. As shown in the schedule, the overall project spans a period of 32-1/2 months. As discussed previously in Section G.2.2.1, the overall duration of the DECON project is not anticipated to vary substantially (more than

FUEL BUILDING: MONTHS AFTER ACCIDENT CLEANUP

MANPOWER PER SHIFT

TASK (SHIFTS PER DAY/DURATION IN MONTHS)		CREW LEADER	UTILITY OPERATOR	LABORER	CRAFTSMAN
CHEMICAL DECONTAMINATION OF CVCS AND PROCESS DECONTAMINATION SOLUTIONS	(3/1.5)	1	2	1	1
REMOVE CHEMICAL VOLUME CONTROL SYSTEM	(2/2.5)	1	1	2	1
REMOVE BORIC ACID SYSTEM	(2/1.2)	1	1	2	2
REMOVE CONCENTRATE HOLDING TANK SYSTEM	(1/0.8)	1	1	2	2
REMOVE SPENT FUEL STORAGE RACKS, NEW FUEL STORAGE RACKS, AND FUEL TRANSFER SYSTEM	(2/0.8)	1	1	2	2
REMOVE ACCIDENT CLEANUP DEMINERALIZER SYSTEM	(2/0.5)	1	1	2	4
REMOVE SPENT FUEL POOL CLEANUP SYSTEM, LINERS, AND CONTAMINATED CONCRETE	(2/1)	1	1	2	2
REMOVE CLOSED COOLING WATER SYSTEM	(2/1.1)	1	1	2	2
FINAL RADIATION SURVEY	(1/1)[a]	0	0	0	0

MANPOWER		MAN MONTHS PER WORKING MONTH[b]																												
LABOR GRADE[c]	TOTAL MAN-MONTHS	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29
CREW LEADER	19.5	3	2.5	2	2	2	0.5	0	0	1	0	0	0	0	1	2	1	2	0.5	0	0	0	0	0	0	0	0	0	0	0
UTILITY OPERATOR	24	6	4	2	2	2	0.5	0	0	1	0	0	0	0	1	2	1	2	0.5	0	0	0	0	0	0	0	0	0	0	0
LABORER	33	3	3.5	4	4	4	1	0	0	1.5	0	0	0	0	1.5	4	2	4	0.5	0	0	0	0	0	0	0	0	0	0	0
CRAFTSMAN	30	3	2.5	2	2	4	1	0	0	1.5	0	0	0	0	1.5	6	2	4	0.5	0	0	0	0	0	0	0	0	0	0	0

(a) ONLY HEALTH PHYSICS PERSONNEL REQUIRED.

(b) MANPOWER REQUIREMENTS SHOWN ARE ROUNDED TO THE NEAREST 0.5 MAN-MONTH PER MONTH. REQUIREMENTS ARE BASED ON NECESSARY LABOR TO COMPLETE TASKS AND DO NOT INCLUDE EXTRA MANPOWER NEEDED TO COMPLY WITH OCCUPATIONAL RADIATION DOSE LIMITS; SEE SECTION G.2.3.

(c) SHIFT ENGINEERS, HEALTH PHYSICS PERSONNEL, AND CRAFT SUPERVISORS NOT INCLUDED IN TASK-WISE ASSESSMENT.

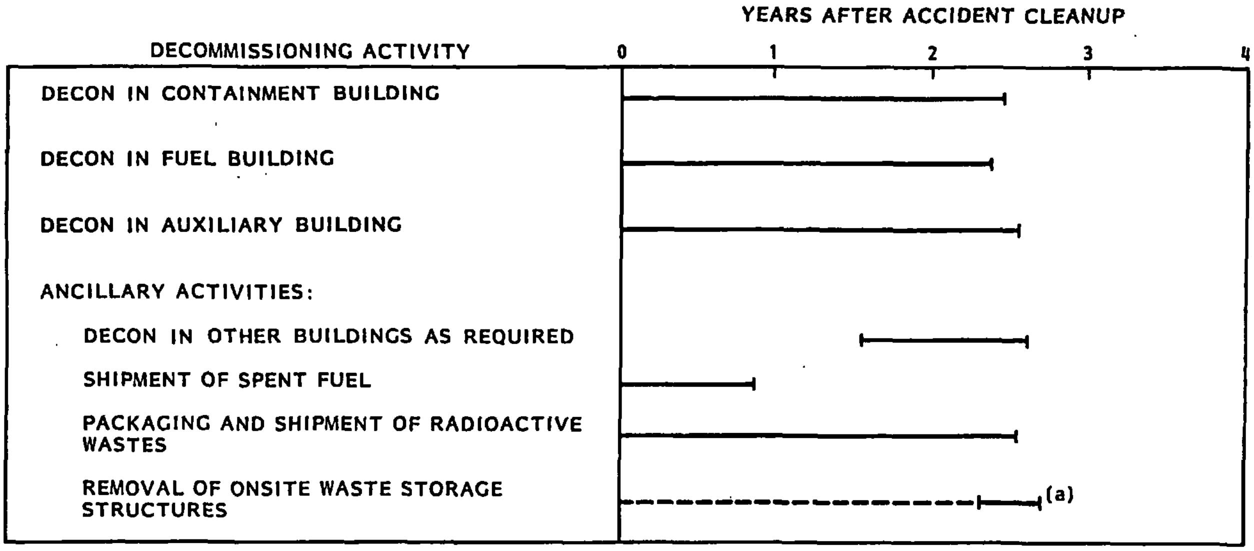
FIGURE G.2-2. DECON Task Schedule and Sequence and Decommissioning Worker Requirements in the PWR Fuel Building Following a Scenario 2 Accident

TABLE G.2-1. Summary of Direct Decommissioning Worker Requirements for DECON at the Reference PWR Following a Scenario 2 Accident[a]

DECON Activity Area	Estimated Decommissioning Worker Requirements (man-months) Crew Leaders	Utility Operators	Laborers	Craftsmen
Containment Building	71	160	347	261
Fuel Building	19.5	24	33	30
Auxiliary Building	20.5	0	33.5	56
Ancillaries				
Decontamination of other site buildings as required[b]	0	0	0	0
Packaging and shipping of radioactive wastes[c]	61	122	122	0
Decontamination of onsite waste storage structures	10	20	20	10
Totals	182	326	555.5	357

(a) Values shown include only labor required to actually perform tasks and do not include the extra labor needed to maintain compliance with occupational radiation dose limits. See Section G.2.3.
(b) Performed by subcontractor, only health physics support required from decommissioning staff.
(c) Includes shipping activities required to complete spent fuel shipment and removal of accident cleanup wastes from the onsite waste storage structures.

about ±1 month) with changes in accident severity, even though some of the individual DECON tasks require a greater level of effort to complete following a more-severe accident and, correspondingly, a reduced level of effort following a less-severe accident.



(a) BROKEN LINE INDICATES OFFSITE SHIPMENT OF STORED WASTES, AND SOLID LINE INDICATES DECONTAMINATION OF STRUCTURES.

FIGURE G.2-3. Duration of PWR DECON Activities Following a Scenario 2 Accident

Decommissioning worker requirements for the overall DECON project are given in Table G.2-1. These include only the labor required to actually complete the tasks and do not include the extra labor needed to maintain compliance with occupational radiation dose limits.

The packaging and shipping of radioactive wastes, including completion of the spent fuel shipment and of the removal and shipment of accident cleanup wastes from the onsite waste storage structures, is assumed to be handled by crews consisting of one crew leader, two utility operators, and two laborers. These crews are assumed to work two shifts per day, 5 days each week, for a total duration of 30-1/2 months.

Decontamination of the onsite waste storage structures, once the stored waste is removed, is estimated to require 5 months to complete assuming two shifts per day, 5 days each week, with each crew consisting of one crew leader, two utility operators, two laborers, and a craftsmen.

G.2.3 External Occupational Radiation Doses

Estimates are made of the external occupational radiation doses that are accumulated by the decommissioning workers during post-accident DECON at the reference PWR. The estimates are based on the estimated exposures (i.e., man-hours of effort required in radiation zone work) and the anticipated dose rates associated with each activity area for all labor categories. Exposure information developed in Reference 1 for DECON following normal shutdown is used where applicable. The estimated average dose rates in the reference PWR following accident cleanup, shown in Table E.4-1, provide additional input.

Basic assumptions used in developing the dose estimates are:

- Every effort is made to minimize personnel exposure to radiation (ALARA philosophy) while accomplishing a task by the use of temporary shielding and remote handling techniques and by keeping workers not actively engaged in the work out of the radiation fields.
- Chemical decontamination efforts are reasonably successful, reducing all radiation dose rates from the decontaminated piping and equipment by at least a factor of 10.
- 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.

No correction for radioactive decay is calculated for the occupational doses because the decay of radioactivity in the plant is governed by ^{137}Cs, with a 30-year half-life, so that any such correction would be quite small, within the limits of error on the dose estimates.

The estimated external occupational radiation doses for DECON at the reference PWR following a postulated scenario 2 accident and the subsequent accident cleanup campaign are presented in Table G.2-2. As shown in the table, the total occupational dose is estimated to be over 3060 man-rem, and the largest contributor to the total dose is the decontamination of the containment building.

Based on the dose rate information presented in Table E.4-1 and assuming some limited variation in manpower requirements with accident severity, the

TABLE G.2-2. Estimated Occupational Radiation Doses for DECON at the Reference PWR Following a Scenario 2 Accident

DECON Activity Area	Average Dose Rate (rem/hr)	Supervisors[a] Exposure (man-hr)	Supervisors[a] Dose (man-rem)	Utility Operators and Laborers Exposure (man-hr)	Utility Operators and Laborers Dose (man-rem)	Craftsmen Exposure (man-hr)	Craftsmen Dose (man-rem)	Health Physics Technicians Exposure (man-hr)	Health Physics Technicians Dose (man-rem)	Task Totals Exposure (man-hr)	Task Totals Dose (man-rem)
Containment Building											
Operating Floor Level	0.010	3 559	35.59	14 318	143.18	6 240	62.40	4 691	46.91	28 808	288.08
Mezzanine Level	0.020	5 076	101.52	18 307	366.14	10 014	200.28	5 364	107.28	38 761	775.22
Ground Floor Level	0.050	3 601	180.05	15 028	751.40	7 701	385.05	4 007	200.35	30 337	1 516.85
Subtotals		12 236	317.16	47 653	1 260.72	23 955	647.73	14 062	354.54	97 906	2 580.15
Fuel Building											
All Levels	0.005	658	3.29	2 384	11.92	1 303	6.52	826	4.13	5 171	25.86
Auxiliary Building											
All Levels	0.010	1 687	16.87	2 395	23.95	3 776	37.76	1 747	17.47	9 605	96.05
Ancillaries											
Packaging and Shipping of Radioactive Wastes:											
• Spent Fuel	0.010	567	5.67	9 [illegible]	90.72	--[b]	--	1 701	17.01	11 340	113.40
• DECON Wastes	0.010	2 050	20.50	8 198	81.98	--	--	2 050	20.50	12 298	122.98
• Accident Cleanup Wastes[c]	0.010	512	5.12	2 050	[illegible]	512	5.12	512	5.12	3 586	35.86
Decontamination of Onsite Waste Storage Structures	0.005	840	4.20	5 040	25.20	[illegible]60	6.30	1 260	6.30	8 400	42.00
Radiation Surveys (Weekly)	0.010	--	--	--	--	--	--	4 676	46.76	4 676	46.76
Subtotals		3 969	35.49	24 360	218.40	1 772	11.42	10 199	95.69	40 300	361.00
Totals		18 550	372.81	76 792	1 514.99	30 806	703.43	26 834	471.83	152 982	3 063.06

(a) Includes shift engineers, crew leaders, craft supervisors, and senior health physics technicians.
(b) A dash indicates that, for the specified activity area, that particular staff category is not used.
(c) Removal of accident cleanup wastes remaining in the onsite waste storage structures at the time DECON commences.

total occupational radiation doses for DECON following accident cleanup after a scenario 1 accident could be a factor of 3 lower than those following accident cleanup after the scenario 2 accident, while the total doses for DECON following accident cleanup after a scenario 3 accident could be 2 or 3 times greater than those for DECON following cleanup after a scenario 2 accident.

Whole-body doses to the decommissioning workers must be limited in accordance with 10 CFR 20.101. Because the decommissioning is preceded by accident cleanup work that is assumed to use the same labor pool, all of the decommissioning workers are assumed to be long-time radiation workers whose annual doses are limited to 5 man-rem/man-year. To determine if sufficient manpower is postulated to comply with the dose limit or if the estimated manpower requirements must be adjusted upwards, the dose estimates presented in Table G.2-2 for the decommissioning workers are compared to the manpower requirements developed in Section G.2. The results of this comparison for DECON following a scenario 2 accident are presented in Table G.2-3. In all cases, the decommissioning worker requirements must be adjusted upward to maintain compliance with the 5 man-rem/man-year limit. The adjusted worker requirements in man-years that are summarized in the table are used in determining the staff labor costs for DECON (presented in Appendix H). Because of the variations in the radiation doses with accident severity, discussed previously, the adjusted worker requirements are expected to be about a factor of 3 lower following a scenario 1 accident and about a factor of 2.5 higher following a scenario 3 accident. The overall impact of this on the staff labor requirements for DECON is discussed in Section G.2.4.

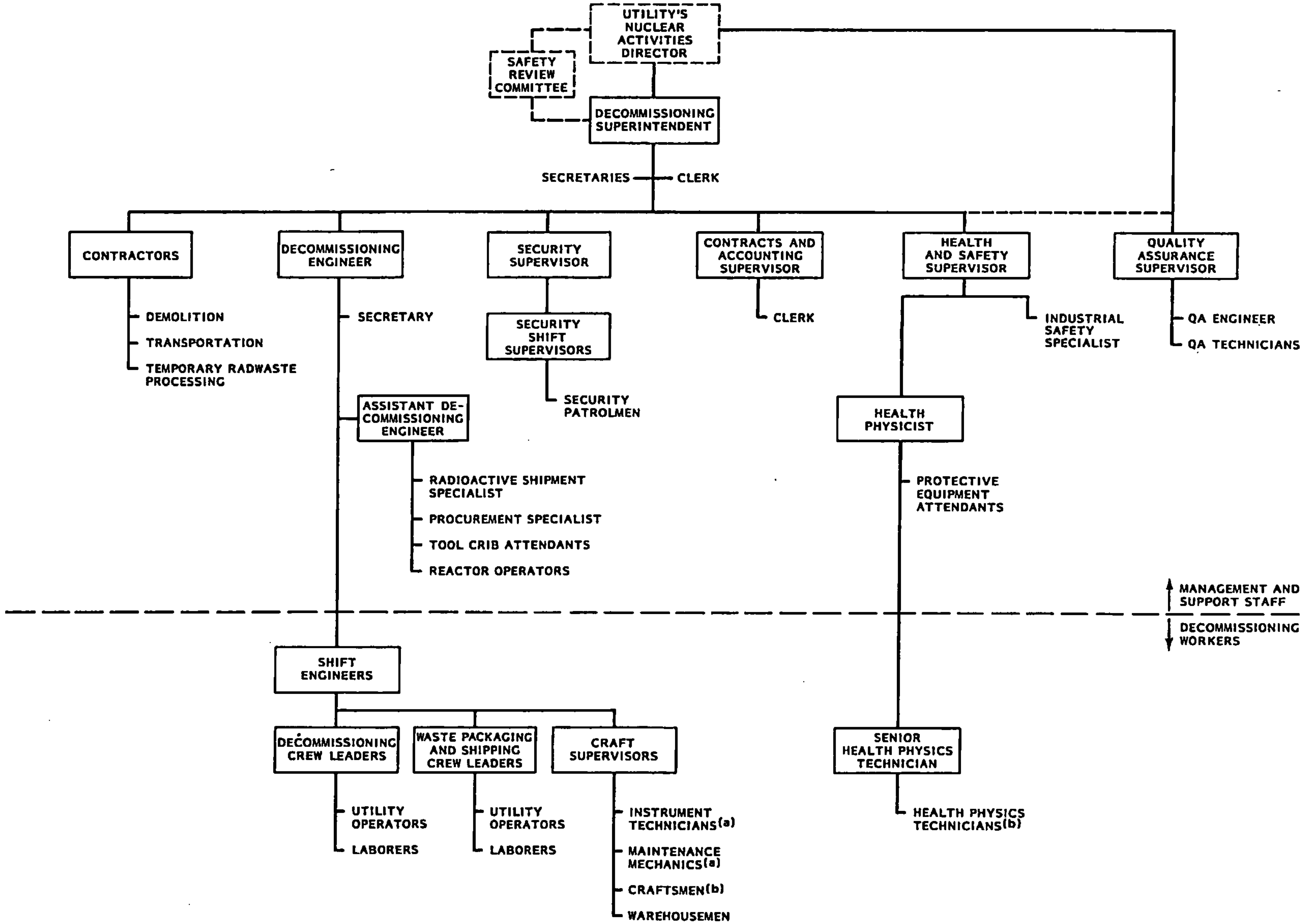
G.2.4 Overall Staff Labor Requirements

The staff organization postulated for post-accident DECON at the reference PWR is shown in Figure G.2-4. Five parallel branches report to a decommissioning superintendent. The operational branch, under a decommissioning engineer, plans and performs the actual decommissioning tasks. The safety branch, under a health and safety supervisor, plans and conducts both radiological and industrial safety programs. The three auxiliary branches handle security, financial, and quality assurance matters. Further discussion pertaining to the staff organization and the functions of key staff members can be found in Chapter 9 of Reference 1.

TABLE G.2-3. Adjustments to Decommissioning Worker Requirements to Comply with Occupational Radiation Dose Limitations for DECON at the Reference PWR Following a Scenario 2 Accident

Decommissioning Worker Category	Estimated Worker Requirements[a] (man-yr)	Estimated Occupational Dose: Total[b] (man-rem)	Estimated Occupational Dose: Individual Average (man-rem/man-yr)	Adjustment Factor[c]	Adjusted Worker Requirements (man-yr)
Crew Leaders	15.17	223.69[d]	14.75	3.0	45.51
Utility Operators and Laborers	73.46	1514.99	20.62	4.2	308.53
Craftsmen	29.75	703.43	23.64	4.8	142.80
Health Physics Technicians[e]	--	471.83	--	--	94.37

(a) Based on Table G.2-1
(b) Based on Table G.2-2
(c) Increase required in worker requirements to reduce average individual dose to $\leq$5 man-rem/man-yr.
(d) Assumes that crew leaders account for 60% of supervisory dose and that the remaining 40% accrues to shift engineers, craft supervisors, and senior health physics technicians.
(e) Although requirements for health physics technicians are not included in Table G.2-1, the number required can be estimated on the basis of the occupational radiation dose limitations, which is the controlling factor in the overall requirements for all of the direct decommissioning workers.



(a) CARRY OUT MAINTENANCE OF ESSENTIAL SYSTEMS AND SERVICES
(b) ASSIGNED TO WORK CREWS AS REQUIRED

FIGURE G.2-4. Postulated Staff Organization for Post-Accident PWR Decommissioning by DECON

The total staff labor requirements for DECON at the reference PWR following a scenario 2 accident are given in Table G.2-4. These requirements are given in equivalent man-years for the planning and preparation phase of the decommissioning (spanning the last 1-1/2 years prior to completion of accident cleanup) as well as for the actual DECON (starting at the time of completion of the cleanup campaign). The requirements presented include the management and support staff as well as the decommissioning workers. The decommissioning worker requirements are based on the adjusted requirements needed to comply with occupational radiation dose limitations, presented previously in Table G.2-3. In addition, the following assumptions are used in developing the information presented in the table:

- The planning and preparation phase requires only the decommissioning superintendent, decommissioning engineer, assistant decommissioning engineer, and secretarial and clerical support. All other planning and preparation functions can be carried out part-time by members of the accident cleanup staff.
- Decommissioning worker supervision is staffed on a two-shift-per-day, 5-day-week basis with one shift engineer, two craft supervisors, and three senior health physics technicians on each shift.
- Any special contract services personnel required to augment the local labor pool to meet staffing requirements are assumed to have the same salary costs in each job category as the normal staff.

These assumptions are generally consistent with those used in References 1 and 4 to develop staffing requirements for DECON following normal plant shutdown.

About 790 man-years of effort are estimated for DECON at the reference PWR following a scenario 2 accident and the subsequent accident cleanup campaign. This includes about 170 man-years for management and support staff and about 620 man-years for the decommissioning workers. The management and support staff requirements are essentially time-dependent and, therefore, are not anticipated to vary substantially with accident severity because the duration of the DECON project varies little from accident scenario to accident

TABLE G.2-4. 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.

scenario (see Section G.2.2). However, the decommissioning worker requirements could vary from about 210 man-years following a scenario 1 accident to about 1560 man-years following a scenario 3 accident, based on the occupational radiation dose limitations discussed previously in Section G.2.3. Thus, the total staff labor requirements following a scenario 1 or scenario 3 accident are anticipated 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 shutdown are about 300 man-years; about 170 man-years for management and support staff and about 130 man-years for the decommissioning workers.[1]

G.3 SAFSTOR AT THE REFERENCE PWR

SAFSTOR comprises those activities required to prepare and maintain the facility so that the risk to public safety is acceptable and the property can be safely stored for a period of time to allow decay of some of the onsite radioactivity, until the eventual (deferred) decontamination of the facility to unrestricted release levels.[a] SAFSTOR consists of: 1) a period of facility and site preparation (preparations for safe storage) that includes offsite shipment of the reactor fuel and concentration and immobilization of dispersible radioactive materials, 2) an interim period of continuing care (safe storage) that includes 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, allowing possession but not operation of the facility, remains in force throughout the safe storage period, since onsite radioactivity remains above unrestricted release levels.

The duration of the safe storage period is undefined. Periods of up to about 100 years are consistent with recommended EPA policy on institutional

---

(a) Deferred decontamination of the facility is not required if the radioactivity decays to unrestricted release levels during the storage period; however, such decay is unlikely within reasonable periods of post-accident storage at an LWR, even following accident cleanup.

control reliance for radioactivity containment.[5] The shorter the safe storage period is, the less the onsite radioactivity decays and, thus, the less advantageous SAFSTOR is as a decommissioning alternative.

SAFSTOR satisfies the requirements for protection of the public while reducing, as compared to DECON, initial commitments of money, occupational radiation dose, and waste disposal space. However, these reduced initial commitments (realized during preparations for safe storage) are offset somewhat by the need for continuing care (the safe storage period) and the eventual deferred decontamination. Furthermore, it should be noted that for post-accident SAFSTOR the decay of the radioactive contamination within the stored facility is considerably slower than for SAFSTOR following normal shutdown, because the decay of the post-accident radionuclide inventory is controlled by ^{137}Cs (with a half-life of $\sim$30 years) rather than by ^{60}Co (with a half-life of 5.27 years). In addition, deferral of decontamination to the end of the safe storage period has the disadvantage that personnel familiar with the facility are not likely to be available to staff the final phase of SAFSTOR.

Decommissioning by SAFSTOR might be desirable if adequate disposal space for decommissioning wastes is not available at shallow-land burial sites. SAFSTOR could also be used to provide for interim onsite storage of spent fuel or of highly radioactive or long-lived wastes. However, it is unlikely that most reactor sites could qualify as permanent waste repositories and, thus, onsite waste storage would ultimately be followed by deferred decontamination of the facility and site. (See Chapter 15 of Volume 1 for an analysis of the impacts of alternate scenarios for waste disposal.)

Details of post-accident SAFSTOR at the reference PWR are discussed in this section, including methods, schedules and manpower requirements, and external occupational radiation doses. These details are based primarily on the analysis of SAFSTOR at the reference PWR following normal shutdown, presented in Appendix H and Chapter 9 of Reference 1, because, after accident cleanup is completed, most of the requirements for SAFSTOR are similar whether or not the reactor has experienced an accident. Where the postulated accident results in significant changes in the requirements for SAFSTOR, the differences are identified and new information is developed to support the analysis. The analysis presented is for SAFSTOR following a scenario 2 accident; variations in SAFSTOR requirements with changes in the severity of the accident are discussed.

The information in this section forms the basis for the estimates of costs and safety impacts of post-accident SAFSTOR at the reference PWR that are developed in Appendices H and J, respectively, and for the comparison of post-accident and normal-shutdown SAFSTOR that is presented in Chapter 17 of Volume 1.

G.3.1 Methods for Preparations for Safe Storage

The methods used for post-accident preparation of the reference PWR for safe storage are generally the same as those used following normal shutdown, described in detail in Reference 1. The methods for post-accident preparation for safe storage are summarized here.

As with DECON, the preparations for the safe storage 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 are available and as the various systems involved complete their required service functions.

In general, the methods used 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.

Many of the actual procedures used are discussed in Appendix D of this study. Further details are provided in Chapter 9 of Reference 1, Appendix U of Reference 3, and Appendix J of Reference 4. The categories are discussed briefly in the following paragraphs.

The particular procedure used to decontaminate, deactivate, and seal each system or piece of equipment is identified during the planning phase of decommissioning (concurrent with the completion of the accident cleanup campaign). Portions of the facility containing significant amounts of radioactivity are isolated by tamper-proof barriers, with all indirect access routes, however unlikely, identified and sealed. Vents with HEPA filters are installed in the HVAC systems servicing these sealed areas to allow for temperature and pressure changes, and the systems are deactivated. Contaminated drains are decontaminated and building sumps are decontaminated and secured.

After the loose, readily removable contamination is removed from the surfaces of plant structures and equipment, the residual surface contamination is fixed in place. Spray painting is the selected method of fixation in this study. Wherever possible, all contaminated exterior and interior surfaces are sprayed to prevent contamination spread during either preparations for safe storage or the subsequent continuing care. Part of the continuing care is to monitor painted areas for deterioration and to recoat them as necessary.

Unsalvageable contaminated equipment and other noncombustible radioactive materials may be transferred within the plant from areas being decontaminated to other secured storage areas. Transferred items are spray painted to fix contamination, as are surfaces exposed by removal of the items. The equipment and ductwork remaining in the work area is decontaminated and spray painted.

The 13-point procedure postulated to be used to prepare contaminated areas throughout the major plant structures for safe storage is as follows:

1. Evaluate initial radiological conditions.
2. Vacuum interior surface areas.
3. Deactivate nonessential systems and equipment.
4. Clean interior surface areas and exposed surfaces of equipment and piping.
5. Clean remaining hot spots.
6. Apply protective paint.

7. Transfer contaminated equipment and materials, where appropriate.

8. Decontaminate and seal vent systems.

9. Install HEPA-filtered vents.

10. Deactivate remaining nonessential systems and equipment.

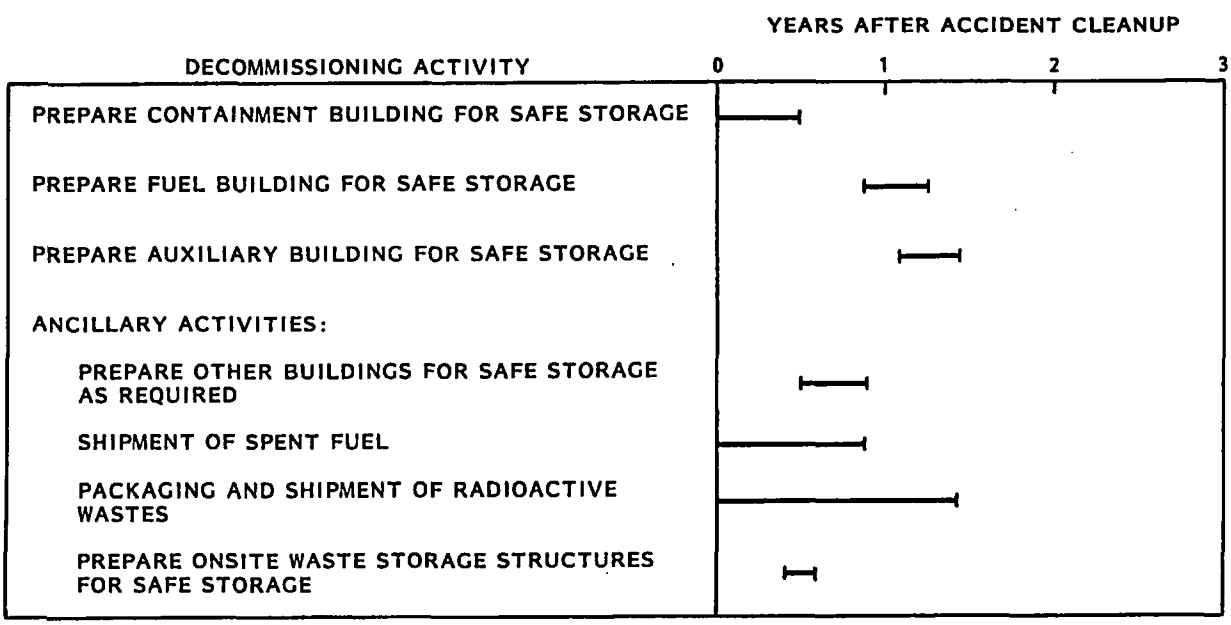
11. Install intrusion, fire, and radiation detection systems as necessary and provide for servicing and offsite readout.

12. Conduct final radiation survey.

13. Secure the structure.

G.3.2 <u>Schedule</u>

The overall schedule and sequence for preparation of the reference PWR for safe storage, following a scenario 2 accident and the subsequent accident cleanup campaign, is shown in Figure G.3-1. As shown in the figure, this initial phase of SAFSTOR is postulated to span a period of 17 months. The



<u>FIGURE G.3-1</u>. Duration of PWR Preparations for Safe Storage Following a Scenario 2 Accident

overall duration of the project is not judged to vary significantly with accident severity, although the level of effort required for some individual tasks within the schedule may vary. In general, activities other than surface decontamination are anticipated to require about the same level of effort as for SAFSTOR following normal shutdown. Furthermore, the decontamination efforts are aimed not at removal of contamination to unrestricted release levels but at removal of all loose, readily removable contamination. The effort anticipated to be required for this type of decontamination is not expected to vary much from accident scenario to accident scenario. However, because higher initial levels of contamination are present following more severe accidents, the levels of residual contamination remaining in the plant following preparations for safe storage are anticipated to increase with increasing accident severity.

Decommissioning worker requirements for preparations for safe storage are not identified here but rather are calculated in the next subsection on the basis of the radiation dose limitations to the workers. This approach is taken because it is demonstrated in Section G.2.3 that dose limitations are the controlling factor for DECON staffing requirements, and the same situation prevails for the preparations for safe storage phase of SAFSTOR.

In the containment building, the requirement for chemical decontamination of the reactor coolant system is eliminated because this task is completed during the accident cleanup campaign (see Table G.1-2, presented previously). The duration of activities in the containment building is thus reduced to approximately 6 months.

Activities in the fuel building begin after completion of the spent fuel shipment and, including chemical decontamination of the CVCS and removal of the accident-water cleanup demineralizer system, span a period of about 4-1/2 months. Work in the auxiliary building overlaps work in the fuel building and takes about 4 months to complete.

The onsite waste storage structures and other buildings on the site are prepared for safe storage between the work in the containment building and the subsequent work in the fuel and auxiliary buildings. The packaging and shipment of those wastes not selected to be stored onsite during the safe storage period, particularly the combustible wastes generated by the decommissioning workers in carrying out their tasks, spans the entire 17-month schedule.

G.3.3 Occupational Radiation Doses and Decommissioning Worker Requirements

The estimates presented here of the external occupational radiation doses that are accumulated by the decommissioning workers during post-accident preparations for safe storage at the reference PWR are based on the estimates for normal-shutdown preparations for safe storage presented in Appendix H of Reference 1 and on the estimates for post-accident DECON presented previously in Section G.2.3 of this appendix. The estimated average dose rates in the reference PWR following accident cleanup, shown in Table E.4-1 of this study, provide the dose rate values. The major assumptions used in developing to exposure information are as follows:

- Exposure times in the containment building are the same as for normal-shutdown SAFSTOR except that the predecommissioning survey and the chemical decontamination of the reactor coolant system occur during accident cleanup, thus eliminating the associated exposure times. Exposure times within containment are distributed between the levels of the building in the same proportions as the DECON exposure times.
- In the fuel building, exposure times for the predecommissioning survey are deleted. For chemical decontamination of the CVCS and removal of the accident-water cleanup demineralizer system, exposure times are the same as for post-accident DECON.
- Auxiliary building exposure times for normal-shutdown SAFSTOR are used with deletion of the predecommissioning survey.
- For the packaging and shipping of combustible wastes, the same waste crews as for DECON are assumed, over a span of 17 months.
- Preparations for safe storage of the onsite waste storage structures is postulated to require 2 months to complete using staffing equivalent to that for DECON in the structures.

- The weekly radiation surveys are assumed to require four times the level of effort for normal-shutdown SAFSTOR, prorated for the duration of the entire project.

The estimated external occupational radiation doses for preparations for safe storage at the reference PWR, following a scenario 2 accident and accident cleanup campaign, are presented in Table G.3-1. As shown in the table, the total occupational dose is estimated to be almost 430 man-rem. The ancillary activities account for about half of this, with containment building doses dominating the remainder.

Based on the dose rate information presented in Table E.4-1 and assuming some small variation in manpower requirements with accident severity, the total occupational doses for preparations for safe storage following a scenario 1 accident would be more than 40% less than those following a scenario 2 accident, while the total doses following a scenario 3 accident would be more than twice those following a scenario 2 accident.

For post-accident DECON at the reference PWR, the total decommissioning worker requirements were determined not by the numbers needed to efficiently perform the work but by the radiation dose limitations on the individual workers (see Section G.2.3). This is also the case for preparations for safe storage. Based on an annual dose limit to individuals of 5 man-rem/man-year, the estimated total decommissioning worker requirements for preparations for safe storage following the scenario 2 accident are shown in Table G.3-2 by worker category. Because the requirements are based on radiation dose, they are anticipated to vary with accident severity in the same manner as the total occupational doses (see previous paragraph).

G.3.4 Overall Staff Labor Requirements

The staff organization postulated for post-accident preparations for safe storage at the reference PWR is the same as that for post-accident DECON, shown previously in Figure G.2-4. Further discussion pertaining to staff organization and the functions of key staff members can be found in Chapter 9 of Reference 1.

TABLE G.3-1. Estimated Occupational Radiation Doses for Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident

SAFSTOR Activity Area	Average Dose Rate (rem/hr)	Supervisors[a] Exposure (man-hr)	Supervisors[a] Dose (man-rem)	Utility Operators and Laborers Exposure (man-hr)	Utility Operators and Laborers Dose (man-rem)	Craftsmen Exposure (man-hr)	Craftsmen Dose (man-rem)	Health Physics Technicians Exposure (man-hr)	Health Physics Technicians Dose (man-rem)	Task Totals Exposure (man-hr)	Task Totals Dose (man-rem)
Containment Building											
Operating Floor Level	0.010	347	3.47	717	7.17	349	3.49	545	5.45	1 958	19.58
Mezzanine Level	0.020	495	9.90	917	18.34	561	11.22	623	12.46	2 596	51.92
Ground Floor Level	0.050	351	17.55	752	37.60	431	21.55	465	23.25	1 999	99.95
Subtotals		1 193	30.92	2 386	63.11	1 341	36.26	1 633	41.16	6 553	171.45
Fuel Building											
All Levels	0.005	99	0.50	905	4.53	540	2.70	530	2.65	2 074	10.38
Auxiliary Building											
All Levels	0.010	229	2.29	397	3.97	279	2.79	1 059	10.59	1 964	19.64
Ancillaries											
Packaging and Shipping of Radioactve Wastes:											
• Spent Fuel	0.010	567	5.67	9 072	90.72	--[b]	--	1 701	17.01	11 340	113.40
• Combustible Wastes	0.010	1 428	14.28	5 712	57.12	--	--	1 428	14.28	8 568	85.68
Preparations for Safe Storage of Onsite Waste Storage Structures	0.005	336	1.68	2 016	10.08	504	2.52	504	2.52	3 360	16.80
Radiation Surveys (Weekly)	0.010	--	--	--	--	--	--	1 122	11.22	1 122	11.22
Subtotals		2 331	21.63	16 800	157.92	504	2.52	4 755	45.03	24 390	227.10
Totals		3 852	55.34	20 488	229.53	2 664	44.27	7 977	99.43	34 981	428.57

(a) Includes shift engineers, crew leaders, craft supervisors, and senior health physics technicians.
(b) A dash indicates that, for the specified activity area, that particular staff category is not used.

TABLE G.3-2. Decommissioning Worker Requirements for Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident

Decommissioning Worker Category	Estimated Total Occupational Dose[a] (man-rem)	Decommissioning Worker Requirements[b] (man-years)
Crew Leaders	33.20[c]	6.7
Utility Operators	122.10[d]	24.5
Laborers	107.43[d]	21.5
Craftsmen	44.27	8.9
Health Physics Technicians	99.43	19.9

(a) Based on Table G.3-1.
(b) Minimum number required to maintain average individual dose ≤5 man-rem/man-year.
(c) Assumes that crew leaders account for 60% of supervisory dose and that the remaining 40% accrues to shift engineers, craft supervisors, and senior health physics technicians.
(d) Assumes that ratio of utility operator to laborer time is the same as for normal-shutdown SAFSTOR (1.14:1, see Table 10.2-2 of Reference 1).

The total staff labor requirements for preparations for safe storage following a scenario 2 accident are given in Table G.3-3. These requirements are given in equivalent man-years for the planning and preparation phase as well as for the preparations for safe storage. Both decommissioning workers and management and support staff are included. The basic assumptions used in developing the requirements are the same as those for DECON, presented previously in Section G.2.4.

Over 190 man-years of effort are estimated for preparations for safe storage at the reference PWR, following a scenario 2 accident and the subsequent accident cleanup campaign. Included are over 90 man-years for management and support staff and almost 100 man-years for the decommissioning workers. Because management and support staff requirements are primarily dependent on project duration and because the overall duration of the preparations for safe storage is not anticipated to vary significantly with accident severity, only minimal variations in the management and support staff requirements are likely with changes in accident severity. However, because

TABLE G.3-3. 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]	1.3
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.

The information in this section forms the basis for the estimates of costs and safety impacts of post-accident ENTOMB developed in Appendices H and J, respectively, and for the comparison of post-accident and normal-shutdown ENTOMB that is presented in Chapter 17 of Volume 1.

G.4.1 Entombment Methods

In this study, it is assumed that construction of the entombment structure should make use of existing plant features to the maximum extent possible, to avoid excessive modifications during the decommissioning process. This reduces the time required for ENTOMB and minimizes costs. For the reference PWR, 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 barrier walls surrounding the entombment area are sealed and, after emplacement of the waste to be entombed, the top is also sealed to complete the structure. The remaining portions of the reference PWR outside of the postulated entombment structure area are not judged to be suitable for entombing because of the limited structural strength of these areas (resulting in the need for substantial modifications during the ENTOMB process) and because of the more numerous, larger building penetrations that would require sealing.[a] These remaining areas, including the fuel and auxiliary buildings in their entirety, are decontaminated in the same manner as for DECON (see Section G.2). Likewise, ENTOMB by total submersion of plant equipment in concrete is not considered because of the severe logistical problems and extreme expense involved in such a project. The upper portion of the PWR containment dome is assumed to be completely decontaminated and 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 for periodic inspections and maintenance.

(a) Entombment is considered for the caisson storage facility and canyon storage facility constructed during the accident cleanup campaign for onsite storage of cleanup-generated wastes, because these facilities are constructed in a manner compatible with entombment.

levels in the facility following an accident, even after substantial accident cleanup efforts, are significantly higher than after normal reactor shutdown, and 2) the post-accident radionuclide inventory decays more slowly than the normal-shutdown inventory because of the presence of longer-lived radionuclides (i.e., ^{90}Sr and ^{137}Cs) released by the accident (see Chapter 8 and Appendix C). ENTOMB is similar to SAFSTOR in that it consists of a period of facility and site preparations followed by a period of continuing care that includes security, surveillance, and maintenance activities, although the continuing care requirements for ENTOMB are less stringent than those for SAFSTOR. Under existing regulations, the nuclear license must remain in force until either the entombed radioactivity decays to unrestricted release levels or the entombment structure is dismantled and the entombed radioactivity removed. Dismantling the entombment structure is much more difficult than dismantling the unentombed facility, since the entombment structure is intended to endure for a long period of time under any credible conditions. 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, with a corresponding commitment to long-term maintenance of the nuclear license.

Details of post-accident ENTOMB at the reference PWR are discussed in this section, including entombment methods, schedules and manpower requirements, and external occupational radiation doses. These details are based largely on the analysis of ENTOMB at the reference PWR following normal shutdown, presented in Chapter 4 of Reference 2, because, after cleanup is completed, most of the requirements for ENTOMB are similar whether or not the reactor has experienced an accident. Where the postulated accident results in significant changes, the differences in the requirements are identified and new information is developed. The analysis of post-accident ENTOMB also makes use of information developed previously in Section G.2 of this appendix concerning post-accident DECON because, for areas outside of the entombment structure, ENTOMB and DECON activities are essentially identical.

The analysis presented in this section is based on the assumption that the reactor has experienced a scenario 2 accident. Variations in the ENTOMB requirements with changes in accident severity are discussed.

decommissioning worker requirements are based on occupational radiation dose limits (see Section G.3.3), these could vary from about 60 man-years following a scenario 1 accident to over 200 man-years following a scenario 3 accident. Therefore, the total staff labor requirements following a scenario 1 or scenario 3 accident are anticipated to be about 155 man-years or about 295 man-years, respectively. The post-accident staff labor requirements for the initial phase of SAFSTOR (i.e., preparations for safe storage) are considerably lower than the corresponding requirements for either DECON or ENTOMB, primarily because of the lower occupational radiation doses accumulated in carrying out the required decommissioning tasks.

G.3.5 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. A post-accident safe storage period of less than about 30 years allows little decay of the stored radioactivity (because the decay is controlled by ^{137}Cs with a 30-year half-life) and therefore is not considered to provide any real advantage over DECON. Furthermore, periods of over about 100 years appear to be inconsistent with recommended EPA policy on institutional control reliance for radioactivity containment.[5] Therefore, two potential safe storage periods are considered in this study: 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 for post-accident safe storage at the reference PWR is anticipated to be approximately the same as the level of effort required following normal shutdown. From Table H.4-4 of Reference 1, the annual labor requirement is estimated to be less than 2 man-year/year. Thus, the total cumulative labor requirement for the 30-year or 100-year safe storage period is conservatively estimated to be 60 man-years or 200 man-years, respectively.

The radiation dose to the safe storage workers in the first year of safe storage following normal shutdown is estimated in Table H.4-4 of Reference 1 to be less than 1.87 man-rem. However, dose rates during preparations for safe storage following a scenario 2 accident are about 3 times higher than those following normal shutdown. Thus, the first-year dose for post-accident safe storage is estimated to be less than about 5.6 man-rem. The total accumulated occupational radiation doses for post-accident safe storage periods of 30 and 100 years, with the radioactive decay controlled by ^{137}Cs, are calculated to be over 120 man-rem and almost 225 man-rem, respectively.

The same basic activities that are performed during DECON are also performed during deferred decontamination. Thus, 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 G.2 for post-accident DECON are controlled by radiation dose limitations to the workers. Thus, these decommissioning worker requirements are anticipated to be reduced by about 50% by 30-year safe storage, and would probably be reduced by about 75% following 100-year safe storage, based on the manpower adjustment factors shown previously in Table G.2-3.

The total occupational radiation doses for deferred decontamination are estimated based on the doses for DECON following a scenario 2 accident and the calculated decay of the residual radioactivity in the plant. The total doses for deferred decontamination following 30-year and 100-year safe storage periods are thus anticipated to be about 50% and 10%, respectively, of the DECON doses, or about 1500 man-rem and 300 man-rem.

G.4 ENTOMB AT THE REFERENCE PWR

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 release level. ENTOMB is generally intended for use where the residual radioactivity will decay to levels permitting unrestricted release of the facility within reasonable time periods; recommended EPA policy on institutional control reliance for radioactivity containment suggests that the entombing period not exceed approximately 100 years.[5] Decommissioning a power reactor by ENTOMB following a reactor accident appears to be less acceptable than following normal shutdown because: 1) residual radioactivity

With the exception of the reactor vessel internals, which are segmented and packaged for offsite disposal, the radioactive materials originating within the entombment structure are entombed onsite, together with as much as possible of the radioactive equipment and structural material from the rest of the plant.[2] ENTOMB does not eliminate the need for offsite disposal of decommissioning wastes but only reduces that need. From Section 4.8 of Reference 2, the estimated available volume within the entombment structure area for placement of radioactive materials from other portions of the plant is about 14,500 m^3. Because of the variety of shapes and sizes of both the volume available within the entombment structure and the contaminated materials to be stored there, as well as the difficulty in placing materials in some portions of the structure, a volume utilization efficiency of 50% is assumed (see Section U.5.2 of Reference 3 and Section K.1.3 of Reference 4). Therefore, up to 7,250 m^3 of the radioactive waste material originating outside the entombment structure can be entombed with the material originating inside the structure.

The entombment methods used for post-accident ENTOMB at the reference PWR are generally the same as those for ENTOMB following normal shutdown, described in Chapter 4 of Reference 2. In addition, the ENTOMB activities outside of the entombment structure area are generally the same as the DECON activities for those areas, discussed previously in Section G.2 of this appendix. The methods used for post-accident ENTOMB are summarized here, with appropriate details included where these are significantly different from normal-shutdown ENTOMB and post-accident DECON methods.

As with DECON, ENTOMB begins in the containment building, which represents 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.

G.4.1.1 <u>Containment Building</u>

As discussed previously, the major ENTOMB activities take place in the containment building. Furthermore, the major differences between ENTOMB and

DECON involve the containment building, because the postulated entombment structure area is in the containment building and the plant areas outside of the entombment structure area are decommissioned very similarly for both DECON and ENTOMB.

ENTOMB requires removal of all radioactive materials exceeding unrestricted release levels outside of the entombment structure area. Therefore, ENTOMB activities in the containment building outside of the entombment structure area are the same as the corresponding DECON activities (described previously in Section G.2.1.1), except that only some of the resulting radioactive waste materials require packaging for offsite shipment and disposal and the remaining waste materials are placed inside the entombment structure. Thus, the new activities in the containment building for ENTOMB center on preparation of the selected area for use as the entombment structure, placement of the radioactive materials to be entombed in the structure, and sealing of the structure once the material placement activities are completed. The reactor vessel internals are removed, segmented, and packaged for offsite disposal, the same as for DECON, at the beginning of the ENTOMB effort.

Several activities are required to prepare the selected area to serve as the entombment structure and to receive the radioactive materials to be entombed. Piping that penetrates the postulated entombment structure is cut off at all points of penetration and the openings are sealed with welded steel plates; the sealed piping sections embedded in the concrete walls are then filled with cast-in-place reinforced concrete to provide a continuous concrete exterior for the entombment structure. Selected sections of piping and conduit within the entombment structure area are cut and removed from position to improve access and facilitate movement into the structure of the materials to be entombed. Because the steam generators and the pressurizer extend above the top of the shielded concrete structures in the containment (i.e., above the top of the entombment structure), they must be relocated to allow the eventual construction of the entombment barrier; this relocation is accomplished by severing the piping connections, removing the equipment mountings, lowering the equipment to rest on the ground-floor slab, and securing the equipment in place to prevent subsequent movement that could result in injuries to workers or damage to the entombment structure. Finally, additional hatchways are cut as needed through the operating floor to facilitate movement of materials into the entombment structure area and, thus, to allow more complete use of the available structure volume. The methods used for these preparation activities are essentially the same as disassembly methods for DECON or are conventional methods used commonly in the building trades and, therefore, additional descriptions of these methods are not included here.

Once the area is prepared to serve as the entombment structure, the radioactive materials originating from outside of the structure that are selected to be entombed can be moved into the structure. The structure is filled from the bottom to the top. Larger items are moved into the entombment structure and stacked, while smaller items are simply dumped on top of the larger items to fill the spaces between them. Some of the radioactive materials from outside of the entombment structure are not amenable to entombment because of their large sizes, extremely high radioactivity levels, or other factors; these materials are packaged and shipped for offsite disposal. Of the materials that are amenable to entombment, 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 filling process may be packaged and shipped offsite simply because the remaining space in the entombment structure can be more easily filled with smaller items).

The volume utilization efficiency achieved in the entombment structure could be increased (and offsite transportation and disposal costs reduced accordingly) by using any of a number of techniques, including careful selection of material placement locations, nesting of piping and other equipment, and relatively complete filling of voids in emplaced waste with smaller items such as concrete rubble. However, this increased efficiency would increase manpower requirements and, consequently, manpower costs and occupational radiation doses.[4] It is beyond the scope of this study to optimize the storage, but optimization should be considered during the planning of any actual ENTOMB project.

On completion of the entombment-structure filling, the structure is sealed to provide a continuous barrier around the entombed radioactivity. The emergency airlock is sealed, and a continuous barrier of cast-in-place reinforced concrete is formed and poured at the operating floor level and the top of the central shielded structures containing the reactor vessel, steam generators, and pressurizer.

After completion of the entombment structure, appropriate security and surveillance systems are installed in the decontaminated portion of the containment dome above the structure, and all utilities not required during continuing care are disconnected. The equipment hatch is sealed and the personnel access hatch is fitted with an intrusion alarm and locked, so that the upper containment dome can serve as a secondary barrier over the entombment structure while still allowing access as needed to carry out the continuing care activities.

Accident severity is anticipated to have little or no effect on either the manpower requirements for efficient completion of ENTOMB or the overall duration of the ENTOMB project. However, because of the accident-generated contamination present, occupational radiation doses are expected to be greater than for ENTOMB following normal shutdown and to increase with increasing accident severity. The methods used for post-accident ENTOMB are anticipated to be the same as those for normal-shutdown ENTOMB, which are discussed in more detail in Chapter 4 of Reference 2 and Appendix K of Reference 4.

G.4.1.2 Fuel and Auxiliary Buildings

The methods used in the fuel and auxiliary buildings for post-accident ENTOMB are the same as those used in these buildings for post-accident DECON, as discussed previously in Sections G.2.1.2 and G.2.1.3 of this appendix. The only difference is that some of the radioactive materials removed from these buildings are disposed of by placement within the entombment structure rather than by packaging and shipping them offsite. As for DECON, the tasks required for ENTOMB and the schedule for performing them are not anticipated to vary with accident severity, although the occupational radiation doses to workers are expected to increase somewhat with increased accident severity. The

ENTOMB schedules for these two buildings may be adjusted somewhat from the corresponding DECON schedules to coordinate properly with the containment building activities and to allow for efficient overall completion of all of the ENTOMB tasks.

G.4.1.3 Ancillaries

The ancillary activities for post-accident ENTOMB at the reference PWR are as follows:

- DECON in other site buildings as required (e.g., the condensate-demineralizer building, the control building, and the turbine building)
- the shipment of the spent fuel from the final reactor core to an offsite repository
- the disposal of the radioactive waste materials generated during ENTOMB (i.e., the contaminated materials removed from outside of the entombment structure area and the combustible wastes generated in carrying out the ENTOMB tasks)
- disposition of the onsite waste storage structures postulated to handle the wastes generated during accident cleanup (including the offsite shipment of stored wastes not selected to be entombed onsite).[a]

The first two activities are the same as for post-accident DECON, as discussed previously in Section G.2.1.4, and are not discussed further here. The last two activities are discussed in the following paragraphs.

Because only part of the radioactive materials in the reference PWR are disposed of offsite, there is less packaging and shipping of radioactive wastes required. However, those wastes removed from outside the containment building that are selected to be entombed must be safely and efficiently

(a) The costs of packaging, shipping, and disposing of these wastes are included in the cost of accident cleanup, even though the removal of these wastes from the site is not anticipated to be completed during the cleanup campaign and time is allotted during the ENTOMB schedule to complete these activities.

transported to and placed in the entombment structure. It is anticipated that the reduced packaging and shipping activity is offset by the internal transport and placement activity and, thus, although the tasks performed are somewhat different, the decommissioning worker requirements for radioactive waste disposal remain about the same as for post-accident DECON.

There are three onsite waste storage structures postulated to be constructed on the site during accident cleanup to provide storage for cleanup-generated wastes (see Appendix E):

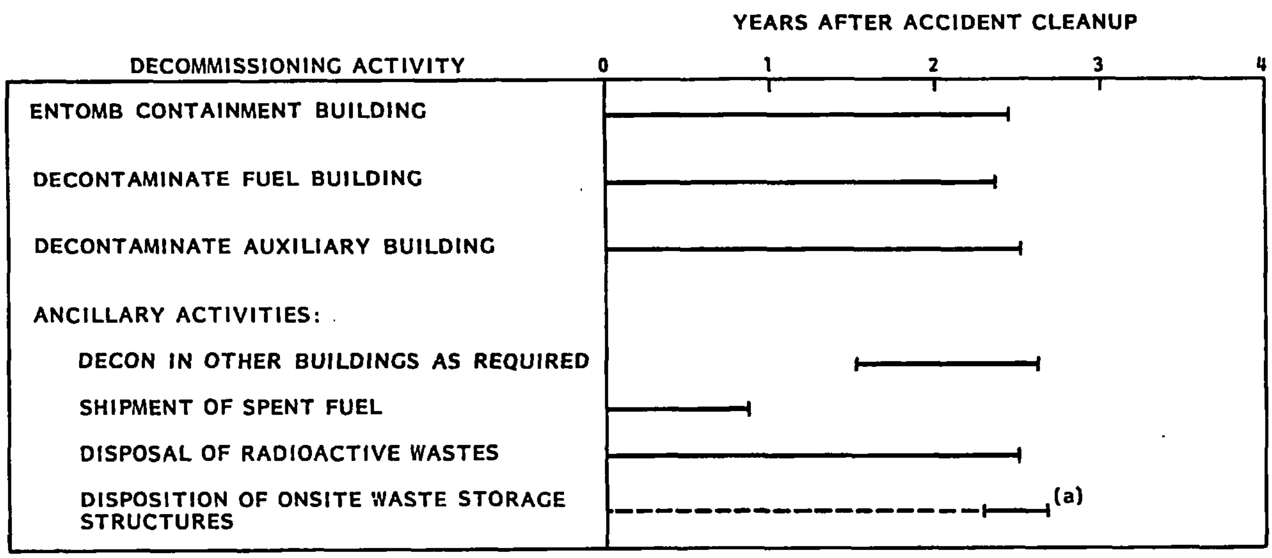
- a warehouse for storage of low-activity wastes
- a caisson storage facility to house high-activity wastes packaged in cask liners
- a canyon storage facility to receive high-activity wastes packaged in drums and boxes.

For post-accident DECON, it is assumed that the wastes are removed from all three structures, after which the structures are removed. For post-accident ENTOMB, it is postulated that this course of action is taken with the warehouse, but that the shielded storage facilities are entombed with the waste in place. Entombing these structures involves the sealing of the cover blocks in place and the decontamination of the upper parts of the structures, and is anticipated to require about the same length of time and the same manpower as DECON. The level of effort required for this task increases with increasing accident severity because of the larger waste storage structures required to handle the cleanup-generated wastes following the more severe accidents.

G.4.2 Schedule

It is assumed that, by the time the accident cleanup campaign is completed, the necessary regulatory approvals are in place so that ENTOMB can proceed. As for DECON, the primary decommissioning activities are postulated to be performed on a two-shift, 5-day-week schedule and selected support activities (i.e., CVCS decontamination and radwaste system operation) and security functions are scheduled on a three-shift, 7-day-week basis. The main control room is manned full time to ensure the availability of essential systems and services.

The overall schedule and sequence for ENTOMB at the reference PWR, following a scenario 2 accident and the subsequent accident cleanup campaign, is shown in Figure G.4-1. As shown in the figure, the ENTOMB project is postulated to span a period of 32-1/2 months, the same as DECON. This is consistent with information presented in References 2 and 3 and is based on the assumption that activities within the containment building require about the same level of effort for either ENTOMB or DECON, while activities outside of containment are virtually the same for the two alternatives. The overall duration of the ENTOMB project is not judged to vary substantially with changes in accident severity, even though the level of effort required for some individual tasks within the schedule is a function of accident severity.



(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.

<u>FIGURE G.4-1.</u> Duration of PWR ENTOMB Activities Following a Scenario 2 Accident

Decommissioning worker requirements for ENTOMB are not identified here but rather are calculated in the next subsection on the basis of the radiation dose limitations to the workers. These dose limitations are demonstrated in Section G.2.3 for DECON to be the controlling factor for staffing requirements, and the same situation prevails for ENTOMB.

G.4.3 Occupational Radiation Doses and Decommissioning Worker Requirements

The estimates presented here of the external occupational radiation doses that are accumulated by the decommissioning workers during post-accident ENTOMB at the reference PWR are based largely on the estimates for DECON presented previously in Section G.2.3. The differences between the two sets of estimates are as follows:

- The exposure hours accumulated within the containment building are assumed to be the same as for DECON. However, it is assumed that only half as much exposure time is required on the ground floor level, with proportionally more time split evenly between the other two levels to make up the difference, because ENTOMB tasks require considerably less time on the ground floor level (see Reference 3).
- The exposure times for the disposal of the accident cleanup wastes in the onsite waste storage structures are reduced by 40% to account for the entombing of the wastes in the shielded storage facilities, which leaves only the wastes in the warehouse facility to be shipped offsite (see Appendix E for waste volumes stored in these facilities).

All other activity areas are assumed to have the same exposure hours as DECON, even though the actual tasks performed in these areas may be somewhat different.

The estimated external occupational radiation doses for ENTOMB at the reference PWR following a postulated scenario 2 accident and the subsequent accident cleanup campaign are presented in Table G.4-1. As shown in the table, the total occupational dose is estimated to be over 2500 man-rem and, as it was for DECON, the area making the largest contribution to the total dose is the containment building.

TABLE G.4-1. Estimated Occupational Radiation Doses for ENTOMB at the Reference PWR Following a Scenario 2 Accident

ENTOMB Activity Area	Average Dose Rate (rem/hr)	Supervisors[a]		Utility Operators and Laborers		Craftsmen		Health Physics Technicians		Task Totals	
		Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)
Containment Building											
Operating Floor Level	0.010	4 459	44.59	18 075	180.75	8 165	81.65	5 693	56.93	36 392	363.92
Mezzanine Level	0.020	5 976	119.52	22 064	441.28	11 939	238.78	6 366	127.32	46 345	926.90
Ground Floor Level	0.050	1 801	90.05	7 514	375.70	3 851	192.55	2 003	100.15	15 169	758.45
Subtotals		12 236	254.16	47 653	997.73	23 955	512.98	14 062	284.40	97 906	2 049.27
Fuel Building											
All Levels	0.005	658	3.29	2 384	11.92	1 303	6.52	826	4.13	5 171	25.86
Auxiliary Building											
All Levels	0.010	1 687	16.87	2 395	23.95	3 776	37.76	1 747	17.47	9 605	96.05
Ancillaries											
Disposal of Radioactive Wastes:											
• Spent Fuel	0.010	567	5.67	9 072	90.72	--[b]	--	1 701	17.01	11 340	113.40
• ENTOMB Wastes	0.010	2 050	20.50	8 198	81.98	--	--	2 050	20.50	12 298	122.98
• Accident Cleanup Wastes[c]	0.010	307	3.07	1 230	12.30	307	3.07	307	3.07	2 151	21.51
Disposition of Onsite Waste Storage Structures	0.005	840	4.20	5 040	25.20	1 260	6.30	1 260	6.30	8 400	42.00
Radiation Surveys (Weekly)	0.010	--	--	--	--	--	--	4 676	46.76	4 676	46.76
Subtotals		3 764	33.44	23 540	210.20	1 567	9.37	9 994	93.64	38 865	346.65
Totals		18 345	307.76	75 972	1 243.80	30 601	566.63	26 629	399.64	151 547	2 517.83

(a) Includes shift engineers, crew leaders, craft supervisors, and senior health physics technicians.
(b) A dash indicates that, for the specified activity area, that particular staff category is not used.
(c) Removal of accident cleanup wastes remaining in the onsite waste storage warehouse at the time ENTOMB commences.

Based on the dose rate information presented in Table E.4-1 and assuming some small variation in manpower requirements with accident severity, the total occupational radiation doses for ENTOMB following a scenario 1 accident could be a factor of 2.5 lower than those following the scenario 2 accident, while the total doses following a scenario 3 accident could be up to 2.3 times more than those for scenario 2.

For post-accident DECON at the reference PWR, the total requirements for decommissioning workers were determined not by the numbers needed to efficiently perform the decommissioning tasks but by the radiation dose limitations on the individual workers (see Section G.2.3). This is also the case for ENTOMB. Based on the assumption that annual doses to workers are limited to 5 man-rem/man-year, the estimated total requirements for the individual categories of decommissioning workers are shown in Table G.4-2. Because the decommissioning worker requirements are based on radiation dose, they are anticipated to vary with accident severity by the same factors as the doses (see previous paragraph).

TABLE G.4-2. Decommissioning Worker Requirements for ENTOMB at the Reference PWR Following a Scenario 2 Accident

Decommissioning Worker Category	Estimated Total Occupational Dose[a] (man-rem)	Decommissioning Worker Requirements[b] (man-years)
Crew Leaders	184.66[c]	37.0
Utility Operators	460.67[d]	92.2
Laborers	783.13[d]	156.7
Craftsmen	566.63	113.4
Health Physics Technicians	399.64	80.0

(a) Based on Table G.4-1.
(b) Minimum number required to maintain average individual dose $\leq$5 man-rem/man-year.
(c) Assumes that crew leaders account for 60% of supervisory dose and that the remaining 40% accrues to shift engineers, craft supervisors, and senior health physics technicians.
(d) Assumes that ratio of utility operator to laborer time is 1:1.7 as for DECON (see Table G.2-1).

G.4.4 Overall Staff Labor Requirements

The staff organization postulated for post-accident ENTOMB at the reference PWR is the same as that for post-accident DECON, shown previously in Figure G.2-4. Further discussion pertaining to staff organization and the functions of key staff members can be found in Chapter 9 of Reference 1.

The total staff labor requirements for ENTOMB at the reference PWR following a scenario 2 accident are given in Table G.4-3. These requirements are given in equivalent man-years for the two decommissioning phases: planning and preparation (concurrent with the completion of accident cleanup) and the actual ENTOMB (following completion of the cleanup campaign). The management and support staff as well as the decommissioning workers are included. The decommissioning worker requirements are based on the requirements to comply with occupational radiation dose limitations, presented previously in Table G.4-2. Other assumptions used in developing the requirements are the same as those given previously in Section G.2.4 for DECON.

Almost 680 man-years of effort are estimated for ENTOMB at the reference PWR following a scenario 2 accident, not including the effort required for accident cleanup (see Appendix E). Included are almost 170 man-years for management and support staff and about 510 man-years for the decommissioning workers. Because management and support staff requirements are essentially a function of project duration and the overall duration for the ENTOMB project is not anticipated to vary significantly with accident severity, variations in management and support staff requirements with accident severity are judged to be minimal. However, the decommissioning worker requirements could vary from about 205 man-years following a scenario 1 accident to almost 1180 man-years following a scenario 3 accident, based on the occupational radiation dose limitations discussed previously in Section G.4.3. Thus, the total staff labor requirements for ENTOMB following a scenario 1 or scenario 3 accident are anticipated to be about 375 man-years or about 1345 man-years, respectively. The post-accident staff labor requirements for ENTOMB are somewhat lower than the corresponding requirements for DECON, primarily because of the lower occupational radiation doses accumulated in decommissioning the contair ment building.

TABLE G.4-3. 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.

G.4.5 Continuing Care and Possible Deferred Decontamination of the Entombed Plant

As discussed previously in Section G.4, the initial decommissioning activities for ENTOMB are followed by a period of continuing care that includes security, surveillance, and maintenance activities. These activities for ENTOMB are judged to require a lower level of effort than the comparable activities for SAFSTOR, because of the more rigorous preparations of the facility and the resulting reduced risk from the facility. Occupational radiation doses during continuing care are anticipated to be minimal. The continuing care period extends until either the entombed radioactivity decays to unrestricted release levels or the entombment structure is dismantled and the entombed radioactivity is removed (i.e., by deferred decontamination). If it becomes desirable to terminate the nuclear license prior to the decay of the entombed radioactivity to levels permitting unrestricted release, deferred decontamination of the entombment structure must be undertaken and the residual radioactivity removed from the facility.

Deferred decontamination following ENTOMB, though not analyzed here in detail, 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 (designed to retain its integrity under any but the most severe conditions) and 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. The occupational radiation doses associated with deferred decontamination decrease with the entombment time in accordance with the radioactive decay of the entombed materials. However, it should be noted that for post-accident ENTOMB the decay is slower than for ENTOMB following normal shutdown because the decay of the post-accident radionuclide inventory is controlled by ^{137}Cs (with a 30.17 year half-life) rather than by ^{60}Co (with a 5.27 year half-life) as is the normal-shutdown inventory. The overall total occupational doses from post-accident ENTOMB with deferred decontamination are thus not anticipated to be significantly less than those from post-accident DECON, while they are anticipated to be greater than those from post-accident SAFSTOR with deferred decontamination.

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. 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.

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.

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

APPENDIX H

DETAILS OF COSTS OF DECOMMISSIONING AT A REFERENCE PWR

This appendix presents the estimated costs of decommissioning the reference PWR via each of the three decommissioning alternatives (DECON, SAFSTOR, and ENTOMB) following a scenario 2 accident and the subsequent accident cleanup campaign. Estimates for decommissioning following the other two postulated accidents are arrived at by appropriate adjustment of post-scenario 2 estimates, taking into account the specific conditions that can affect costs. The costs developed here are based on the detailed descriptions of the decommissioning activities and requirements presented previously in Appendix G and on the cost estimating bases presented in Appendix I. Cost information for the decommissioning of the reference PWR following normal shutdown, from References 1 and 2, is also used where applicable. The decommissioning costs are all adjusted to early 1981, and are developed on a consistent basis with the costs of accident cleanup which are presented in Appendix F.

This appendix is made up of three major sections which address the costs of post-accident DECON, SAFSTOR, and ENTOMB, respectively. Within each section, cost estimates are developed for staff labor, waste management activities, energy, special tools and equipment, miscellaneous supplies, specialty contractors, nuclear insurance, and license fees. In addition, for SAFSTOR and ENTOMB, the costs of continuing care and deferred decontamination are examined.

The costs of decontaminating and disposing of systems and facilities required for accident cleanup (including the accident-water cleanup demineralizer system, the treated water storage tanks, and the solid waste storage facilities) are included in the estimates presented in this appendix. Cost variations relating to alternate scenarios for waste disposal are addressed separately in Chapter 15 of Volume 1.

H.1 DECON COSTS

The estimated costs of DECON at the reference PWR, following a scenario 2 accident and the subsequent accident cleanup campaign, are summarized and totaled in Table H.1-1. The total cost of DECON is estimated at about $67.9 million, including a 25% contingency. The major contributors to the total DECON cost are staff labor, waste management, and energy at approximately 49%, 30%, and 12% of the total, respectively. Combined costs for special tools and equipment and for miscellaneous supplies make up about 5.5% of the total. The remaining costs, about 3.5% of the total, are contributed by specialty contractors, nuclear insurance, and license fees. Detailed cost data for the individual cost categories shown in Table H.1-1 are presented and discussed in the following subsections.

Although no detailed cost estimates are made for DECON following either a scenario 1 (less severe) or a scenario 3 (more severe) accident, the overall variation in DECON costs with accident severity can be approximated using information presented in the following subsections. Based on this information, the total cost of DECON at the reference PWR following a scenario 1 or scenario 3 accident, including a contingency of 25%, is estimated to be about $49.3 million or about $105.8 million, respectively. These costs represent 73% and 157%, respectively, of the cost of DECON following a scenario 2 accident. The primary difference in the cost estimates for the three accident scenarios is in the staff labor costs, specifically the decommissioning worker costs. In addition, the waste management and energy costs contribute somewhat to the difference.

H.1.1 Costs of Staff Labor

The costs of staff labor during DECON at the reference PWR, following a scenario 2 accident and accident cleanup, are shown in detail in Table H.1-2. Just under 50% of the total DECON cost is associated with staff labor. A total staff labor cost of about $26.8 million is estimated for DECON after a scenario 2 accident, not including specialty contractor labor. About 3/4 of the total staff labor cost, or about $20.7 million, is for the decommissioning workers; the remaining 1/4, about $6.1 million, is for the management and support staff.

TABLE H.1-1. Summary of Estimated Costs of DECON at the Reference PWR Following a Scenario 2 Accident

Cost Category	Estimated Costs ($ millions)[a]		Percent of Total
Staff Labor			
Management and Support Staff	6.112		
Decommissioning Workers	20.714		
Total Staff Labor Costs		26.826	49.4
Waste Management			
Neutron-Activated Materials	4.511		
Contaminated Materials	10.049		
Radioactive Wastes	1.665		
Total Waste Management Costs		16.225	29.9
Energy		6.524	12.0
Special Tools and Equipment		1.028	1.9
Miscellaneous Supplies		1.892	3.5
Specialty Contractors		0.636	1.2
Nuclear Insurance and License Fees		1.138	2.1
Subtotal		54.286	100.0
Contingency (25%)		13.571	
Totals, DECON Costs		67.857	

(a) Costs adjusted to early 1981; the number of significant figures shown is for computational accuracy only.

Based on the discussion of staff labor requirements presented in Section G.2.4 of Appendix G, the requirements for and, consequently, the costs of the management and support staff are not anticipated to vary substantially with the severity of the accident that has occurred. However, the decommissioning worker requirements could vary from about 210 man-years following a scenario 1 accident to about 1560 man-years following a scenario 3 accident. Accordingly, the decommissioning worker costs and the total staff labor costs following a scenario 1 accident are estimated at approximately $7.0 million and $13.1 million, respectively, and the corresponding costs following a

TABLE H.1-2. Estimated Costs of Staff Labor During DECON at the Reference PWR Following a Scenario 2 Accident

Position	Unit Cost for Labor[(a)] ($ thousands/man-year)	Total Staff Labor Required[(b)] (man-years)	Total Staff Labor Costs[(c)] ($ thousands)
Management and Support Staff			
Decommissioning Superintendent	89.4	4.5	402.3
Secretary	24.4	11.5	280.6
Clerk	24.4	6.4	156.2
Decommissioning Engineer	76.2	4.5	342.9
Assistant Decommissioning Engineer	52.6	4.2	220.9
Radioactive Shipment Specialist	39.5	2.7	106.7
Procurement Specialist	39.3	2.7	106.1
Tool Crib Attendant	27.8	5.4	150.1
Reactor Operator	34.8	21.7	755.2
Security Supervisor	55.9	2.7	150.9
Security Shift Supervisor	36.8	10.8	397.4
Security Patrolmen	25.6	28.2	721.9
Contracts and Accounting Supervisor	47.1	3.0	141.3
Health and Safety Supervisor	60.5	3.0	181.5
Health Physicist	47.3	2.7	127.7
Protective Equipment Attendant	27.8	5.4	150.1
Industrial Safety Specialist	52.6	2.7	142.0
Quality Assurance Supervisor	52.6	3.0	157.8
Quality Assurance Engineer	47.3	2.7	127.7
Quality Assurance Technician	27.8	10.8	300.2
Consultant (Safety Review)	100.0	1.4	140.0
Instrument Technician	32.5	10.8	351.0
Maintenance Mechanic	32.5	10.8	351.0
Warehouseman	27.8	5.4	150.1
Subtotals		167.0	6 111.6
Decommissioning Workers			
Shift Engineer	52.6	5.4	284.0
Crew Leader	44.8	45.5	2 038.4
Utility Operator	32.5	114.1	3 708.3
Laborer	31.1	194.4	6 045.8
Craft Supervisor	47.3	10.8	510.8
Craftsman	32.5	142.8	4 641.0
Senior Health Physics Technician	39.5	16.3	643.9
Health Physics Technican	30.1	94.4	2 841.4
Subtotals		623.7	20 713.6
Totals		790.7	26 825.2

(a) Data from Table I.1-1 of Appendix I.
(b) Data from Table G.2-4 of Appendix G.
(c) Number of figures shown is for computational completeness only.

scenario 3 accident are estimated at about $51.8 million and $57.9 million. Thus, the total DECON staff labor costs following a scenario 1 accident are about half of those following a scenario 2 accident which, in turn, are about 45% of those following a scenario 3 accident.

H.1.2 Costs of Waste Management

Three distinct types of radioactive waste materials require packaging, shipping, and disposal during DECON: 1) neutron-activated materials, 2) contaminated materials, and 3) radioactive wastes. The total waste management costs during DECON at the reference PWR, following a scenario 2 accident and the subsequent accident cleanup campaign, are about $16.2 million, representing about 30% of the total DECON cost. The waste management costs include the container, transportation, and burial site costs, but does not include the direct labor costs for removing and packaging these materials (see staff labor costs in Section H.1.1). Wastes generated during the accident cleanup campaign preceding DECON are not included here, but are reported separately in Appendix F.

Table H.1-3 presents a breakdown of the costs and other pertinent parameters associated with waste management during DECON. The management of each of the three types of radioactive waste materials is discussed individually in the following subsections.

H.1.2.1 Neutron-Activated Materials

All of the neutron-activated materials are contained in the reactor pressure vessel, the vessel internal structures, and in the surrounding steel and concrete biological shield, all of which are located in the containment building. A detailed breakdown of these materials is presented in Table G.4-3 of Reference 1, which provides the basis for the cost estimates developed here for management of neutron-activated materials. During the accident cleanup campaign following a scenario 2 accident, the upper core support assembly, upper support columns, and upper core grid plate are postulated to be removed to facilitate reactor defueling (see Appendix E) and to be disposed of as part of the accident cleanup activities. Thus, these materials are not included in the waste management estimates for DECON.

TABL ent

Waste Category	Estimated Mass (kg)	Burial Site Costs(d,e) ling Charge $)	State Surcharge ($)	Liner Surcharge ($)	Curie Surcharge ($)	Total Waste Management Costs ($)
Neutron-Activated Steels	~574 880	750	4 750	155 510	503 680	4 068 060
Neutron-Activated Concrete	~884 500	930	7 160	--	--	443 190
Contaminated Equipment	3 901 030	010	56 160	--	--	3 083 960
Contaminated Concrete	13 571 880	980	115 040	--	--	6 964 920
Compactible, Combustible Trash(f)	52 000	940	1 160	--	--	357 500
Compactible, Noncombustible Trash(f)	168 500	820	3 750	--	--	201 940
Noncompactible Trash	510 000	880	9 350	--	--	501 750
Spent Resins	~12 000	350	310	11 160	8 490	111 270
Spent Filter Cartridges	~4 200	010	100	--	4 410	48 090
Solidified Evaporator Bottoms	~47 000	750	1 410	14 890	--	444 450
Totals	19 726 000	420	199 190	181 560	516 580	16 225 130

(a) Based on information from Table I.2-1 of
(b) Based on information from Table I.2-2 of
(c) Based on information from Table I.3-4 of
(d) Charges for individual shipments may var
(e) Based on information from Table I.4-1 of
(f) Values shown are for waste after treatme

TABLE H.1-3. Estimated Costs of Radioactive Waste Management During PWR DECON Following a Scenario 2 Accident

To calculate the waste management costs presented in Table H.1-3 for neutron-activated steels and concrete, the basic information pertinent to post-accident DECON is taken from Table G.4-3 of Reference 1 and then the costs are recalculated based on the cost estimating bases in Appendix I of this study. After a scenario 2 accident and accident cleanup, the total radioactivity expected to be present in the neutron-activated materials is approximately 4.8 million curies. The packaged materials require about 203 truckload shipments to a shallow-land burial facility and occupy over 1100 m^3 of space at the burial facility. The total estimated cost of managing the neutron-activated materials is about \$4.5 million, of which almost \$4.1 million is attributable to neutron-activated steels and about \$440,000 to concrete.

The neutron activation of the PWR components takes place during normal reactor operations, not as a result of the postulated reactor accident and therefore the inventory of neutron-activated materials in the plant at the start of the accident cleanup campaign is unaffected by accident severity. However, it is postulated that more of the reactor vessel internals are removed to facilitate defueling following a scenario 3 accident than following either a scenario 1 or scenario 2 accident (see Appendix E). Specifically, the extra components removed following a scenario 3 accident are (as listed in Table G.4-3 of Reference 1) the guide tubes, thermal shields, core shroud, lower grid plate, lower support columns, lower core forging, and miscellaneous internals. Deleting these components from the waste management requirements for DECON reduces the overall cost of managing neutron-activated wastes following a scenario 3 accident to about \$3.2 million.

H.1.2.2 Contaminated Materials

Materials considered to be contaminated include nearly all piping and equipment present in the containment, auxiliary, fuel, and control buildings. In addition, significant areas of concrete surfaces in these buildings and in the onsite waste storage structures (constructed during accident cleanup) are also assumed to be contaminated and, consequently, to require surface removal to a depth of about 50 mm.

A detailed breakdown of the contaminated materials at the reference PWR requiring disposal during DECON following normal shutdown is presented in Tables G.4-4 and G.4-5 of Reference 1. This information provides the basis for the cost estimates developed in this study for the management of contaminated waste materials during DECON following a scenario 2 accident. The inventory of contaminated materials presented in Reference 1 is modified to account for the impacts of the postulated accident and the subsequent accident cleanup activities as follows:

- the amount of contaminated concrete and steel liner plate removed during internal surface decontamination in the containment building is assumed to increase from 5 to 19 fiberglassed plywood boxes of about 3.5 m^3 each
- two fiberglassed plywood boxes each of concrete and miscellaneous equipment are assumed to be removed during the decontamination of the onsite waste storage structures
- three additional fiberglassed plywood boxes are assumed for disposal of the demineralizer system installed in the spent fuel storage pool during the accident cleanup campaign
- the overall level of contamination on all of the materials removed is assumed to be $\sim$3-5 times higher than the levels assumed for DECON following normal shutdown.

Following this contaminated-material inventory modification, the waste management costs are recalculated based on the cost estimating bases in Appendix I of this study. ,

After a scenario 2 accident and accident cleanup, disposal of the contaminated materials is estimated to require 1001 truckload shipments and 16,150 m^3 of space at a shallow-land burial site. The total cost of managing the contaminated waste materials is estimated to be about \$10.0 million, almost \$3.1 million for contaminated equipment and almost \$7.0 million for contaminated concrete.

Because, as stated previously, the vast majority of the materials in the major site buildings are considered to be contaminated, accident severity is anticipated to have little effect on the amount of contaminated material requiring disposal. However, accident severity is expected to influence the level of contamination on these materials. It is judged that the contaminated-material management costs would not increase or decrease by more than about 5% for a scenario 1 or a scenario 3 accident. Thus, the total cost of managing contaminated waste material during DECON following a scenario 1 accident is postulated to be about $9.5 million, and following a scenario 3 accident the cost is postulated to be about $10.5 million.

H.1.2.3 Radioactive Wastes

While not a prime "product," radioactive wastes (radwastes) result directly from DECON at the reference PWR. The radioactive waste materials considered in this study are as follows:

- radioactive trash (consisting of a compactible and combustible fraction, a compactible but noncombustible fraction, and a noncompactible [and also noncombustible] fraction)
- spent ion exchange resins
- spent filter cartridges
- solidified evaporator bottoms.

The total management costs for radioactive wastes during DECON following a scenario 2 accident are estimated to be almost $1.7 million. These wastes are estimated to require 158 truck shipments and to occupy 1516 m^3 of space at a shallow-land burial site. The details of these estimates are discussed in the following paragraphs.

The estimates of radioactive trash from DECON are developed using the same assumptions as are used for post-accident cleanup (see Section E.4 of Appendix E), which are derived from Reference 3. In summary, these assumptions are as follows:

- Radioactive trash is generated at a rate of 0.05 m^3/man-hr in a radiation area (i.e., 0.05 m^3/exposure-hour).
- 90% of the trash is compactible and 75% of the compactible trash is also combustible.

- Compaction reduces trash volumes by a factor of 5.
- Compaction and combustion together reduce trash volumes by a factor of 50.
- Average radioactivity content of the trash before treatment is 0.035 Ci/m^3.

In addition, the noncompactible trash is assumed to be packaged in 3.46-m^3 fiberglassed plywood boxes, and the treated (compacted, incinerated) waste is assumed to be packaged in standard 0.21-m^3 steel waste drums. The waste management costs for the radioactive trash are then calculated using the cost estimating bases provided in Appendix I. The total cost of managing radioactive trash during DECON at the reference PWR following a scenario 2 accident is about $1.05 million.

The waste management requirements and costs for the other radioactive wastes (i.e., spent resins, spent filter cartridges, and solidified evaporator bottoms) are developed from information in Section G.4.2.3 of Reference 1 for DECON following normal shutdown. The projected volumes of the spent resins and the solidified evaporator bottoms are reduced by 50% from those for normal-shutdown DECON because, following an accident, chemical decontamination of the reactor coolant system takes place during accident cleanup rather than during decommissioning, reducing the generation of these wastes during DECON. The costs are recalculated based on the information in Appendix I of this study. The total cost of managing these radioactive wastes during DECON following a scenario 2 accident is estimated to be over $600,000.

The waste management requirements and costs for the radioactive wastes are not judged to be significantly altered by changes in accident severity (no more than ±10% from the scenario 2 values for scenario 1 or scenario 3). This is because the radioactive trash is a function of exposure hours, which vary little with accident severity, and the other radioactive wastes are functions of the tasks performed (which do not change significantly) rather than of contamination levels in the plant.

H.1.3 Costs of Energy

Electricity is the principal cost item associated with providing essential systems and services during the decommissioning of the reference PWR. From Section 10.1.8 of Reference 1, the plant base load during decommissioning is estimated to be about 11 MW. Over the 32.5 months of DECON, this results in a total of about 261,000 MWh. At a unit cost of $25/MWh (see Section I.6 of Appendix I), the total energy cost during DECON at the reference PWR following a scenario 2 accident is about $6.5 million. Because the requirements of DECON and the duration of the project vary little with the severity of the postulated accident, energy costs are anticipated to be decreased or increased by no more than 5% of the scenario 2 value following a scenario 1 or scenario 3 accident, respectively.

H.1.4 Costs of Special Tools and Equipment

Based on information presented in Table D.3-1 of Appendix D, the estimated costs of the special tools and equipment that are required for DECON at the reference PWR following a scenario 2 accident are presented in Table H.1-4. The estimates assume reuse of some equipment used for the accident cleanup campaign preceding DECON. The estimated total cost of special tools and equipment is approximately $1.0 million, or about 2% of the total cost for DECON after a scenario 2 accident. Accident severity (within the range of accident scenarios considered in this study) is judged to have no significant impact on the requirements for and the costs of special tools and equipment during DECON.

H.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 (e.g., cleansing agents, rags, mops, and wiping materials), and expendable tools and materials. The estimated costs of these items are given in Table H.1-5. The quantities of decontamination chemicals and ion exchange resins are assumed to be 50% of those given in Table 10.1-6 of Reference 1 because of the elimination of the requirement for chemical

TABLE H.1-4. Costs of Special Tools and Equipment for DECON at the Reference PWR Following a Scenario 2 Accident

Item	Estimated Unit Costs[a] ($ thousands)	Estimated Number Required[b]	Estimated Total Costs ($ thousands)
Underwater Plasma-Arc Torch	20	2	40
Underwater Oxyacetylene Torch	5	2	10
Portable Plasma-Arc Torch	20	4	80
Portable Oxyacetylene Torch	1	2	2
Guillotine Pipe Saw	4	10	40
Power-Operated Reciprocating Hacksaw	1	10	10
Underwater Lights and Viewing Aids	--	AR[c]	5 total
Underwater Tools	--	AR	25 total
Submersible Pump with Disposable Filters	2	4	8
Scaffolding and Safety Nets	--	AR	25 total
Mobile Chemical Decontamination Unit	20	5	100
Mobile Chemical Mixing and Heating Unit	5	4	20
Power-Operated Mobile Manlift	40	7	280
9100-kg Mobile Hydraulic Crane	28	1	28
9100-kg Forklift	28	4	112
Rigging Materials	--	AR	25 total
Concrete Drill With HEPA-Filtered Dust Collection System	2	4	8
Concrete Surface Spaller	5	4	20
Front-End Loader (light-duty)	20	3	60
Vacuum Cleaner (HEPA-filtered)	1-5[d]	3	~12
Portable Ventilation Enclosure	2-10[d]	10	~75
Filtered Exhaust Fan Unit	5	4	20
Supplied-Air Plastic Suit	0.05	250	13
Polyurethane Foam Generator	5	2	10
Total Costs			1028

(a) Data from Table I.5-1 of Appendix I.
(b) Derived from Table D.3-1 of Appendix D; does not include items anticipated to be reusable from accident cleanup activities.
(c) AR = as required.
(d) Depends on size and complexity.

TABLE H.1-5. Estimated Costs for Miscellaneous Supplies During DECON at the Reference PWR Following a Scenario 2 Accident

Item	Quantity	Total Cost[a] ($ thousands)
Decontamination Chemicals (EDTA/Oxalic Acid/Citric Acid)	9050 kg	15
Ion Exchange Resins	15 m^3	75
Filters	Unspecified	400
Protective Clothing	Unspecified	731
Cleaning Supplies	Unspecified	400
Expendable Tools and Materials	Unspecified	271
Total Costs		1892

(a) Rounded to the nearest $1000.

decontamination of the reactor coolant system, as discussed previously. The cost of protective clothing is based on the value in the reference adjusted by the ratio of exposure hours for post-accident DECON and for DECON following normal shutdown. Expendable tools and materials costs are estimated based on an average cost of $100,000/year over the span of the DECON project.

The total estimated cost of miscellaneous supplies during DECON at the reference PWR after a scenario 2 accident is about $1.9 million and represents 3.5% of the total DECON cost. The costs are not judged to vary significantly with changes in accident severity within the range of accident scenarios considered in this study.

H.1.6 Costs of Specialty Contractors

The requirements for and costs of specialty contractors (including environmental surveillance services) for DECON at the reference PWR are judged to be substantially the same whether or not the reactor has experienced an accident. Therefore, it is assumed that the specialty contractor cost presented in Table 10.1-1 of Reference 1, when adjusted by a factor of 1.2 to update the cost to 1981, is appropriate for this analysis. Thus, the total cost of specialty contractors during post-accident DECON at the reference PWR is estimated to be about $650,000, regardless of the severity of the postulated accident.

H.1.7 Costs of Nuclear Insurance and License Fees

The costs of nuclear liability insurance, for an assumed policy limit of $160 million carried through the DECON period, and of fees charged for licensing services performed by the NRC during DECON are shown in Table H.1-6. These costs total over $1.1 million, or about 2% of the costs of DECON following a scenario 2 accident, and are judged not to be significantly altered by changes in accident severity.

TABLE H.1-6. Estimated Costs of Nuclear Liability Insurance and License Fees During DECON at the Reference PWR Following a Scenario 2 Accident

Category	Unit Cost ($)	Total Cost ($)
Nuclear Liability Insurance	400 000/yr	1 100 000[a]
License Fees:[b]		
Facility License Amendment (Class IV)	12 300	24 600[c]
Routine Health, Safety, and Environmental Inspections	650/yr	1 950[d]
Routine Safeguards Inspection	11 800/yr	11 800[e]
Total Costs		1 138 350

(a) Prorated by quarters for the duration of the decommissioning project (i.e., ~2-3/4 years for DECON).
(b) From 10 CFR 170.[4]
(c) Based on two license amendments: one to allow possession but not operation of the plant, obtained prior to decommissioning, and one to terminate the license following completion of DECON.
(d) Based on annual inspections for the duration of the decommissioning project (i.e., 3 annual inspections for DECON).
(e) Based on having spent fuel onsite for 10.5 months (i.e., one yearly fee charged).

H.2 SAFSTOR COSTS

The estimated costs of SAFSTOR at the reference PWR, after the reactor has experienced a postulated scenario 2 accident and the accident cleanup campaign has been completed, are summarized and totaled in Table H.2-1. Costs are included for the three phases of SAFSTOR:

TABLE H.2-1. Summary of Estimated Costs of SAFSTOR at the Reference PWR Following a Scenario 2 Accident

Cost Category	Estimated Costs ($ millions)[a]		Percent of Total
Preparations for Safe Storage			
Staff Labor			
Management and Support Staff	3.492		
Decommissioning Workers	3.406		
Total Staff Labor Costs		6.898	51.7
Waste Management (Radioactive Wastes)		0.853	6.4
Energy		3.413	25.6
Special Tools and Equipment		0.329	2.5
Miscellaneous Supplies		0.929	7.0
Specialty Contractors		0.292	2.2
Nuclear Insurance and License Fees		0.625	4.7
Subtotal		13.339	100.1[b]
Contingency (25%)		3.335	
Totals, Preparations for Safe Storage Costs		16.674	
Annual Continuing Care Costs		0.111	
Deferred Decontamination Costs			
After 30-Year Safe Storage		57.6	
After 100-Year Safe Storage		44.7	
Total SAFSTOR Costs			
With 30-Year Safe Storage		77.6	
With 100-Year Safe Storage		72.5	

(a) Costs adjusted to early 1981; the number of significant figures shown is for computational accuracy only.
(b) Total does not equal 100 because individual percentages are rounded to the nearest one-tenth.

- preparations for safe storage
- continuing care (i.e., the safe storage period)
- deferred decontamination.

The total costs shown in the table for each phase of SAFSTOR include a 25% contingency. The cost estimates for each phase of SAFSTOR are discussed in the following subsections.

The total costs of SAFSTOR at the reference PWR following a scenario 2 accident are estimated to be about $77.6 million with a 30-year safe storage period and about $72.5 million with a 100-year safe storage period. Preparations for safe storage are estimated to cost a total of about $16.7 million, 21 to 23% of the total SAFSTOR costs. Annual continuing care costs during the safe storage period are estimated at about $111,000 for cumulative totals of $3.3 million or $11.1 million for 30 or 100 years, respectively, of safe storage. Deferred decontamination, representing the majority of the total SAFSTOR costs, is estimated to require total expenditures of $57.6 million after a 30-year safe storage period or $44.7 million after 100 years of storage. All costs are given in constant 1981 dollars, with no cost escalation included to account for possible inflationary effects.

Although no detailed analyses are made of the costs of SAFSTOR following either a scenario 1 (less severe) or a scenario 3 (more severe) accident, the overall variation in SAFSTOR costs with accident severity can be approximated using information presented in the following subsections. Following a scenario 1 accident, the total cost of SAFSTOR (including contingency) is estimated to be about $60 million with 30-year safe storage or just over $58 million with 100-year storage. Following a scenario 3 accident, these costs are estimated at about $115 million or about $103 million with 30-year or 100-year safe storage periods, respectively. The cost differences between the three accident scenarios result primarily from the deferred decontamination, with a smaller impact from the preparations for safe storage.

H.2.1 Costs of Preparations for Safe Storage

The total estimated cost of preparations for safe storage at the reference PWR, following a scenario 2 accident and the subsequent accident cleanup campaign, is shown in Table H.2-1, presented previously, to be about $16.7 million including a 25% contingency. The major contributors to the total cost are staff labor and energy at about 52% and 26% of the total, respectively. Waste management costs account for approximately 6% of the total, and the combined costs of special tools and equipment and of miscellaneous supplies make up close to 10% of the total. The remaining costs, representing about 7% of the total, are contributed by specialty contractors, nuclear insurance and license fees. The details of these costs are given, by cost category, in the following pages.

Based on information presented in the following pages, the total cost of preparations for safe storage at the reference PWR following a scenario 1 or scenario 3 accident (including the 25% contingency) is estimated to be about \$14.8 million or about \$21.5 million, respectively. These represent about 90% and 130%, respectively, of the cost of preparations for safe storage following a scenario 2 accident. The primary difference in the cost estimates for the three postulated accident scenarios is in the staff labor costs, specifically the decommissioning worker costs. In addition, the costs of energy and waste management contribute somewhat to the difference.

H.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. Over 50% of the total cost of preparations for safe storage is attributable to staff labor. A total staff labor cost of almost \$6.9 million is estimated, not including specialty contractor labor. The total cost is split nearly evenly between the management and support staff (almost \$3.5 million) and the decommissioning workers (about \$3.4 million).

Based on the discussion of staff labor requirements presented in Section G.3.4 of Appendix G, the requirements for and, consequently, the costs of the management and support staff are not anticipated to be significantly affected by the postulated accident scenario. However, the decommissioning worker requirements are anticipated to change with accident scenario. Accordingly, the decommissioning worker costs and the total staff labor costs are estimated to be about \$2.1 million and \$5.6 million, respectively, following a scenario 1 accident and about \$6.9 million and \$10.4 million following a scenario 3 accident. Thus, the total staff labor costs during preparations for safe storage following a scenario 1 accident are about 81% of those following a scenario 2 accident which, in turn, are about 66% of those following a scenario 3 accident.

TABLE H.2-2. Estimated Costs of Staff Labor During Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident

Position	Unit Cost for Labor[a] ($ thousands/man-year)	Total Staff Labor Required[b] (man-years)	Total Staff Labor Costs[c] ($ thousands)
Management and Support Staff			
Decommissioning Superintendent	89.4	3.3	295.0
Secretary	24.4	8.3	202.5
Clerk	24.4	3.8	92.7
Decommissioning Engineer	76.2	3.3	251.5
Assistant Decommissioning Engineer	52.6	2.9	152.5
Radioactive Shipment Specialist	39.5	1.4	55.3
Procurement Specialist	39.3	1.4	55.0
Tool Crib Attendant	27.8	2.8	77.8
Reactor Operator	34.8	11.3	393.2
Security Supervisor	55.9	1.4	78.3
Security Shift Supervisor	36.8	5.7	209.8
Security Patrolman	25.6	14.8	378.9
Contracts and Accounting Supervisor	47.1	1.8	84.8
Health and Safety Supervisor	60.5	1.8	108.9
Health Physicist	47.3	1.4	66.2
Protective Equipment Attendant	27.8	2.8	77.8
Industrial Safety Specialist	52.6	1.4	73.6
Quality Assurance Supervisor	52.6	1.8	94.7
Quality Assurance Engineer	47.3	1.4	66.2
Quality Assurance Technician	27.8	5.7	158.5
Consultant (Safety Review)	100.0	0.7	70.0
Instrument Technician	32.5	5.7	185.3
Maintenance Mechanic	32.5	5.7	185.3
Warehouseman	27.8	2.8	77.8
Subtotals		93.4	3 491.6
Decommissioning Workers			
Shift Engineer	52.6	2.8	147.3
Crew Leader	44.8	6.7	300.2
Utility Operator	32.5	24.5	796.3
Laborer	31.1	21.5	668.7
Craft Supervisor	47.3	5.7	269.6
Craftsman	32.5	8.9	289.3
Senior Health Physics Technician	39.5	8.5	335.8
Health Physics Technician	30.1	19.9	599.0
Subtotals		98.5	3 406.2
Totals		191.9	6 897.8

(a) Data from Table I.1-1 of Appendix I.
(b) Data from Table G.3-3 of Appendix G.
(c) Number of figures shown is for computational completeness only.

H.2.1.2 Costs of Waste Management

Only one type of radioactive waste materials, termed radioactive wastes in previous discussions of waste management, requires packaging, shipping, and offsite disposal during preparations for safe storage. Included in this waste type are radioactive trash, spent ion exchange resins, spent filter cartridges, and solidified evaporator bottoms. Table H.2-3 presents a breakdown of the costs and other pertinent parameters associated with waste management during preparations for safe storage. The total cost of waste management is estimated to be about $850,000, representing about 6.5% of the overall total cost of preparations for safe storage.

The estimates of radioactive trash from preparations for safe storage are developed using the same assumptions as those used to estimate trash from DECON, presented previously in Section H.1.2.3. The estimates for the other radioactive waste materials are the same as those for DECON because they result from activities that must be carried out regardless of the alternative selected for decommissioning the reference PWR.

The requirements for and costs of waste management during preparations for safe storage are not judged to be significantly altered by changes in accident scenario (no more than ±10% from the scenario 2 values for scenario 1 or scenario 3), as discussed in Section H.1.2.3 for management of these wastes during DECON.

H.2.1.3 Costs of Energy

Electricity is the principal cost item resulting from the need to provide essential systems and services during preparations for safe storage. Using the assumptions presented in Section H.1.3 for estimating energy costs during DECON, the total energy usage and resulting cost during preparations for safe storage at the reference PWR are about 136,500 MWh and over $3.4 million, assuming the reactor has experienced a scenario 2 accident. The costs are anticipated to vary in the range of ±5% for the other accident scenarios, as discussed previously for DECON.

TABLE H.2-3. E ario 2 Accident

Waste Category	Estimated Mass (kg)	ndling rcharge ($)	State Surcharge ($)	Liner Surcharge ($)	Curie Surcharge ($)	Total Waste Management Costs ($)
		Burial Site Costs(d,e)				
Compactible, Combustible Trash(f)	12 000	11 120	270	--	--	84 630
Compactible, Noncombustible Trash(f)	38 500	1 240	860	--	--	47 980
Noncompactible Trash	118 000	4 070	2 160	--	--	116 120
Spent Resins	~12 000	3 350	310	11 160	8 490	111 270
Spent Filter Cartridges	~4 200	2 010	100	--	4 410	48 090
Solidified Evaporator Bottoms	~47 000	15 750	1 410	14 890	--	444 450
Totals	~231 700	37 540	5 110	26 050	12 900	852 540

(a) Based on information from Table I.2-1
(b) Based on information from Table I.2-2
(c) Based on information from Table I.3-4
(d) Charges for individual shipments may
(e) Based on information from Table I.4-1
(f) Values shown are for waste after trea

TABLE H.2-3. Estimated Costs of Radioactive Waste Management During PWR Preparations for Safe Storage Following a Scenario 2 Accident

H.2.1.4 Costs of Special Tools and Equipment

Based on information presented in Table D.3-1 of Appendix D, the estimated costs of special tools and equipment during preparations for safe storage at the reference PWR are presented in Table H.2-4. It is assumed that some of the equipment used for the accident cleanup campaign preceding preparations for safe storage is reused (as specified in Table D.3-1). The total estimated cost is about $330,000, representing 2.5% of the overall cost of preparations for safe storage. Within the range of accident scenarios considered in this study, accident severity is judged to have no significant impact on the requirements for and costs of these special tools and equipment.

H.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. The cost estimates are arrived at using the same assumptions as those used for the corresponding DECON estimates, presented previously in Section H.1.5. The total estimated cost of miscellaneous supplies during preparations for safe storage is about $930,000, or about 7% of the overall cost of preparations for safe storage. These costs are not judged to vary significantly with changes in accident severity within the range of accident scenarios considered in this study.

H.2.1.6 Costs of Specialty Contractors

The requirements for and costs of specialty contractors during preparations for safe storage at the reference PWR are judged to be substantially the same whether or not the reactor has experienced an accident. (This is the same assumption as for DECON; see Section H.1.6). Therefore, the specialty contractor costs presented in Table 10.2-1 of Reference 1 are adjusted by a factor of 1.2 to update the costs to 1981. The total cost of specialty contractors during post-accident preparations for safe storage at the reference PWR is estimated to be just over $290,000, regardless of the accident scenario.

TABLE H.2-4. Costs of Special Tools and Equipment for Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident

Item	Estimated Unit Costs[a] ($ thousands)	Estimated Number Required[b]	Estimated Total Costs ($ thousands)
Portable Plasma-Arc Torch	20	2	40
Portable Oxyacetylene Torch	1	2	2
Guillotine Pipe Saw	4	2	8
Power-Operated Reciprocating Hacksaw	1	2	2
Underwater Lights and Viewing Aids	--	AR[c]	5 total
Underwater Tools	--	AR	5 total
Submersible Pump with Disposable Filters	2	2	4
Scaffolding and Safety Nets	--	AR	10 total
Mobile Chemical Decontamination Unit	20	5	100
Mobile Chemical Mixing and Heating Unit	5	4	20
Power-Operated Mobile Manlift	40	1	40
Rigging Materials	--	AR	10 total
Concrete Drill with HEPA-Filtered Dust Collection System	2	1	2
Concrete Surface Spaller	5	1	5
Front-End Loader (light-duty)	20	1	20
Vacuum Cleaner (HEPA-filtered)	1-5[d]	3	~12
Portable Ventilation Enclosure	2-10[d]	3	~25
Supplied-Air Plastic Suit	0.05	100	5
Polyurethane Foam Generator	5	2	10
Paint Sprayer	0.5-1[d]	4	4
Total Costs			329

(a) Data from Table I.5-1 of Appendix I.
(b) Derived from Table D.3-1 of Appendix D; does not include items anticipated to be reusable from accident cleanup activities.
(c) AR = as required.
(d) Depends on size and complexity.

TABLE H.2-5. Estimated Costs of Miscellaneous Supplies During Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident

Item	Quantity	Total Cost[a] ($ thousands)
Decontamination Chemicals (EDTA/Oxalic Acid/Citric Acid)	9050 kg	15
Ion Exchange Resins	15 m^3	75
Filters	Unspecified	400
Protective Clothing	Unspecified	167
Cleaning Supplies	Unspecified	130
Expendable Tools and Materials	Unspecified	142
Total Costs		929

(a) Rounded to the nearest $1000.

H.2.1.7 Costs of Nuclear Insurance and License Fees

The estimated costs of nuclear liability insurance and licensing fees during preparations for safe storage are shown in Table H.2-6. These costs total about $625,000 (almost 5% of the total cost of preparations for safe storage) and do not vary with changes in accident severity.

H.2.2 Cost of Continuing Care During Safe Storage

The cost of continuing care during safe storage at the reference PWR is judged to be substantially unaffected by whether or not the reactor has experienced an accident. Thus, the estimated annual costs of continuing care are updated from those presented in Section H.4. of Reference 1 and are shown in Table H.2-7. The total estimated annual cost during the safe storage period is almost $111,000.

H.2.3 Cost of Deferred Decontamination to Terminate SAFSTOR

The cost of deferred decontamination at the reference PWR can be estimated using the cost results reported in Reference 1, in the same manner as has been reported for possible decommissioning of TMI-2.[3] It is

TABLE H.2-6. Estimated Costs of Nuclear Liability Insurance and License Fees During Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident

Category	Unit Cost ($)	Total Cost ($)
Nuclear Liability Insurance	400 000/yr	600 000[a]
License Fees:[b]		
Facility License Amendment (Class IV)	12 300	12 300[c]
Routine Health, Safety, and Environmental Inspections	650/yr	1 300[d]
Routine Safeguards Inspection	11 800/yr	11 800[e]
Total Costs		625 400

(a) Prorated by quarters for the duration of the decommissioning project (i.e., ~1-1/2 years for preparations for safe storage).
(b) From 10 CFR 170.
(c) Based on one license amendment obtained prior to decommissioning to allow possession but not operation of the plant.
(d) Based on annual inspections for the duration of the decommissioning project (i.e., 2 annual inspections for preparations for safe storage).
(e) Based on having spent fuel onsite for 10.5 months (i.e., one yearly fee charged).

assumed that the ratio of deferred decontamination costs (after a specified span of safe storage) to DECON costs (in effect, immediate decontamination costs) 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 of this study. Based on this assumption, a comparison of the costs of DECON and deferred decontamination, both following normal shutdown and following a scenario 2 accident, is presented in Table H.2-8. The estimated cost of deferred decontamination following a scenario 2 accident is \$57.6 million after 30 years of safe storage or \$44.7 million after 100-year safe storage period. Applying the same assumption to SAFSTOR following the other accident scenarios considered in this study results in deferred decontamination costs of \$42.0 million and \$32.6 million after 30 and

TABLE H.2-7. Estimated Annual Costs During Safe Storage of the Reference PWR

Cost Category	Estimated Annual Costs ($ thousands)
Labor	62.23[a]
Equipment and Supplies	1.30[b]
Annual Allowance for Repairs	6.00[a]
Utilities and Services	8.50[c]
License Fee	0.65[d]
Nuclear Liability Insurance	10.00[e]
Subtotal	88.68
Contingency (25%)	22.17
Total Annual Cost	110.85

(a) Updated from Table H.4-1 of Reference 1 by a factor of 1.2 (see Section I.7 of Appendix I).
(b) Updated from Table H.4-1 of Reference 1 by a factor of 1.3 (see Section I.7 of Appendix I).
(c) Updated from Table H.4-1 of Reference 1 by a factor of 1.7 (see Section I.7 of Appendix I).
(d) From 10 CFR 170.
(e) Assumed cost.

TABLE H.2-8. Comparison of DECON and Deferred Decontamination Costs for the Reference PWR[a]

Decommissioning Activity	Following Normal Shutdown: Cost ($ millions)	Following Normal Shutdown: % of DECON Cost	Estimated Cost Following a Scenario 2 Accident ($ millions)
DECON	34.1[b,c]	100	67.8[d]
Deferred Decontamination After 30 Years	29.0[c,e]	85	57.6
Deferred Decontamination After 100 Years	22.5[c,e]	66	44.7

(a) Costs include 25% contingency.
(b) From Table 10.1-1 of Reference 1.
(c) Cost for facility demolition is deleted, $6.41 million + 25% contingency = $8.01 million.
(d) From Table H.1-1 of this study.
(e) From Table 10.4-1 of Reference 1.

100 years, respectively, of safe storage following a scenario 1 accident, and $90.3 million and $70.1 million after 30 and 100 years, respectively, following a scenario 3 accident.

H.2.4 Total SAFSTOR Costs

The total estimated costs of SAFSTOR at the reference PWR following a scenario 2 accident are summarized in Table H.2-9, based on the detailed cost estimates presented previously in Sections H.2.1 through H.2.3.

TABLE H.2-9. Total Costs of SAFSTOR at the Reference PWR Following a Scenario 2 Accident

	Estimated Costs ($ millions)	
Phase of SAFSTOR	30-Year Safe Storage	100-Year Safe Storage
Preparations for Safe Storage	16.7	16.7
Safe Storage (Continuing Care)	3.3	11.1
Deferred Decontamination	57.6	44.7
Total SAFSTOR Costs	77.6	72.5

H.3 ENTOMB COSTS

The estimated costs of ENTOMB at the reference PWR, following a scenario 2 accident and the subsequent accident cleanup campaign, are summarized and totaled in Table H.3-1. The total cost of ENTOMB is estimated at about $52.5 million, including a contingency of 25%. The major contributor to the total ENTOMB cost is staff labor at about 55% of the total. Waste management and energy costs add about 19% and 16%, respectively, to the total. Combined costs of special tools and equipment and of miscellaneous supplies make up almost 7% of the total. The remaining costs, representing less than 4% of the total, are attributed to specialty contractors, nuclear insurance, and license fees. Continuing care of the entombed plant is estimated to cost approximately $55,000 annually. Detailed cost information for the individual cost categories shown in the table is presented and discussed in the following subsections.

TABLE H.3-1. Summary of Estimated Costs of ENTOMB at the Reference PWR Following a Scenario 2 Accident

Cost Category	Estimated Costs ($ millions)		Percent of Total[a]
Staff Labor			
Management and Support Staff	6.112		
Decommissioning Workers	17.060		
Total Staff Labor Costs		23.172	55.1
Waste Management			
Neutron-Activated Materials	2.242		
Contaminated Materials	3.909		
Radioactive Wastes	1.652		
Total Waste Management Costs		7.803	18.6
Energy		6.524	15.5
Special Tools and Equipment		1.003	2.4
Miscellaneous Supplies		1.885	4.5
Specialty Contractors		0.521	1.2
Nuclear Insurance and License Fees		1.126	2.7
Subtotal		42.034	100.0
Contingency (25%)		10.508	
Total ENTOMB Costs		52.542	
Annual Continuing Care Costs		0.055	

(a) Costs adjusted to early 1981; the number of significant figures shown is for computational accuracy only.

Although no detailed cost estimates are made for ENTOMB following either a scenario 1 (less severe) or a scenario 3 (more severe) accident, the overall variation in ENTOMB costs with accident severity can be approximated using information presented in the following subsections. The total cost of ENTOMB following a scenario 1 or scenario 3 accident, including the 25% contingency, is estimated to be about $38.5 million or $79.6 million, respectively. These costs are about 75% and 150%, respectively, of the cost of ENTOMB after the postulated scenario 2 accident. The principal difference in the cost estimates for the three accident scenarios is in the decommissioning worker portion of the staff labor costs. Other significant differences are attributed to waste management and energy costs.

H.3.1 Costs of Staff Labor

The costs of staff labor during ENTOMB at the reference PWR, assuming the plant has experienced a scenario 2 accident, are shown in detail in Table H.3-2. Approximately 55% of the total ENTOMB cost is attributable to staff labor. A total staff labor cost of almost $23.2 million is estimated, not including specialty contractor labor. About 25% of the total is for management and support staff and the other 75% is for the decommissioning workers.

Based on the discussion of staff labor requirements presented in Section G.4.4 of Appendix G, the requirements and resulting costs for the management and support staff are judged not to vary significantly with changes in the postulated accident scenario. However, the decommissioning worker requirments are a function of accident consequences. Based on the information in Section G.4.4, the costs of the decommissioning workers and the overall staff labor costs are estimated to be about $6.8 million and $12.9 million, respectively, following a scenario 1 accident and approximately $39.3 million and $45.4 million following a scenario 3 accident. Thus, the total ENTOMB staff labor costs following a scenario 1 accident are about 55% of those following a scenario 2 accident which, in turn, are about 50% of those following a scenario 3 accident.

H.3.2 Costs of Waste Management

As is the case during DECON, three distinct types of radioactive waste materials require packaging, shipping, and offsite disposal during ENTOMB: 1) neutron-activated materials, 2) contaminated materials, and 3) radioactive wastes. The total waste management costs during ENTOMB at the reference PWR, following a scenario 2 accident and the subsequent accident cleanup campaign, are about $7.8 million, representing almost 19% of the total ENTOMB cost. The waste management costs include the container, transportation, and burial site costs, but do not include the direct labor costs for removing and packaging these materials.

TABLE H.3-2. Estimated Costs of Staff Labor During ENTOMB at the Reference PWR Following a Scenario 2 Accident

Position	Unit Cost for Labor[(a)] ($ thousands/man-year)	Total Staff Labor Required[(b)] (man-years)	Total Staff Labor Costs[(c)] ($ thousands)
Management and Support Staff			
Decommissioning Superintendent	89.4	4.5	402.3
Secretary	24.4	11.5	280.6
Clerk	24.4	6.4	156.2
Decommissioning Engineer	76.2	4.5	342.9
Assistant Decommissioning Engineer	52.6	4.2	220.9
Radioactive Shipment Specialist	39.5	2.7	106.7
Procurement Specialist	39.3	2.7	106.1
Tool Crib Attendant	27.8	5.4	150.1
Reactor Operator	34.8	21.7	755.2
Security Supervisor	55.9	2.7	150.9
Security Shift Supervisor	36.8	10.8	397.4
Security Patrolman	25.6	28.2	721.9
Contracts and Accounting Supervisor	47.1	3.0	141.3
Health and Safety Supervisor	60.5	3.0	181.5
Health Physicist	47.3	2.7	127.7
Protective Equipment Attendant	27.8	5.4	150.1
Industrial Safety Specialist	52.6	2.7	142.0
Quality Assurance Supervisor	52.6	3.0	157.8
Quality Assurance Engineer	47.3	2.7	127.7
Quality Assurance Technician	27.8	10.8	150.1
Consultant (Safety Review)	100.0	1.4	140.0
Instrument Technician	32.5	10.8	351.0
Maintenance Mechanic	32.5	10.8	351.0
Warehouseman	27.8	2.8	150.1
Subtotals		167.0	6 111.6
Decommissioning Workers			
Shift Engineer	52.6	5.4	284.0
Crew Leader	44.8	37.0	1 657.6
Utility Operator	32.5	92.2	2 996.5
Laborer	31.1	156.7	4 873.4
Craft Supervisor	47.3	10.8	510.8
Craftsman	32.5	113.4	3 685.5
Senior Health Physics Technician	39.5	16.3	643.9
Health Physics Technician	30.1	80.0	2 408.0
Subtotals		511.8	17 059.7
Totals		678.8	23 171.3

(a) Data from Table I.1-1 of Appendix I.
(b) Data from Table G.4-3 of Appendix G.
(c) Number of figures shown is for computational completeness only.

A breakdown of the costs and other pertinent parameters associated with waste management during ENTOMB is presented in Table H.3-3. The management of each of the three types of waste materials is discussed individually in the following subsections.

H.3.2.1 Neutron-Activated Materials

All of the neutron-activated materials in the reference PWR are contained in the reactor pressure vessel, the vessel internal structures, and in the surrounding steel and concrete biological shield. However, the vessel and the biological shield are assumed to be entombed in place and, thus, offsite disposal of neutron-activated materials during ENTOMB is limited to the vessel internals. Furthermore, during the accident cleanup campaign that precedes ENTOMB, it is postulated that certain of the vessel internals are removed to facilitate reactor defueling and that these are disposed of as accident cleanup wastes (see Appendix E). Therefore, requirements for the management of neutron-activated waste materials during post-accident ENTOMB are reduced from those during post-accident DECON, and are considerably less than those for DECON following normal shutdown.

The costs of waste management for neutron-activated materials during ENTOMB, presented in Table H.3-3, are a subset of those during DECON, presented previously in Table H.1-3. The total radioactivity expected to be present in the neutron-activated materials shipped offsite during ENTOMB is less than about 4.8 million curies. The packaged materials require about 75 truckload shipments to a shallow-land burial facility and occupy over 210 m^3 of space at the burial facility. The total estimated management cost of these materials following a scenario 2 accident is over $2.2 million.

As discussed previously in Section H.1.2.1, the waste management requirements for neutron-activated materials are not a function of accident severity except indirectly, because of the extent of vessel internals removal required during accident cleanup to defuel the reactor. Thus, the requirements and costs for management of these materials following a scenario 1 accident are the same as those following a scenario 2 accident. However, because additional internals are postulated to be removed following a scenario 3 accident, the

Tccident

Waste Category	Estimated Mass (kg)	Burial Site Costs(d,e) ling harge ($)	State Surcharge ($)	Liner Surcharge ($)	Curie Surcharge ($)	Total Waste Management Costs ($)
Neutron-Activated Steels	~163 250	080	2 250	103 420	476 870	2 241 940
Contaminated Equipment(f)	~883 860					
		210	20 480	--	--	1 027 380
Contaminated Concrete(f)	~5 626 000	200	47 680	--	--	2 881 280
Compactible, Combustible Trash(g)	51 500	710	1 140	--	--	350 320
Compactible, Noncombustible Trash(g)	166 500	750	3 710	--	--	200 000
Noncompactible Trash	506 000	440	9 280	--	--	498 290
Spent Resins	~12 000	350	310	11 160	8 490	111 270
Spent Filter Cartridges	~4 200	010	100	--	4 410	48 090
Solidified Evaporator Bottoms	~47 000	750	1 410	14 890	--	444 450
Totals	~7 460 000	500	86 360	129 470	489 770	7 803 020

(a) Based on information from Table I.2-1 of
(b) Based on information from Table I.2-2 of
(c) Based on information from Table I.3-4 of
(d) Charges for individual shipments may vary
(e) Based on information from Table I.4-1 of
(f) Estimates based on 47% of contaminated ma
(g) Values shown are for waste after treatmen

TABLE H.3-3. Estimated Costs of Radioactive Waste Management During PWR ENTOMB Following a Scenario 2 Accident costs of neutron-activated waste management are estimated to be reduced by more than half to about $900,000 following that accident.

H.3.2.2 Contaminated Materials

Contaminated materials in the reference PWR are assumed to include nearly all piping and equipment in the major site buildings, as well as significant areas of concrete surfaces in these buildings and in the onsite waste storage structures. The overall amount of contaminated material is discussed in more detail in Section H.1.2.2 for DECON. However, much of this contaminated material is located in or can be placed in the postulated entombment structure, thus significantly reducing the amount of this material requiring offsite disposal during ENTOMB.

As stated previously in Section G.4.1 of Appendix G, the contaminated materials originating within the entombment structure area in the containment building are entombed onsite. In addition, the entombment structure has the capacity to receive approximately 7250 m^3 of the contaminated materials originating outside of the structure. Based on the detailed breakdown of contaminated materials in the plant following normal shutdown presented in Tables G.4-4 and G.4-5 of Reference 1, with modifications as described in Section H.1.2.2 of this appendix to account for the effects of the postulated accident, the 7250 m^3 represents about 53% of the total contaminated material originating outside the entombment structure (almost 13,700 m^3). Thus, about 47% of this material, or nearly 6450 m^3, requires packaging, shipping, and offsite disposal. For the purposes of the cost estimate in this study, the percentage split between materials destined for entombment and those disposed of offsite is assumed to apply equally to both contaminated equipment and contaminated concrete.

The costs and other pertinent parameters associated with contaminated materials management during ENTOMB at the reference PWR following a scenario 2 accident are shown in Table H.3-3, presented previously. Offsite disposal of the contaminated materials is estimated to require 381 truckload shipments and 6430 m^3 of space at a shallow-land burial site. The total cost for this disposal is estimated to be about $3.9 million, over $1.0 million for contaminated equipment and almost $2.9 million for contaminated concrete.

Changes in accident severity are assumed to result in the same volume change of contaminated materials requiring offsite disposal during ENTOMB as during DECON, as discussed previously in Section H.1.2.2. The percentage change for ENTOMB is thus twice as much as for DECON because only about 1/2 as much material is shipped offsite during ENTOMB. Assuming the cost of contaminated material disposal for ENTOMB following a scenario 2 accident is decreased or increased 10% following a scenario 1 or scenario 3 accident, the corresponding total costs of management of contaminated waste materials are $3.5 million or $4.3 million, respectively.

H.3.2.3 Radioactive Wastes

The costs of management of radioactive wastes (i.e., trash, spent resins and filter cartridges, and solidified evaporator bottoms) during ENTOMB are estimated using the same assumptions as are applied to estimating these costs during DECON, as presented previously in Section H.1.2.3. The total cost of management of these wastes during ENTOMB is estimated to approach $1.7 million. Disposal at a shallow-land burial site is estimated to require 157 truckload shipments and over 1500 m^3 of burial space.

Assuming that changes in accident severity have only minor effects on these costs (no more than $\pm$10% from the scenario 2 values for scenario 1 or scenario 3), in the same manner as for DECON, the total costs associated with radioactive wastes following a scenario 1 or scenario 3 accident are approximately $1.5 million or $1.8 million, respectively.

H.3.3 Costs of Energy

Because energy costs during decommissioning are assumed to be a function of project duration, and because this duration is postulated to be the same for ENTOMB as for DECON, the energy cost during post-accident ENTOMB is estimated to be the same as that during post-accident DECON, as presented previously in Section H.1.3. Thus, following a scenario 1, scenario 2, or scenario 3 accident, ENTOMB energy costs are $6.2 million, $6.5 million, or $6.8 million, respectively.

H.3.4 Costs of Special Tools and Equipment

Based on information presented in Table D.3-1 of Appendix D and assuming that, as specified in the table, some of the equipment used for accident cleanup prior to ENTOMB is reused during the decommissioning, the estimated costs of special tools and equipment during post-accident ENTOMB are presented in Table H.3-4. The total estimated cost is just over $1.0 million, representing approximately 2.5% of the overall ENTOMB cost following a scenario 2 accident. Within the range of accident scenarios considered in this study, accident severity is assumed to have no significant impact on the requirements for and costs of special tools and equipment.

H.3.5 Costs of Miscellaneous Supplies

A variety of expendable supplies are used during ENTOMB, the estimated costs of which are presented in Table H.3-5. The cost estimates employ the same assumptions as those used for the corresponding DECON estimates, presented previously in Section H.1.5. The total estimated cost of miscellaneous supplies during post-accident ENTOMB is about $1.9 million, and is judged to be independent of the severity of the postulated accident.

H.3.6 Costs of Specialty Contractors

The requirements for and costs of specialty contractors during ENTOMB at the reference PWR are judged to be substantially the same whether or not the reactor has experienced an accident. On this basis, the specialty contractor costs (including environmental surveillance services) presented in Table 4.5-1 of Reference 2 are adjusted by a factor of 1.2 to update the costs to 1981. The total cost of specialty contractors during post-accident ENTOMB at the reference PWR is estimated to be about $520,000, regardless of the accident scenario considered.

H.3.7 Costs of Nuclear Insurance and License Fees

The estimated costs of nuclear liability insurance and license fees during ENTOMB are shown in Table H.3-6. These costs total over $1.1 million, representing close to 3% of the total cost for ENTOMB. These costs do not vary substantially with accident severity.

TABLE H.3-4. Costs of Special Tools and Equipment for ENTOMB at the Reference PWR Following a Scenario 2 Accident

Item	Estimated Unit Cost[(a)] ($ thousands)	Estimated Number Required[(b)]	Estimated Total Costs ($ thousands)
Underwater Plasma-Arc Torch	20	1	20
Underwater Oxyacetylene Torch	5	1	5
Portable Plasma-Arc Torch	20	4	80
Portable Oxyacetylene Torch	1	2	2
Guillotine Pipe Saw	4	10	40
Power-Operated Reciprocating Hacksaw	1	10	10
Underwater Lights and Viewing Aids	--	AR[(c)]	5 total
Underwater Tools	--	AR	25 total
Submersible Pump with Disposable Filters	2	4	8
Scaffolding and Safety Nets	--	AR	25 total
Mobile Chemical Decontamination Unit	20	5	100
Mobile Chemical Mixing and Heating Unit	5	4	20
Power-Operated Mobile Manlift	40	7	280
9100-kg Mobile Hydraulic Crane	28	1	28
9100-kg Forklift	28	4	112
Rigging Materials	--	AR	25 total
Concrete Drill with HEPA-Filtered Dust Collection System	2	4	8
Concrete Surface Spaller	5	4	20
Front-End Loader (light duty)	20	3	60
Vacuum Cleaner (HEPA-filtered)	1-5[(d)]	3	~12
Portable Ventilation Enclosure	2-10[(d)]	10	~75
Filtered Exhaust Fan Unit	5	4	20
Supplied-Air Plastic Suit	0.05	250	13
Polyurethane Foam Generator	5	2	10
Total Cost			1003

(a) Data from Table I.5-1 of Appendix I.
(b) Derived from Table D.3-1 of Appendix D, does not include items anticipated to be reusable from accident cleanup activities.
(c) AR = as required.
(d) Depends on size and complexity.

TABLE H.3-5. Estimated Costs of Miscellaneous Supplies During ENTOMB at the Reference PWR Following a Scenario 2 Accident

Item	Quantity	Total Cost[a] ($ thousands)
Decontamination Chemicals (EDTA/Oxalic Acid/Citric Acid)	9050 kg	15
Ion Exchange Resins	15 m^3	75
Filters	Unspecified	400
Protective Clothing	Unspecified	724
Cleaning Supplies	Unspecified	400
Expendable Tools and Materials	Unspecified	271
Total Costs		1885

(a) Rounded to the nearest $1000.

TABLE H.3-6. Estimated Costs of Nuclear Liability Insurance and License Fees During ENTOMB at the Reference PWR Following a Scenario 2 Accident

Category	Unit Cost ($)	Total Cost ($)
Nuclear Liability Insurance	400 000/yr	1 100 000[a]
License Fees:[b]		
Facility License Amendment (Class IV)	12 300	12 300[c]
Routine Health, Safety, and Environmental Inspections	650/yr	1 950[d]
Routine Safeguards Inspection	11 800/yr	11 800[e]
Total Costs		1 126 050

(a) Prorated by quarters for the duration of the decommissioning project (i.e., ∿2-3/4 years for ENTOMB).
(b) From 10 CFR 170.
(c) Based on one license amendment obtained prior to decommissioning to allow possession but not operation of the plant.
(d) Based on annual inspections for the duration of the decommissioning project (i.e., 3 annual inspections for ENTOMB).
(e) Based on having spent fuel onsite for 10.5 months (i.e., one yearly fee charged).

H.3.8 Costs of Continuing Care and Possible Deferred Decontamination

As discussed in Section G.4 of Appendix G, existing regulations require the licensee for the entombed plant to continue surveillance and maintenance of the entombment structure as a nuclear waste repository until such time as the structure is released for unrestricted use. Furthermore, a comprehensive radiation survey is required prior to such release to verify that the radioactive contamination either meets acceptable release limits or is removed from the site.

The costs of continuing care (i.e., maintenance and surveillance) of the entombment structure are assumed to be less than those of continuing care of the reference PWR in safe storage (see Section H.2.2). Based on information in References 2 and 3, these costs following ENTOMB are likely to be about 1/2 of the corresponding costs during the safe storage period of SAFSTOR. Thus, the annual continuing care costs following ENTOMB at the accident-damaged reference PWR are estimated to total about $55,000. These costs are assumed to be unaffected by the severity of the accident, within the range of accident scenarios considered in this study.

While selection of ENTOMB as the decommissioning alternative entails a commitment of the site for as long as required for the entombed radioactivity to decay to unrestricted release levels, there may be instances where earlier release of the site becomes imperative and decontamination of the entombed structure is necessary. Because of the many variables involved, no firm estimate of the costs of possible deferred decontamination of the entombment structure is made. However, deferred decontamination following ENTOMB is anticipated to be an extensive, time-consuming, and costly 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 the necessity to break into the entombment structure (designed to retain its integrity under any but the most severe conditions) and remove the more-or-less randomly placed radioactive materials stored inside. Therefore, the costs during deferred decontamination following ENTOMB are anticipated to be similar to those during deferred decontamination for SAFSTOR.[3,5] The costs of deferred decontamination for SAFSTOR are presented previously in Section H.2.3.

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. 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.

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.

4. U.S. Code of Federal Regulations, Title 10, Part 170, "Fees for Facilities and Materials Licenses and Other Regulatory Services, Under the Atomic Energy Act of 1954, As Amended" (10 CFR 170), U.S. Government Printing Office, Washington, D.C., 1981.

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.

APPENDIX I

COST ESTIMATING BASES

Cost data are presented that can be used to develop cost estimates for accident cleanup and for decommissioning. Categories for which basic cost data are presented include: labor, waste packaging, transportation, waste disposal, equipment, and services and supplies. The data presented are all early-1981 costs, whereas earlier decommissioning studies in this series used a 1978 cost base. The updating of costs from the 1978 to the 1981 cost base is discussed in Section I.7.

I.1 LABOR COSTS

Labor cost data for typical accident cleanup and decommissioning staff positions are given in Table I.1-1. The 1978 data base used in earlier decommissioning studies and referenced in the table has been adjusted by a factor of 1.2, based on building trades labor cost trends reported in the Handy-Whitman Index.[1] The base pay rates in Table I.1-1 are increased by 70% for nonunion employees and by 50% for union employees to account for owner costs such as fringe benefits, taxes, and insurance.

The labor costs shown are representative of average labor costs rather than labor costs for a particular decommissioning project at a given location. A recent decommissioning costs study[2] estimates that regional labor costs can deviate by as much as 17% from the national average. Costs at individual locations might deviate even more. In addition, the owner cost will depend on the values used to estimate fringe benefits, taxes, insurance, and other owner overhead expenses.

I.2 WASTE PACKAGING COSTS

The costs of packaging radioactive waste materials prior to shipment to a shallow-land burial site or other authorized waste repository include the shipping container cost, the cost of additional shielding provided by

TABLE I.1-1. Decommissioning Labor Cost Data

Position	Base Pay ($/yr)	Assumed Overhead Rate (%)	Cost ($/yr)	Reference[a]
Management & Support Staff				
Plant Superintendent	52 600	70	89 400	b
Asst. Plant Superintendent	44 900	70	76 200	b
Decommissioning Superintendent	52 600	70	89 400	b
Decommissioning Engineer	44 900	70	76 200	b
Asst. Decommissioning Engineer	31 000	70	52 600	b
Secretary/Word Processor/Clerk	16 200	50	24 400	e
Construction Engineering Supervisor	36 000	70	61 200	e
Construction Engineer	30 800	70	52 400	e
Estimator	27 600	70	46 900	e
Draftsman/Engineering Technician	20 000	50	30 000	e
Health & Safety Supervisor	35 500	70	60 500	e
Health Physicist	27 800	70	47 300	e
Industrial Safety Specialist	31 000	70	52 600	e
Radioactive Shipment Specialist	23 200	70	39 500	e
Contracts and Accounting Supervisor	27 700	70	47 100	b
Accountant	23 100	70	39 300	e
Contracts/Insurance/Procurement Specialist	23 100	70	39 300	e
Security Supervisor	32 900	70	55 900	e
Security Shift Supervisor	21 600	70	36 800	e
Security Patrolman	17 000	50	25 600	c
Quality Assurance Supervisor	31 000	70	52 600	e
Quality Assurance Engineer	27 800	70	47 300	e
Quality Assurance Technician	18 600	50	27 800	e
Plant Operations Supervisor	36 000	70	61 200	e
Plant Chemist	30 800	70	52 400	e
Chemist	27 600	70	46 900	e
Operations Engineer	30 800	70	52 400	e
Engineer	27 600	70	46 900	e
Operations Shift Supervisor	30 800	70	52 400	e
Senior Reactor Operator	27 600	70	46 900	e
Reactor Operator	23 200	50	34 800	e
Cleanup Superintendent	36 000	70	61 200	e
Warehouseman/Attendant (Tool Crib, Protective Equipment)	18 600	50	27 800	d
Consultant	100 000	—	100 000	e
Cleanup and Decommissioning Workers				
Shift Supervisor/Shift Engineer	31 000	70	52 600	e
Craft Supervisor	27 800	70	47 300	e
Craftsman/Instrument Technician/ Maintenance Mechanic	21 600	50	32 500	d
Crew Leader/Foreman	26 300	70	44 800	d
Utility Operator	21 600	50	32 500	d
Laborer/Power Plant Helper	20 800	50	31 100	c
Senior Health Physics Technician	23 200	70	39 500	e
Health Physics Technician/Safety Technician	20 200	50	30 100	d

(a) References for 1978 data base, which has been adjusted upward by a factor of 1.2 to update it to early-1981.
(b) U.S. Dept. of Labor, Bureau of Labor Statistics, Bulletin March 1975.
(c) R. S. Means Co., Building Construction Cost Data - 1975, 33rd Edition.
(d) Hanford Atomic Metal Trades Council Pay Scales.
(e) Author's estimate.

overpacks and casks, and the cost of a solidifying or dewatering agent for radioactive liquids or wet wastes. These costs are discussed in the following subsections.

I.2.1 Shipping Container Costs

The shipping containers assumed to be used for packaging radioactive materials for disposal are listed in Table I.2-1. Because of increases in labor and material costs, some container costs have increased significantly since 1978. Suppliers and users of these containers were consulted to obtain 1981 cost information.

TABLE I.2-1. Unit Costs of Shipping Containers for Radioactive Materials

Description	Burial Volume (m^3)	Estimated Unit Cost ($)
Standard Steel Drum 0.21 m^3, 23 kg empty	0.21	30
Small Steel Drum 0.11 m^3, 18 kg empty	0.11	20
Polyethylene Drum Liner	(a)	1
Fiberglassed Plywood Box 1.2 m x 1.2 m x 2.4 m, 175 kg empty	3.46	400
Fiberglassed Plywood Box Specially Fabricated	Variable	40/m^2 of Surface
Steel Cask Liner 0.63 m OD x 1.02 m high, 150 kg empty	0.33	500
Steel Cask Liner 1.38 m OD x 1.9 m high, 680 kg empty	2.84	2 000
Shielded Cask Liner 1.38 m OD x 1.9 m high	2.84	15 000
Stainless Steel Canister for Spent Fuel 0.35 m OD x 4.2 m high	0.40	6 000
Steel Box Specially Fabricated	Variable	275/m^2 of Surface

(a) Included in outer steel drum, no added burial volume.

I.2.2 Overpack and Cask Charges

Some packaged wastes with high surface dose rates require transport to a burial site in reusable overpacks or shielded casks. In general, it is more economical to rent such containers than to purchase them, especially the larger ones or those used infrequently or for a short time period. The overpacks and casks assumed for transportation of high activity or high surface dose rate decommissioning wastes are listed in Table I.2-2 together with physical characteristics and estimated rental charges.

TABLE I.2-2. Rental Charges for Reusable Shielded Casks

Description	Empty Weight (kg)	Daily Rental ($)
Truck Cask for Spent Fuel (1 PWR or 2 BWR Assemblies)	22 000	800
1.24 m OD x 1.56 m high 150-mm Pb thickness (B3 cask)	9 300	225
1.63 m OD x 2.34 m high 100-mm Pb thickness	16 300	300
1.95 m OD x 1.04 m high 50-mm Pb thickness (7D-3L cask)	7 000	225
1.4 m x 1.4 m x 6.1 m shielded autoloader for plywood boxes	16 400	300
2.44 m x 2.44 m x 6.10 m double-walled steel with fire-resistant insulation (Super Tiger)	6 800	300
IF-300 Spent Fuel Rail Cask (7 PWR or 18 BWR Assemblies)	120 000	4 000

I.2.3 Additional Shielding Costs

In some cases, additional lead shielding must be added to shipping containers to reduce surface radiation dose rates. The addition of this shielding is estimated to cost an average of $1.48/kg, including labor and energy, based on the 1978 estimate used in previous studies adjusted by a factor of 1.2.

I.2.4 Solidifying Agent Costs

The solidifying agents assumed to be used for packaging of wet solid and liquid wastes are listed in Table I.2-3 together with their respective costs.

TABLE I.2-3. Solidifying Agent Costs

Item	Estimated Unit Cost ($)
Cement (45-kg bag)	6/bag
Diatomaceous Earth (23-kg bag)	12/bag
Vinyl Ester Styrene (0.21-m^3 drum)	125/drum

I.3 TRANSPORTATION COSTS

Most radioactive wastes from cleanup and decommissioning operations are assumed to be transported to a disposal site by exclusive-use truck. The exception is the transport of spent fuel, which is assumed to be by rail. The transportation costs for both truck and rail shipments are discussed in the following subsections.

I.3.1 Shipment by Exclusive-Use Truck

Shipments of radioactive wastes to a shallow-land burial site or to an authorized waste repository are assumed to be by truck. Transportation costs for these shipments are based on the published rates of a carrier licensed to transport radioactive materials.[3] To compute transportation costs, the following assumptions are made:

- One-way shipping distance is 1600 km.
- Shipments not requiring casks or overpacks are separate one-way shipments destined for west of the Mississippi River (the highest rate category). Cask or overpack shipments are continuous excursion round-trips.
- A fuel surcharge is levied at a rate of 18%.[a]

(a) The fuel surcharge rate is subject to change as fuel prices increase or decrease. The 18% rate was in effect as of February 12, 1981.

- Where applicable, overweight charges are computed at the rate for the state of Washington, and regulations and conditions governing overweight and oversize shipments in the state of Washington are assumed.

A trend that could add significantly to future nuclear transportation costs is the requirement by state and local governments for permits in advance of each radioactive material shipment through their jurisdiction. A major carrier plans to charge its customers $25, plus the cost of the permit, for each such permit required.(4) In the future, these permit charges could be substantial for long-distance shipments. However, no such permit charges are included in the transportation cost estimates of this study.

The rate schedule for truck shipments of legal size and weight that forms the basis for transportation costs in this study is shown in Table I.3-1. The gross vehicle weight (GVW) for legal-weight shipments by truck is assumed to be less than 21.32 Mg. The maximum allowed GVW is assumed to be 38.55 Mg.(3) Overweight charges by states vary widely. The additional charges assumed in this study to be levied by the carrier and the state for overweight shipments are shown in Table I.3-2.

Oversize (as well as overweight) shipments may be required in certain instances. Table I.3-3 summarizes the applicable requirements for oversize shipments on two-lane highways. The oversize shipments assumed in this study are estimated to cost $1000/shipment more than legal-size shipments of the same weight. This additional cost covers the expense of special permits and escort cars.

Example shipping costs, calculated for several different payloads and for one-way and round-trip shipments, are shown in Table I.3-4. For a one-way 1600-km shipment, the base charge is that shown in Column 2 of Table I.3-1. To this must be added the 18% fuel surcharge, any applicable overweight charges shown in Table I.3-2, and any applicable oversize costs.

Casks and overpacks are assumed to be picked up loaded at the site of accident cleanup and decommissioning operations, delivered to the disposal site to be unloaded, and then returned to the original site. Thus, each

TABLE I.3-1. Transportation Rates for Legal-Size and -Weight Shipments[a,b,c]

Kilometers One-way (Not Over)	Rate in Cents/Kilometer Column 1[d]	Column 2[e]	Column 3[f]	Kilometers One-Way (Not Over)	Rate in Cents/Kilometer Column 1[d]	Column 2[e]	Column 3[f]
160	233	244	168	1200	86	103	71
200	214	226	155	1280	82	100	71
240	196	209	143	1360	81	99	71
280	179	192	133	1440	80	98	71
320	155	169	121	1520	79	97	71
360	147	162	115	1600	77	95	71
400	141	156	108	1760	77	94	71
440	134	150	101	1920	77	94	71
480	128	144	96	2080	77	93	71
520	125	141	91	2240	77	92	71
560	121	137	88	2400	77	92	71
600	116	132	84	2560	77	91	71
640	111	128	82	2720	77	91	71
680	108	124	80	2880	77	90	71
720	102	119	78	3040	77	89	71
760	100	117	76	3200	77	89	71
800	96	114	75	3360	77	88	71
880	94	111	73	3520	77	88	71
960	92	109	71	3680	77	87	71
1040	89	106	71	3840	77	86	71
1120	87	104	71	4000 and Beyond	77	86	71

(a) Reproduced from the published rates of a carrier[3] licensed to transport radioactive materials.
(b) Effective August 15, 1980.
(c) Rates do not include a fuel surcharge, which amounted to 18% of the base rate as of February 13, 1981.
(d) Column 1 rates applicable to one-way shipments having a destination east of the Mississippi River.
(e) Column 2 rates applicable to one-way shipments having a destination west of the Mississippi River.
(f) Column 3 rates apply to continuous excursion moves in which a subsequent shipment is made available to the carrier within 24 hours after arrival at the point of loading or unloading.

TABLE I.3-2. Additional Charges When Gross Vehicle Weight Exceeds 21.32 Mg[a,b]

Gross Vehicle Weight (Mg)	State Surcharge ($)	Carrier Surcharge ($)	Total Overweight Surcharge ($)
21.32 to 23.12	10 + 0.031/km	0.131/km	10 + 0.162/km
23.13 to 25.84	10 + 0.062/km	0.131/km	10 + 0.193/km
25.85 to 28.56	10 + 0.093/km	0.131/km	10 + 0.224/km
28.57 to 31.28	10 + 0.155/km	0.131/km	10 + 0.286/km
31.29 to 34.00	10 + 0.218/km	0.131/km	10 + 0.349/km
34.01 to 36.72	10 + 0.280/km	0.131/km	10 + 0.411/km
36.73 to 38.55	10 + 0.373/km	0.131/km	10 + 0.504/km

(a) State surcharge is based on rates for the state of Washington.
(b) Carrier surcharge is based on the published rates[3] of a carrier licensed to transport radioactive materials.

TABLE I.3-3. Requirements for Oversize Truck Shipments[a]

Characteristic Dimension of Vehicle/ Load Combination	Special Permit Required in Excess of:	Escort Car Required in Excess of:	Maximum Allowed Dimensions
Width	2.44 m (8 ft)	3.05 m (10 ft)	4.27 m (14 ft)
Height	4.11 m (13.5 ft)	--[b,c]	--[b,c]
Length	19.81 m (65 ft)	30.48 m (100 ft)	--[b]

(a) Based on regulations in the state of Washington for two-lane highways. See Reference 5.
(b) No specific requirement, but escort car may be required at discretion of Highway Department.
(c) Heights exceeding 4.42 m (14.5 ft) are generally considered unacceptable because of the special routing and preparations required.

3200-km round trip consists of two 1600-km one-way moves, with charges based on the continuous excursion rates shown in Column 3 of Table I.3-1. From the reference rate schedule, the basic charge for the round trip is $2272. With the additional 18% fuel surcharge, this is increased to $2681. Applicable overweight charges must also be added. To ensure rapid turnaround on these shipments and to minimize cask rental charges, a second driver is assumed to be used, costing an additional $0.093/km.[3]

TABLE I.3-4. Example Shipping Costs of Truck Shipments

Status	Number of Drivers	Payload (Mg)	GVW (Mg)	Cost ($)
Legal weight, one-way[a]	1	8.61	21.31	1794
Overweight, one-way[a]	1	19.95	32.65	2362
Oversize and overweight, one-way[a]	1	19.95	32.65	3362
Overweight, one-way[a]	1	25.85	38.55	2610
Overweight, round-trip[b]	2	19.95	32.65	4105
Overweight, round-trip[b]	2	25.85	38.55	4601

(a) 1600-km distance.
(b) Shipments involving casks or overpacks, with overweight charges applicable both directions. Charges computed on the basis of two 1600-km trips.

I.3.2 Shipment by Rail

Shipment by rail is assumed for the spent fuel removed from the reactor core during accident cleanup. Assuming a round-trip distance of 3200 km, the shipping cost is estimated to be about $80/Mg. This amounts to about $18,250 for a rail car carrying a GE IF-300 cask.[6]

I.4 WASTE DISPOSAL COSTS

A basic assumption of this study is that nearly all of the radioactive material resulting from cleanup and decommissioning of the reference reactor can be disposed of by burial at a commercial shallow-land burial facility. The only exceptions are the undamaged spent fuel, which is assumed to be placed in extended storage at an independent spent fuel storage installation (ISFSI), and the high-activity waste from accident-water processing and the damaged fuel assemblies and fuel core debris, which are assumed to be placed in interim storage at a federal repository. The unit costs of waste disposal are given in the following subsections.

I.4.1 Shallow-Land Burial

The shallow-land burial costs used in this study are based on a November 1980 price list from U.S. Ecology, Inc.,[7] which operates burial sites at Richland, Washington, and Beatty, Nevada. These prices are comparable to those charged by Chem-Nuclear Services, Inc.,[8] at their Barnwell, South Carolina disposal site. Burial ground charges are shown in Table I.4-1.

I.4.2 Disposal of Wastes at a Federal Repository

At the present time, only shallow-land burial grounds are available for the disposal of commercial radioactive wastes. As explained in Sections 5.3.3 and D.5.2, some wastes from the post-accident cleanup and decommissioning of a light water reactor (LWR) may not meet the acceptance criteria set forth in 10 CFR Part 61[9] for disposal by shallow-land burial. No regulatory framework has yet been developed to specifically address the disposal of wastes that are not acceptable for near-surface disposal. Accordingly, the disposition of these wastes may have to be determined on a case-by-case basis. Under the terms of a Memorandum of Understanding[10] between the NRC and the DOE, DOE has agreed to assume responsibility for the storage and disposal of the damaged fuel core and other highly radioactive wastes from decontamination activities at TMI-2. The costs of disposition will ultimately be determined under an agreement to be negotiated between DOE and the owner.

Since a high-level waste repository does not presently exist, in this study, the high-activity wastes resulting from processing of the accident water and the damaged fuel assemblies and fuel core debris removed from the reactor during post-accident cleanup are assumed to be sent to a federal repository for storage and disposal. Storage and disposal costs at a federal repository have not been established at the time this report is being written. A recent study[11] of DOE's spent fuel program gives \$234/kg U as the estimated unit cost of disposal of spent fuel at a federal repository. This unit cost is the basis for the estimated spent fuel disposal costs given in this study. The disposal cost of a PWR assembly (461 kg U) is estimated to be about \$108,000 and the disposal cost of a BWR assembly (189 kg U) is estimated to be about \$44,000.

Estimated storage costs of other wastes postulated to be sent to a federal repository are chosen to be consistent with the spent fuel costs given above. Wastes from accident-water processing are assumed to be packaged in 0.3-m^3 cask liners for which estimated interim storage costs are \$2500/liner. Evaporator bottoms and irradiated hardware are assumed to be packaged in 2.85-m^3

TABLE I.4-1. Commercial Shallow-Land Burial Charges[a,b]

I. DISPOSAL CHARGES, NON-TRU WASTE

A. Steel Drums, Wood Boxes

Container Surface Dose Rate (R/hr)[c]			Price/Unit Volume ($/m^3)
0.00	to	0.20	307.20
0.201	to	1.00	335.45
1.01	to	2.00	376.05
2.01	to	5.00	459.05
5.01	to	10.00	542.00
10.01	to	20.00	702.65
20.01	to	40.00	870.40
40.01	to	60.00	1332.95
60.01	to	80.00	1601.30
80.01	to	100.00	1765.50
	>100		by request

B. Disposable Liners

Container Surface Dose Rate (R/hr)[c]			Surcharge/Liner ($)	Price/Unit Volume ($/m^3)
0.00	to	0.20	None	307.20
0.201	to	1.00	119.00	307.20
1.01	to	2.00	292.00	307.20
2.01	to	5.00	411.00	307.20
5.01	to	10.00	594.00	307.20
10.01	to	20.00	758.00	307.20
20.01	to	40.00	941.00	307.20
40.01	to	60.00	1116.00	307.20
60.01	to	80.00	1288.00	307.20
80.01	to	100.00	1463.00	307.20
	>100		by request	by request

II. SURCHARGES

A. State of Washington Surcharge:	$10.60/m^3
B. Curie Surcharge (per load):	
Less than 100 curies	No charge
101 to 300 curies	$660.00
301 to License Limits (i.e., 50,000 Ci)	$660.00 + $0.09/Ci
C. Handling Surcharge	
0 - 4.54 Mg	No charge
>4.54 Mg	$87.50 + $0.044/kg over 4.54 Mg
Special Equipment	By special quotation
D. Cask Handling Fee:	$335.00 per cask

(a) Reproduced from the published rates[7] of a licensed burial ground operator.
(b) Prices effective November 17, 1980.
(c) Maximum reading at container surface, irrespective of physical size or configuration.

steel liners for which estimated interim storage costs are $10,000/liner. The fuel core debris is assumed to be packaged in stainless steel canisters costing $25,000/canister to store.

I.5 EQUIPMENT COSTS

Equipment costs from the 1978 data base have been reviewed and updated as appropriate to reflect 1981 costs. Costs of construction-type items (hoists, cranes, lifts, etc.) are based on costs shown in the 1981 catalog of building construction costs published by the R. S. Means Company.[12] Equipment costs are shown in Table I.5-1.

I.6 SERVICES AND SUPPLIES

Various types of services and supplies are required for accident cleanup and decommissioning. The estimated unit costs of the major items are discussed here.

Electricity

A principal services cost item is electric power. Costs of electric power vary widely with location and usage rate. In this study, a unit wholesale cost of $0.025/kWh, or $25/MWh, is assumed for electricity.

Fuel Oil

Another energy service cost item is fuel oil. A unit cost of $264/m^3 ($1.00/gal) is assumed for fuel oil.

Decontamination Chemicals

The unit costs of the chemicals used for the EDTA/oxalic/citric acid solution for the decontamination of internal surfaces of the reactor coolant system are estimated to be:

- EDTA - $1.30/kg
- Oxalic Acid - $1.74/kg
- Citric Acid - $1.83/kg.

For a mixture of these three chemicals, one-third each by weight, the cost is $1.62/kg.

TABLE I.5-1. Special Tools and Equipment

Item	Estimated Unit Cost ($ thousands)
Underwater Manipulator	1000
Underwater Plasma-Arc Torch	20
Underwater Oxyacetylene Torch	5
Arc Saw	120
Portable Plasma-Arc Torch	20
Portable Oxyacetylene Torch	1
Guillotine Pipe Saw	4
Power-Operated Reciprocating Hacksaw	1
Nibbler	1
Closed Circuit TV System	10-100[a]
Submersible Pump with Disposable Filter	2
High-Pressure Water Jet	20
Mobile Chemical Decontamination Unit	20
Mobile Chemical Mixing & Heating Unit	5
Powered Floor Scrubber	0.3
Wet-Dry Vacuum Cleaner (HEPA Filtered)	1-5[a]
Supplied-Air Plastic Suit	0.05
Respirator Facepiece	0.1
Shielded Vehicle with Manipulator Arms and Interchangeable Tools	120
Power-Operated Mobile Manlift	40
9100-kg Mobile Hydraulic Crane	28
9100-kg Forklift	28
Concrete Drill with HEPA Filtered Dust Collection System	2
Concrete Surface Spaller	5
Front-End Loader (Light Duty)	20
Portable Filtered Ventilation Enclosure	2-10[a]
Filtered-Exhaust Fan Unit	5
Blasting Mat	0.5
Polyurethane Foam Generator	5
Paint Sprayer	0.5-1[b]
Disposable Ion Exchange Liners	5
HEPA Filter	0.2
Roughing Filter	0.1
Waste Compactor	12
Incinerator	100-300[b]

(a) Depends on size and complexity.
(b) Depends on capacity of system.

The unit costs of the chemicals used to make up the oxalic-peroxide-gluconic (OPG) solution are estimated to be:

- Oxalic Acid - $1.74/kg
- Hydrogen Peroxide - $2.14/kg
- Gluconic Acid - $2.43/kg
- Sodium Gluconate - $1.01/kg.

For the OPG solution of specified concentration (see Section E.4.1 for the chemical composition of OPG solution), the total unit cost for chemicals is $56.40/m^3 of solution.

Ion Exchange Resins

The disposable ion exchange liners used in the submerged demineralizer system are estimated to cost $5000 each, including the zeolite resins, the canister, and the necessary hardware to seal the unit for disposal. For the other ion exchange resins required, an average unit cost of $5000/m^3 is assumed.

I.7 COST UPDATING FROM 1978 TO 1981 COST BASE

As noted previously, the cost data used in this study are all early-1981 costs, while earlier decommissioning studies in this series used a 1978 cost base. To facilitate comparisons between the costs reported in this study and costs presented in previous studies, appropriate factors for adjusting costs from the original 1978 data base to the 1981 base are presented by cost category in Table I.7-1. These cost updating factors are based on an analysis of cost indices and other measures of actual cost escalations over the period in question. The cost updating factors are rounded to two significant figures.

The unit cost information in this study is developed from the same sources as the unit cost information in previous decommissioning studies and, thus, the cost updating factors presented in Table I.7-1 are based on cost escalations shown by these sources. Actual cost escalations during the period are likely to vary from area to area. In addition, different sources of information may report somewhat different values for cost escalations over the same period. Therefore, care should be taken to ensure the use of appropriate data in escalating costs for any specific project.

TABLE I.7-1. Decommissioning Cost Updating Factors: 1978 to 1981

Cost Category	Adjustment Factor
Staff Labor	1.2[a]
Waste Management	
Container Costs	1.3[b]
Transportation Costs	1.4[c]
Burial Site Costs	3.0[d]
Energy	
Electricity	1.7[b]
Fuel Oil	2.0[b]
Special Tools and Equipment	1.3[b]
Miscellaneous Supplies	1.3[b]
Specialty Contractors	1.2[a]
Nuclear Insurance	1.4[b]
License Fees	1.0[e]

(a) Based on labor cost data from Reference 1.
(b) Author's estimate, based on conversations with suppliers or users.
(c) Based on rates of a carrier licensed to transport radioactive materials, as reported in Reference 3.
(d) Based on actual price lists of a licensed shallow-land burial site; see Reference 7.
(e) License fees are set forth in Reference 13.

REFERENCES

1. Handy-Whitman Index of Public Utility Construction Costs, to July 1, 1980, Bulletin No. 112, Whitman, Requart and Associates, Baltimore, Maryland, 1980.

2. W. J. Manion and T. S. La Guardia, Decommissioning Handbook, DOE/EV/10128-1, Nuclear Energy Services, Inc., Danbury, Connecticut, November 1980.

3. MF-I.C.C. No. 195, Local Commodity Tariff No. 1045-E, Tri-State Motor Transit Co., Joplin, Missouri, August 1980.

4. S. A. Dupree and L. C. O'Malley, Economics of Radioactive Material Transportation in the Light-Water Reactor Nuclear Fuel Cycle, SAND80-0035, Sandia National Laboratories, Albuquerque, New Mexico, October 1980.

5. Regulations and Conditions Governing Overweight and Over-Dimension Vehicle Permits, Washington State Highway Commission, Department of Highways, Olympia, Washington, September 1974.

6. Technology for Commercial Radioactive Waste Management, DOE/ET-0028, Vol. 4, Section 6.2, U.S. Department of Energy, Washington, D.C., May 1979.

7. Washington Nuclear Center and Nevada Nuclear Center Schedule of Charges, U.S. Ecology, Inc., November 17, 1980.

8. Barnwell Low-Level Radioactive Waste Disposal Facility Rate Schedule, Chem-Nuclear Services, Inc., January 1, 1981.

9. "Proposed Rule 10 CFR Part 61: Licensing Requirements for Land Disposal of Radioactive Waste," Federal Register, Vol. 46, No. 142, pp. 38081-38100, July 24, 1981.

10. "Memorandum of Understanding Between the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy Concerning the Removal and Disposition of Solid Nuclear Wastes from Cleanup of the Three Mile Island Unit 2 Nuclear Plant," March 15, 1982.

11. Department of Energy Report on Fee for Spent Nuclear Fuel Storage and Disposal Services, DOE/SR-0006, U.S. Department of Energy, Washington, D.C., October 1980.

12. R. S. Godfrey, ed., Building Construction Cost Data - 1981, 39th Edition, R. S. Means, Co., Inc., Kingston, Massachusetts, 1980.

13. U.S. Code of Federal Regulations, Title 10, Part 170, "Fees for Facilities and Materials Licenses and Other Regulatory Services Under the Atomic Energy Act of 1954, As Amended" (10 CFR 170), U.S. Government Printing Office, Washington, D.C., 1981.

APPENDIX J

SAFETY ASSESSMENT DETAILS

The purpose of this appendix is to quantify the parameters and define the methodology for estimating the impacts to public and occupational safety from post-accident cleanup and decommissioning of the reference PWR power station. Radiological and nonradiological impacts of both routine activities and selected generic industrial and transportation accidents during post-accident cleanup and decommissioning are evaluated.

The following sections contain detailed discussions of the technical approach to safety assessment and of the safety impacts resulting from accident cleanup and from decommissioning. A summary of this information is given in Chapter 14 of Volume 1.

A basic assumption of the analyses presented in this appendix is that the radioactive waste materials from accident cleanup and decommissioning are shipped offsite for disposal at the time of decommissioning. The safety impacts of alternate scenarios for waste disposal are addressed in Chapter 15 of Volume 1.

J.1 TECHNICAL APPROACH

To estimate the safety impacts from post-accident cleanup and decommissioning of the reference PWR, the following basic assumptions are made:

1. Appropriate radiation protection and contamination control techniques are applied to conform with the principle of keeping occupational radiation doses and radioactivity levels in effluents as low as reasonably achievable (ALARA).
2. The analysis of public safety impacts resulting from the release of radioactive materials during accident cleanup is based largely on information developed in Reference 1 concerning the cleanup of

TMI-2, with appropriate adjustments to account for differences in assumed fuel burnup and accident severities for the reference PWR.

3. The assessments of the safety impacts from post-accident decommissioning use information pertaining to the decommissioning of the reference PWR following normal shutdown, developed in Appendix J of Reference 2, to the maximum extent possible. Appropriate adjustments are made to account for differences between post-accident and normal-shutdown radionuclide inventories and decommissioning requirements.

4. The integrity of the containment building is maintained until all radioactive materials above unrestricted release levels are either removed or adequately confined.

5. The spent fuel removed from the reactor core during accident cleanup is assumed to be shipped from the site during the first 10-1/2 months of the decommissioning activities that follow completion of accident cleanup. Fuel handling accidents, except for transportation accidents, are covered in Reference 1 and in the FSAR for the reference PWR[3] and are not considered further in this study because the consequences of such accidents are significantly less than the consequences of accidents that are included in the analyses in this appendix.

6. HEPA filters in the plant ventilation systems are tested in place on a regular basis and replaced as required. The measured particle collection efficiency of these filters is 99.95%.[4] Atmospheric releases of radioactivity are assumed to pass through a single HEPA filter with a transmission factor of 5×10^{-4}.[4]

7. Unneeded hazardous chemicals and equipment are removed from the plant after the reference PWR is stabilized following the postulated reactor accident. Decontamination agents such as phosphoric acid, ethylenediaminetetraacetic acid, oxalic acid, and citric acid are available in the plant. Unneeded ion exchange resins and resin beds are removed.

8. In areas with high levels of radioactive contamination, a temporarily installed "greenhouse," or contamination control envelope, is assumed to be used. The contamination control envelope is assumed to be vented through a HEPA filter with a transmission factor of 5×10^{-4} to reduce the airborne radionuclide concentrations in the PWR buildings from selected accident cleanup and decommissioning operations.

9. The leakage rate from the contamination control envelope is assumed to be a function of the operations involved and the time at which the operation occurs. The leakage rate is assumed to vary between 0.1 and 10% of the airborne concentrations within the contamination control envelope.[2]

10. The airborne concentrations of dust or liquid droplets are assumed to be 1×10^{-2} g/m^3, equal to the concentrations observed at the Elk River reactor decommissioning.[5,6] For tasks involving blasting or explosions, the airborne concentrations are assumed to be a factor of 10 higher, or 1×10^{-1} g/m^3.[7]

11. All offsite radioactive waste shipments are assumed to be in accordance with Department of Transportation (DOT) regulations as described in Section D.5 of Appendix D. Spent fuel is assumed to be shipped by rail and the other radioactive materials are assumed to be shipped by truck. The one-way shipping distance in either case is 1600 km.

12. Radiation doses to the maximum-exposed individual and to the population residing within 80 km of the reference site are calculated using the environmental data and assumptions discussed in Appendix E of Reference 2. These methods are consistent with those outlined in Regulatory Guide 1.109.[8]

Other assumptions relating to specific accident cleanup and decommissioning tasks are discussed where they apply to the analysis.

J.2 ACCIDENT CLEANUP

The 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 appendix, accident cleanup is addressed separately from decommissioning.

This section contains the detailed analysis of the safety impacts resulting from accident cleanup activities. The radiological and nonradiological impacts of both routine activities and selected generic industrial and transportation accidents are considered. Radiological safety impacts to the public are assessed in Section J.2.1. Occupational safety impacts of accident cleanup are discussed in Section J.2.2. Transportation safety impacts, both public and occupational, are addressed in Section J.2.3.

J.2.1 Public Safety Aspects of Accident Cleanup

The public safety impacts of onsite activities during accident cleanup are discussed in the following subsections. Public radiation doses from atmospheric releases that result from routine tasks and from postulated industrial accidents during accident cleanup are considered. Nonradiological safety impacts to the public from onsite activities are judged to be negligible and are not considered further. Public safety impacts from offsite shipment of radioactive waste materials during accident cleanup are included in the assessment of transportation safety impacts, presented in Section J.2.3.

During accident cleanup, the routine tasks and the postulated industrial accidents can generate airborne radioactivity in the plant, primarily in the form of solid particulates and/or suspended liquid droplets. The airborne radionuclide concentration depends on the particular task or accident considered and on the corresponding radionuclide inventory at the location involved. (The post-accident radionuclide inventories in the reference PWR are discussed in detail in Appendix C.) Contamination control measures, where applied, and HEPA filters in plant ventilation systems reduce the levels of radioactivity in the air leaving the plant.

In the following subsections, the atmospheric releases and corresponding radiation doses to the public during accident cleanup at the reference PWR are discussed. The atmospheric releases are estimated by determining the realistic maximum atmospheric release for each task or industrial accident and then using this value whenever the particular release situation occurs, even for areas with lower levels of radioactive contamination.

J.2.1.1 Public Radiation Doses from Routine Tasks During Accident Cleanup

A complete discussion of the tasks required for accident cleanup of the reference PWR is contained in Appendix E. Details are given for each of the three reactor accident scenarios considered in this study. To quantify the radiation doses to the public that result from these tasks, atmospheric releases of radioactivity are estimated for the particular radionuclide inventories involved, and the resulting doses to the maximum-exposed individual and to the population are calculated.

The atmospheric releases for accident cleanup at the reference PWR are based on estimated values for releases from equivalent activities during cleanup of TMI-2, as reported in Reference 1. The release values used for this study are adjusted from those reported for TMI-2 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. These adjustments are based on the post-accident radionuclide inventories existing at TMI-2, as reported in Reference 1, and on the post-accident radionuclide inventories postulated to exist at the reference PWR following each of the three reactor accident scenarios, presented and discussed in Appendix C of this study.

The radiation doses to the public from these releases are calculated using the dose models discussed in Appendix E of Reference 2, in conjunction with characteristics of the reference site described in Appendix A of this report. Dose conversion factors are used that are appropriate for the post-accident radionuclide inventories at the reference PWR. Each of the atmospheric releases is assumed to be a chronic release (i.e., one that occurs at a uniform rate for a period of 1 year) to allow direct comparisons of the impacts of individual accident cleanup tasks. The first-year doses and the fifty-year committed dose equivalents to both the maximum-exposed individual and to the population residing within 80 km of the site are calculated for each accident cleanup task. The calculated doses include direct exposure, inhalation, and ingestion pathways; radiation doses from air submersion are not calculated since they have been shown to be insignificant in Reference 2.

The estimated atmospheric releases of radioactivity and the resulting doses to the maximum-exposed individual from routine tasks during accident cleanup following a scenario 1, scenario 2, or scenario 3 accident are shown in Tables J.2-1, J.2-2, and J.2-3, respectively. The releases and the resulting doses to the population residing within 80 km of the site from these tasks following a scenario 1, scenario 2, or scenario 3 accident are shown in Tables J.2-4, J.2-5, and J.2-6, respectively. The releases and resulting doses following a scenario 1 accident are about an order of magnitude lower than those following a scenario 2 accident which, in turn, are about half of those following a scenario 3 accident.

The doses shown in Tables J.2-1 through J.2-6 are not totalled 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.[9]

The atmospheric releases and resulting public radiation doses presented in this study for accident cleanup are based on cleanup activities in the containment building of the reference PWR. 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 Appendix E). The releases and resulting public doses from accident cleanup activities in these other buildings are judged to be insignificant in comparison to those from accident cleanup activities in the containment building and are not considered further.

TABLE J.2-1. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases Resulting from Routine Tasks During Post-Accident Cleanup After the PWR Scenario 1 Accident[a]

Cleanup Tasks	Reference Radionuclide Inventory[b]	Annual Atmospheric Release (Ci/yr)[c]	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
			Total-Body	Bone	Lung	GI-LLI	Total-Body	Bone	Lung	GI-LLI
Preparations for Accident Cleanup										
• Vent Containment	^{85}Kr	1.6×10^{3}	5.9×10^{-6}	5.9×10^{-6}	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 Decon. of Containment										
• Remote Washdown, High Pressure Spray, Hands-On Decon., Install Local Shielding	Accident Scenario 1	8.0×10^{-6}	7.9×10^{-9}	6.2×10^{-9}	2.6×10^{-9}	1.8×10^{-9}	1.2×10^{-8}	9.0×10^{-9}	3.3×10^{-9}	1.8×10^{-9}
• ^{3}H Evaporation Loss	^{3}H	1.0×10^{3}	1.4×10^{-3}	--	1.4×10^{-3}	1.3×10^{-3}	1.5×10^{-3}	--	1.5×10^{-3}	1.4×10^{-3}
Defueling the Reactor										
• Remove Reactor Pressure Vessel Head, Internals, and Core	^{3}H	9.0×10^{2}	1.3×10^{-3}	--	1.3×10^{-3}	1.2×10^{-3}	1.4×10^{-3}	--	1.4×10^{-3}	1.3×10^{-3}
• Vent ^{85}Kr	^{85}Kr	6.2×10^{0}	2.3×10^{-8}	2.3×10^{-8}	2.3×10^{-8}	2.3×10^{-8}	2.3×10^{-8}	2.3×10^{-8}	2.3×10^{-8}	2.3×10^{-8}
Cleanup of Primary Coolant System										
• OPG Solution	--[d]	--								
• EDTA Solution	^{3}H	6.2×10^{2}	8.7×10^{-4}	--	8.7×10^{-4}	8.1×10^{-4}	9.3×10^{-4}	--	9.3×10^{-4}	8.7×10^{-4}
• Immobilization of Decon. Solutions	Accident Scenario 1	8.8×10^{-8}	8.7×10^{-11}	6.9×10^{-11}	2.9×10^{-11}	2.0×10^{-11}	1.3×10^{-10}	1.1×10^{-10}	3.6×10^{-11}	2.0×10^{-11}
Immobilization of Process Solid Wastes	Accident Scenario 1	1.8×10^{-5}	1.8×10^{-8}	1.4×10^{-8}	5.9×10^{-9}	4.1×10^{-9}	2.7×10^{-8}	2.2×10^{-8}	7.4×10^{-9}	4.1×10^{-9}
Packaging and Handling Solid Wastes	Accident Scenario 1	4.0×10^{-5}	4.0×10^{-8}	3.1×10^{-8}	1.3×10^{-8}	9.2×10^{-9}	6.0×10^{-8}	4.8×10^{-8}	1.6×10^{-8}	9.2×10^{-9}
Process Decon. Liquid Wastes	Accident Scenario 1	7.7×10^{-5}	7.6×10^{-8}	6.0×10^{-8}	2.5×10^{-8}	1.8×10^{-8}	1.2×10^{-7}	9.2×10^{-8}	3.2×10^{-8}	1.8×10^{-8}

(a) Doses are not totalled 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.
(b) The radionuclide inventories used in the dose calculations are discussed in Appendix C.
(c) For comparison purposes, dose calculations are based on a continuous release for 1 year. These releases are estimated by adjusting the releases from accident scenario 2 for reduced radionuclide release fractions.
(d) No release is calculated for the OPG solution cleanup.

TABLE J.2-2. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases Resulting from Routine Tasks During Post-Accident Cleanup After the PWR Scenario 2 Accident[a]

Cleanup Tasks	Reference Radionuclide Inventory[b]	Annual Atmospheric Release (Ci/yr)[c]	First-Year Dose (rem) Total-Body	First-Year Dose (rem) Bone	First-Year Dose (rem) Lung	First-Year Dose (rem) GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Total-Body	Fifty-Year Committed Dose Equivalent (rem) Bone	Fifty-Year Committed Dose Equivalent (rem) Lung	Fifty-Year Committed Dose Equivalent (rem) GI-LLI
Preparations for Accident Cleanup										
• Vent Containment	^{85}Kr	3.1×10^{4}	1.2×10^{-4}	1.2×10^{-4}	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 Decon. of Containment										
• Remote Washdown, High Pressure Spray, Hands-On Decon., Install Local Shielding	Accident Scenario 2	1.0×10^{-4}	9.4×10^{-8}	8.9×10^{-8}	4.8×10^{-8}	2.8×10^{-8}	1.9×10^{-7}	4.7×10^{-7}	1.0×10^{-7}	2.8×10^{-8}
• ^{3}H Evaporation Loss	^{3}H	5.2×10^{3}	7.3×10^{-3}	--	7.3×10^{-3}	6.8×10^{-3}	7.8×10^{-3}	--	7.8×10^{-3}	7.3×10^{-3}
Defueling the Reactor										
• Remove Reactor Pressure Vessel Head, Internals, and Core	^{3}H	4.5×10^{3}	6.3×10^{-3}	--	6.3×10^{-3}	5.9×10^{-3}	6.8×10^{-3}	--	6.8×10^{-3}	6.3×10^{-3}
• Vent ^{85}Kr	^{85}Kr	1.2×10^{2}	4.4×10^{-7}	4.4×10^{-7}	4.4×10^{-7}	4.4×10^{-7}	4.4×10^{-7}	4.4×10^{-7}	4.4×10^{-7}	4.4×10^{-7}
Cleanup of Primary Coolant System										
• OPG Solution	--[d]	--								
• EDTA Solution	^{3}H	3.1×10^{3}	4.3×10^{-3}	--	4.3×10^{-3}	4.0×10^{-3}	4.7×10^{-3}	--	4.7×10^{-3}	4.3×10^{-3}
• Immobilization of Decon. Solutions	Accident Scenario 2	1.1×10^{-6}	1.0×10^{-9}	9.8×10^{-10}	5.3×10^{-10}	3.1×10^{-10}	2.1×10^{-9}	5.2×10^{-9}	1.1×10^{-9}	3.1×10^{-10}
Immobilization of Process Solid Wastes	Accident Scenario 2	2.2×10^{-4}	2.1×10^{-7}	2.0×10^{-7}	1.1×10^{-7}	6.2×10^{-8}	4.2×10^{-7}	1.0×10^{-6}	2.2×10^{-7}	6.2×10^{-8}
Packaging and Handling Solid Wastes	Accident Scenario 2	5.0×10^{-4}	4.7×10^{-7}	4.5×10^{-7}	2.4×10^{-7}	1.4×10^{-7}	9.5×10^{-7}	2.4×10^{-6}	5.0×10^{-7}	1.4×10^{-7}
Process Decon. Liquid Wastes	Accident Scenario 2	9.6×10^{-4}	9.0×10^{-7}	8.5×10^{-7}	4.6×10^{-7}	2.7×10^{-7}	1.8×10^{-6}	4.5×10^{-6}	9.6×10^{-7}	2.7×10^{-7}

(a) Doses are not totalled 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.
(b) The radionuclide inventories used in the dose calculations are discussed in Appendix C.
(c) For comparison purposes, dose calculations are based on a continuous release for 1 year. These releases are estimated by direct comparison to those reported in Reference 1.
(d) No release is calculated for the OPG solution cleanup.

TABLE J.2-3. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases Resulting from Routine Tasks During Post-Accident Cleanup After the PWR Scenario 3 Accident[a]

Cleanup Tasks	Reference Radionuclide Inventory[b]	Annual Atmospheric Release (Ci/yr)[c]	First-Year Dose (rem) Total-Body	First-Year Dose (rem) Bone	First-Year Dose (rem) Lung	First-Year Dose (rem) GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Total-Body	Fifty-Year Committed Dose Equivalent (rem) Bone	Fifty-Year Committed Dose Equivalent (rem) Lung	Fifty-Year Committed Dose Equivalent (rem) GI-LLI
Preparations for Accident Cleanup										
• Vent Containment	^{85}Kr	2.4×10^{5}	8.9×10^{-4}	8.9×10^{-4}	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 Decon. of Containment										
• Remote Washdown, High Pressure Spray, Hands-On Decon., Install Local Shielding	Accident Scenario 3	1.4×10^{-4}	1.3×10^{-7}	1.3×10^{-7}	7.6×10^{-8}	4.3×10^{-8}	3.1×10^{-7}	8.8×10^{-7}	1.8×10^{-7}	4.3×10^{-8}
• ^{3}H Evaporation Loss	^{3}H	1.0×10^{4}	1.4×10^{-2}	--	1.4×10^{-2}	1.3×10^{-2}	1.5×10^{-2}	--	1.5×10^{-2}	1.4×10^{-2}
Defueling the Reactor										
• Remove Reactor Pressure Vessel Head, Internals, and Core	^{3}H	9.0×10^{3}	1.3×10^{-2}	--	1.3×10^{-2}	1.2×10^{-2}	1.4×10^{-2}	--	1.4×10^{-2}	1.3×10^{-2}
• Vent ^{85}Kr	^{85}Kr	9.3×10^{2}	3.4×10^{-6}	3.4×10^{-6}	3.4×10^{-6}	3.4×10^{-6}	3.4×10^{-6}	3.4×10^{-6}	3.4×10^{-6}	3.4×10^{-6}
Cleanup of Primary Coolant System										
• OPG Solution	--[d]	--								
• EDTA Solution	^{3}H	6.2×10^{3}	8.7×10^{-3}	--	8.7×10^{-3}	8.1×10^{-3}	9.3×10^{-3}	--	9.3×10^{-3}	8.7×10^{-3}
• Immobilization of Decon. Solutions	Accident Scenario 3	1.6×10^{-6}	1.5×10^{-9}	1.5×10^{-9}	8.6×10^{-10}	5.0×10^{-10}	3.5×10^{-9}	1.0×10^{-8}	2.1×10^{-9}	5.0×10^{-10}
Immobilization of Process Solid Wastes	Accident Scenario 3	3.2×10^{-4}	3.0×10^{-7}	3.0×10^{-7}	1.7×10^{-7}	9.9×10^{-8}	7.0×10^{-7}	2.0×10^{-6}	4.2×10^{-7}	9.9×10^{-8}
Packaging and Handling Solid Wastes	Accident Scenario 3	7.2×10^{-4}	6.7×10^{-7}	6.8×10^{-7}	3.9×10^{-7}	2.2×10^{-7}	1.6×10^{-6}	4.5×10^{-6}	9.4×10^{-7}	2.2×10^{-7}
Process Decon. Liquid Wastes	Accident Scenario 3	1.4×10^{-3}	1.3×10^{-6}	1.3×10^{-6}	7.6×10^{-7}	4.3×10^{-7}	3.1×10^{-6}	8.8×10^{-6}	1.8×10^{-6}	4.3×10^{-7}

(a) Doses are not totalled 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.
(b) The radionuclide inventories used in the dose calculations are discussed in Appendix C.
(c) For comparison purposes, dose calculations are based on a continuous release for 1 year. These releases are estimated by adjusting the releases from accident scenario 2 for increased radionuclide release fractions.
(d) No release is calculated for the OPG solution cleanup.

TABLE J.2-4. Radiation Doses to the Population from Atmospheric Releases Resulting from Routine Tasks During Post-Accident Cleanup After the PWR Scenario 1 Accident[a]

Cleanup Tasks	Reference Radionuclide Inventory[b]	Annual Atmospheric Release (Ci/yr)[c]	First-Year Dose (man-rem)[d]				Fifty-Year Committed Dose Equivalent (man-rem)[d]			
			Total-Body	Bone	Lung	GI-LLI	Total-Body	Bone	Lung	GI-LLI
Preparations for Accident Cleanup										
• Vent Containment	^{85}Kr	1.6×10^{3}	7×10^{-3}	7×10^{-3}	7×10^{-3}	7×10^{-3}	7×10^{-3}	7×10^{-3}	7×10^{-3}	7×10^{-3}
Initial Decon. of Containment										
• Remote Washdown, High Pressure Spray, Hands-On Decon., Install Local Shielding	Accident Scenario 1	8.0×10^{-6}	6×10^{-6}	4×10^{-6}	2×10^{-6}	1×10^{-6}	9×10^{-6}	7×10^{-6}	2×10^{-6}	1×10^{-6}
• ^{3}H Evaporation Loss	^{3}H	1.0×10^{3}	9×10^{-1}	--	9×10^{-1}	7×10^{-1}	1×10^{0}	--	9×10^{-1}	8×10^{-1}
Defueling the Reactor										
• Remove Reactor Pressure Vessel Head, Internals, and Core	^{3}H	9.0×10^{2}	8×10^{-1}	--	8×10^{-1}	7×10^{-1}	9×10^{-1}	--	8×10^{-1}	7×10^{-1}
• Vent ^{85}Kr	^{85}Kr	6.2×10^{0}	3×10^{-5}	3×10^{-5}	3×10^{-5}	3×10^{-5}	3×10^{-5}	3×10^{-5}	3×10^{-5}	3×10^{-5}
Cleanup of Primary Coolant System										
• OPG Solution	--[e]	--								
• EDTA Solution	^{3}H	6.2×10^{2}	6×10^{-1}	--	6×10^{-1}	5×10^{-1}	6×10^{-1}	--	6×10^{-1}	5×10^{-1}
• Immobilization of Decon. Solutions	Accident Scenario 1	8.8×10^{-8}	6×10^{-8}	5×10^{-8}	2×10^{-8}	1×10^{-8}	1×10^{-7}	8×10^{-8}	3×10^{-8}	1×10^{-8}
Immobilization of Process Solid Wastes	Accident Scenario 1	1.8×10^{-5}	1×10^{-5}	1×10^{-5}	4×10^{-6}	2×10^{-6}	2×10^{-5}	2×10^{-5}	5×10^{-6}	2×10^{-6}
Packaging and Handling Solid Wastes	Accident Scenario 1	4.0×10^{-5}	2×10^{-5}	2×10^{-5}	1×10^{-5}	6×10^{-6}	4×10^{-5}	4×10^{-5}	1×10^{-5}	6×10^{-6}
Process Decon. Liquid Wastes	Accident Scenario 1	7.7×10^{-5}	6×10^{-5}	4×10^{-5}	2×10^{-5}	1×10^{-5}	8×10^{-5}	7×10^{-5}	2×10^{-5}	1×10^{-5}

(a) Doses are not totalled because they occur during different years.
(b) The radionuclide inventories used in the dose calculations are discussed in Appendix C.
(c) For comparison purposes, dose calculations are based on a continuous release for 1 year. These releases are estimated by adjusting the releases from accident scenario 2 for reduced radionuclide release fractions.
(d) Population doses are rounded to one significant figure.
(e) No release is calculated for the OPG solution cleanup.

TABLE J.2-5. Radiation Doses to the Population from Atmospheric Releases Resulting from Routine Tasks During Post-Accident Cleanup After the PWR Scenario 2 Accident[a]

Cleanup Tasks	Reference Radionuclide Inventory[b]	Annual Atmospheric Release (Ci/yr)[c]	First-Year Dose (man-rem)[d]				Fifty-Year Committed Dose Equivalent (man-rem)[d]			
			Total-Body	Bone	Lung	GI-LLI	Total-Body	Bone	Lung	GI-LLI
Preparations for Accident Cleanup										
• Vent Containment	^{85}Kr	3.1×10^{4}	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-1}	1×10^{-1}
Initial Decon. of Containment										
• Remote Washdown, High Pressure Spray, Hands-On Decon., Install Local Shielding	Accident Scenario 2	1.0×10^{-4}	7×10^{-5}	7×10^{-5}	4×10^{-5}	2×10^{-5}	1×10^{-4}	4×10^{-4}	1×10^{-4}	2×10^{-5}
• ^{3}H Evaporation Loss	^{3}H	5.2×10^{3}	5×10^{0}	--	5×10^{0}	4×10^{0}	5×10^{0}	--	5×10^{0}	4×10^{0}
Defueling the Reactor										
• Remove Reactor Pressure Vessel Head, Internals, and Core	^{3}H	4.5×10^{3}	4×10^{0}	--	4×10^{0}	3×10^{0}	4×10^{0}	--	4×10^{0}	3×10^{0}
• Vent ^{85}Kr	^{85}Kr	1.2×10^{2}	5×10^{-4}	5×10^{-4}	5×10^{-4}	5×10^{-4}	5×10^{-4}	5×10^{-4}	5×10^{-4}	5×10^{-4}
Cleanup of Primary Coolant System										
• OPG Solution	--[e]	--								
• EDTA Solution	^{3}H	3.1×10^{3}	3×10^{0}	--	3×10^{0}	2×10^{0}	3×10^{0}	--	3×10^{0}	2×10^{0}
• Immobilization of Decon. Solutions	Accident Scenario 2	1.1×10^{-6}	8×10^{-7}	7×10^{-6}	4×10^{-7}	2×10^{-7}	2×10^{-6}	4×10^{-5}	1×10^{-6}	2×10^{-7}
Immobilization of Process Solid Wastes	Accident Scenario 2	2.2×10^{-4}	1×10^{-4}	1×10^{-4}	9×10^{-5}	4×10^{-5}	3×10^{-4}	9×10^{-4}	2×10^{-4}	4×10^{-5}
Packaging and Handling Solid Wastes	Accident Scenario 2	5.0×10^{-4}	4×10^{-4}	3×10^{-4}	2×10^{-4}	9×10^{-5}	7×10^{-4}	2×10^{-3}	5×10^{-4}	9×10^{-5}
Process Decon. Liquid Wastes	Accident Scenario 2	9.6×10^{-4}	7×10^{-4}	7×10^{-4}	4×10^{-4}	2×10^{-4}	1×10^{-3}	4×10^{-3}	9×10^{-4}	2×10^{-4}

(a) Doses are not totalled because they occur during different years.
(b) The radionuclide inventories used in the dose calculations are discussed in Appendix C.
(c) For comparison purposes, dose calculations are based on a continuous release for 1 year. These releases are estimated by direct comparison to those reported in Reference 1.
(d) Population doses are rounded to one significant figure.
(e) No release is calculated for the OPG solution cleanup.

TABLE J.2-6. Radiation Doses to the Population from Atmospheric Releases Resulting from Routine Tasks During Post-Accident Cleanup After the PWR Scenario 3 Accident[a]

Cleanup Tasks	Reference Radionuclide Inventory[b]	Annual Atmospheric Release (Ci/yr)[c]	First-Year Dose (man-rem)[d]				Fifty-Year Committed Dose Equivalent (man-rem)[d]			
			Total-Body	Bone	Lung	GI-LLI	Total-Body	Bone	Lung	GI-LLI
Preparations for Accident Cleanup										
• Vent Containment	^{85}Kr	2.4×10^5	1×10^0	1×10^0	1×10^0	1×10^0	1×10^0	1×10^0	1×10^0	1×10^0
Initial Decon. of Containment										
• Remote Washdown, High Pressure Spray, Hands-On Decon., Install Local Shielding	Accident Scenario 3	1.4×10^{-4}	1×10^{-4}	1×10^{-4}	7×10^{-5}	3×10^{-5}	2×10^{-4}	7×10^{-4}	2×10^{-4}	3×10^{-5}
• ^3H Evaporation Loss	^3H	1.0×10^4	9×10^0	--	9×10^0	8×10^0	9×10^0	--	9×10^0	8×10^0
Defueling the Reactor										
• Remove Reactor Pressure Vessel Head, Internals, and Core	^3H	9.0×10^3	8×10^0	--	8×10^0	7×10^0	9×10^0	--	8×10^0	7×10^0
• Vent ^{85}Kr	^{85}Kr	9.3×10^2	4×10^{-3}	4×10^{-3}	4×10^{-3}	4×10^{-3}	4×10^{-3}	4×10^{-3}	4×10^{-3}	4×10^{-3}
Cleanup of Primary Coolant System										
• OPG Solution	--[e]	--								
• EDTA Solution	^3H	6.2×10^3	6×10^0	--	6×10^0	5×10^0	6×10^0	--	6×10^0	5×10^0
• Immobilization of Decon. Solutions	Accident Scenario 3	1.6×10^{-6}	1×10^{-6}	1×10^{-6}	8×10^{-7}	3×10^{-7}	3×10^{-6}	8×10^{-6}	3×10^{-6}	3×10^{-7}
Immobilization of Process Solid Wastes	Accident Scenario 3	3.2×10^{-4}	2×10^{-4}	2×10^{-4}	2×10^{-4}	6×10^{-5}	6×10^{-4}	2×10^{-3}	6×10^{-4}	6×10^{-5}
Packaging and Handling Solid Wastes	Accident Scenario 3	7.2×10^{-4}	5×10^{-4}	5×10^{-4}	3×10^{-4}	1×10^{-4}	1×10^{-3}	3×10^{-3}	1×10^{-3}	1×10^{-4}
Process Decon. Liquid Wastes	Accident Scenario 3	1.4×10^{-3}	1×10^{-3}	1×10^{-3}	7×10^{-4}	3×10^{-4}	3×10^{-3}	7×10^{-3}	3×10^{-3}	3×10^{-4}

(a) Doses are not totalled because they occur during different years.
(b) The radionuclide inventories used in the dose calculations are discussed in Appendix C.
(c) For comparison purposes, dose calculations are based on a continuous release for 1 year. These releases are estimated by adjusting the releases from accident scenario 2 for increased radionuclide release fractions.
(d) Population doses are rounded to one significant figure.
(e) No release is calculated for the OPG solution cleanup.

J.2.1.2 Public Radiation Doses from Releases Due to Postulated Industrial Accidents During Accident Cleanup

During accident cleanup, unexpected situations may arise that lead to the accidental release of radioactivity from the plant to the atmosphere or to the nearby river. The scenarios considered in this study for these accidental situations during accident cleanup at the reference PWR are the same as those analyzed for accident cleanup at TMI-2, as presented in Reference 1. As is done for the routine releases discussed in the preceding subsection, the release values for TMI-2 are adjusted to account for differences in fuel burnup and in release fractions of radionuclides escaping from the core at the time of the postulated accident. The radiation doses resulting from these releases are calculated using the same radiation dose models used to analyze doses from routine releases.

Estimates of releases of radioactivity due to industrial accidents during accident cleanup and of the resulting first-year doses and fifty-year committed dose equivalents to the maximum-exposed individual are presented in Tables J.2-7, J.2-8, and J.2-9 for accident cleanup following a scenario 1, scenario 2, or scenario 3 accident, respectively. Each release is assumed to occur during a one-hour period so that a comparison of the releases and associated doses can be made. As shown in the tables, the release resulting in the greatest doses to the maximum-exposed individual is attributable to the potential mishandling of waste from the demineralizer system installed in the spent fuel pool to clean up accident water. Postulated releases due to industrial accidents and resulting doses during accident cleanup following a scenario 1 reactor accident are approximately one order of magnitude lower than those following a scenario 2 accident which, in turn, are about one order of magnitude lower than those following a scenario 3 accident.

While it is beyond the scope of this study to evaluate every potential industrial accident situation that could result in a release of radioactivity during accident cleanup, the releases presented here are judged to represent the range of credible events and to reflect realistic maximum impacts to the public from industrial accident situations. Multiple-failure-event accidents are not considered (i.e., each release considered is the result of a single failure and does not require a chain of failure events to occur).

TABLE J.2-7. Radiation Doses to the Maximum-Exposed Individual from Releases Due to Industrial Accidents During Post-Accident Cleanup - Scenario 1

Accident	Total Release (Ci/h)[a]	Reference Radionuclide Inventory[b]	First-Year Dose (rem) Total-Body	Bone	Lung	GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Total-Body	Bone	Lung	GI-LLI
Liquid Release to River[c]	9.0×10^{2}	^{3}H	8.9×10^{-5}	--	8.9×10^{-5}	8.9×10^{-5}	8.9×10^{-5}	--	8.9×10^{-5}	8.9×10^{-5}
	3.1×10^{0}	Accident Scenario 1	4.3×10^{-4}	3.2×10^{-4}	6.2×10^{-5}	7.8×10^{-6}	4.7×10^{-4}	3.8×10^{-4}	6.8×10^{-5}	7.8×10^{-6}
Release of Trapped Fission Products	6.2×10^{0}	^{85}Kr	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}	2.0×10^{-6}
Accident-Water Cleanup Demin. System Waste Handling	9.6×10^{-1}	Accident Scenario 1	1.1×10^{-2}	7.7×10^{-3}	2.3×10^{-3}	7.8×10^{-5}	1.2×10^{-2}	9.1×10^{-3}	2.5×10^{-3}	7.8×10^{-5}
Transportation Accident	9.6×10^{-2}	Accident Scenario 1	1.1×10^{-3}	7.7×10^{-4}	2.3×10^{-4}	7.8×10^{-6}	1.2×10^{-3}	9.1×10^{-4}	2.5×10^{-4}	7.8×10^{-6}
HEPA Failure During Liquid Waste Treatment	5.3×10^{-3}	Accident Scenario 1	5.8×10^{-5}	4.2×10^{-5}	1.3×10^{-5}	4.3×10^{-7}	6.4×10^{-5}	5.0×10^{-5}	1.4×10^{-5}	4.3×10^{-7}
Storage Area Fire	3.2×10^{-3}	Accident Scenario 1	3.5×10^{-5}	2.6×10^{-5}	7.7×10^{-6}	2.6×10^{-7}	3.8×10^{-5}	3.0×10^{-5}	8.3×10^{-6}	2.6×10^{-7}
Other Waste Handling	2.7×10^{-3}	Accident Scenario 1	3.0×10^{-5}	2.2×10^{-5}	6.5×10^{-6}	2.2×10^{-7}	3.2×10^{-5}	2.6×10^{-5}	7.0×10^{-6}	2.2×10^{-7}
Spill of Decon. Liquids from RCS	1.6×10^{-4}	Accident Scenario 1	1.8×10^{-6}	1.3×10^{-6}	3.1×10^{-7}	1.3×10^{-8}	1.9×10^{-6}	1.5×10^{-6}	4.2×10^{-7}	1.3×10^{-8}
HEPA Filter Failure	7.6×10^{-5}	Accident Scenario 1	8.4×10^{-7}	6.1×10^{-7}	1.8×10^{-7}	6.1×10^{-9}	9.1×10^{-7}	7.2×10^{-7}	2.0×10^{-7}	6.1×10^{-9}
Solid Waste Handling	2.2×10^{-5}	Accident Scenario 1	2.4×10^{-7}	1.8×10^{-7}	5.3×10^{-8}	1.8×10^{-9}	2.6×10^{-7}	2.1×10^{-7}	5.7×10^{-8}	1.8×10^{-9}
Chem. Decon. Waste Handling	8.0×10^{-8}	Accident Scenario 1	8.8×10^{-10}	6.4×10^{-10}	1.9×10^{-10}	6.5×10^{-12}	9.6×10^{-10}	7.6×10^{-10}	2.1×10^{-10}	6.5×10^{-12}

(a) For comparison, all releases are assumed to occur in a one-hour period.
(b) The radionuclide inventories used in the dose calculations are discussed in Appendix C.
(c) All releases are to the atmosphere except for the liquid release to the river.

TABLE J.2-8. Radiation Doses to the Maximum-Exposed Individual from Releases Due to Industrial Accidents During Post-Accident Cleanup - Scenario 2

Accident	Total Release (Ci/h)(a)	Reference Radionuclide Inventory(b)	First-Year Dose (rem) Total-Body	Bone	Lung	GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Total-Body	Bone	Lung	GI-LLI
Liquid Release to River(c)	3.1×10^{3}	^{3}H	3.1×10^{-4}	--	3.1×10^{-4}	3.1×10^{-4}	3.1×10^{-4}	--	3.1×10^{-4}	3.1×10^{-4}
	8.6×10^{0}	Accident Scenario 2	1.1×10^{-3}	9.9×10^{-4}	1.5×10^{-4}	1.2×10^{-4}	1.7×10^{-3}	2.9×10^{-3}	2.9×10^{-4}	1.2×10^{-4}
Release of Trapped Fission Products	1.2×10^{2}	^{85}Kr	3.8×10^{-5}	3.8×10^{-5}	3.8×10^{-5}	3.8×10^{-5}	3.8×10^{-5}	3.8×10^{-5}	3.8×10^{-5}	3.8×10^{-5}
Accident-Water Cleanup Demin. System Waste Handling	1.2×10^{1}	Accident Scenario 2	1.3×10^{-1}	3.0×10^{-1}	4.1×10^{-1}	9.4×10^{-3}	2.9×10^{-1}	2.5×10^{0}	8.2×10^{-1}	9.4×10^{-3}
Transportation Accident	1.2×10^{0}	Accident Scenario 2	1.3×10^{-2}	3.0×10^{-2}	4.1×10^{-2}	9.4×10^{-4}	2.9×10^{-2}	2.5×10^{-1}	8.2×10^{-2}	9.4×10^{-4}
HEPA Failure During Liquid Waste Treatment	6.6×10^{-2}	Accident Scenario 2	7.3×10^{-4}	1.7×10^{-3}	2.2×10^{-3}	5.1×10^{-5}	1.6×10^{-3}	1.4×10^{-2}	4.5×10^{-3}	5.1×10^{-5}
Storage Area Fire	4.0×10^{-2}	Accident Scenario 2	4.4×10^{-4}	1.0×10^{-3}	1.3×10^{-3}	3.1×10^{-5}	9.6×10^{-4}	8.4×10^{-3}	2.7×10^{-3}	3.1×10^{-5}
Other Waste Handling	3.4×10^{-2}	Accident Scenario 2	3.7×10^{-4}	8.5×10^{-4}	1.2×10^{-3}	2.7×10^{-5}	8.2×10^{-4}	7.1×10^{-3}	2.3×10^{-3}	2.7×10^{-5}
Spill of Decon. Liquids from RCS	2.0×10^{-3}	Accident Scenario 2	2.2×10^{-5}	5.0×10^{-5}	6.8×10^{-5}	1.6×10^{-6}	4.8×10^{-5}	4.2×10^{-4}	1.4×10^{-4}	1.6×10^{-6}
HEPA Filter Failure	9.4×10^{-4}	Accident Scenario 2	1.0×10^{-5}	2.4×10^{-5}	3.2×10^{-5}	7.3×10^{-7}	2.3×10^{-5}	2.0×10^{-4}	6.4×10^{-5}	7.3×10^{-7}
Solid Waste Handling	2.7×10^{-4}	Accident Scenario 2	3.0×10^{-6}	6.8×10^{-6}	9.2×10^{-6}	2.1×10^{-7}	6.5×10^{-6}	5.7×10^{-5}	1.8×10^{-5}	2.1×10^{-7}
Chem. Decon. Waste Handling	1.0×10^{-6}	Accident Scenario 2	1.1×10^{-8}	2.5×10^{-8}	3.4×10^{-8}	7.8×10^{-10}	2.4×10^{-8}	2.1×10^{-7}	6.8×10^{-8}	7.8×10^{-10}

(a) For comparison, all releases are assumed to occur in a one-hour period.
(b) The radionuclide inventories used in the dose calculations are discussed in Appendix C.
(c) All releases are to the atmosphere except for the liquid release to the river.

TABLE J.2-9. Radiation Doses to the Maximum-Exposed Individual from Releases Due to Industrial Accidents During Post-Accident Cleanup - Scenario 3

Accident	Total Release (Ci/h)[a]	Reference Radionuclide Inventory[b]	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
			Total-Body	Bone	Lung	GI-LLI	Total-Body	Bone	Lung	GI-LLI
Liquid Release to River[c]	9.0×10^{3}	^{3}H	8.9×10^{-4}	--	8.9×10^{-4}	8.9×10^{-4}	8.9×10^{-4}	--	8.9×10^{-4}	8.9×10^{-4}
	3.8×10^{1}	Accident Scenario 3	5.0×10^{-3}	4.7×10^{-3}	6.9×10^{-4}	6.9×10^{-4}	8.8×10^{-3}	1.7×10^{-2}	7.6×10^{-4}	6.9×10^{-4}
Release of Trapped Fission Products	9.3×10^{2}	^{85}Kr	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}
Accident-Water Cleanup Demin. System Waste Handling	8.2×10^{1}	Accident Scenario 3	9.0×10^{-1}	2.7×10^{0}	3.9×10^{0}	9.0×10^{-2}	2.5×10^{0}	3.9×10^{0}	8.0×10^{0}	9.0×10^{-2}
Transportation Accident	8.2×10^{0}	Accident Scenario 3	9.0×10^{-2}	2.7×10^{-1}	3.9×10^{-1}	9.0×10^{-3}	2.5×10^{-1}	3.9×10^{-1}	8.0×10^{-1}	9.0×10^{-3}
HEPA Failure During Liquid Waste Treatment	4.5×10^{-1}	Accident Scenario 3	5.0×10^{-3}	1.5×10^{-2}	2.2×10^{-2}	5.0×10^{-4}	1.4×10^{-2}	2.2×10^{-2}	4.4×10^{-2}	5.0×10^{-4}
Storage Area Fire	2.8×10^{-1}	Accident Scenario 3	3.1×10^{-3}	9.3×10^{-3}	1.3×10^{-2}	3.1×10^{-4}	8.4×10^{-3}	1.3×10^{-2}	2.7×10^{-2}	3.1×10^{-4}
Other Waste Handling	2.3×10^{-1}	Accident Scenario 3	2.5×10^{-3}	7.5×10^{-3}	1.1×10^{-2}	2.5×10^{-3}	6.9×10^{-3}	1.1×10^{-2}	2.2×10^{-2}	2.5×10^{-3}
Spill of Decon. Liquids from RCS	1.4×10^{-2}	Accident Scenario 3	1.5×10^{-4}	4.5×10^{-4}	6.7×10^{-4}	1.5×10^{-5}	4.2×10^{-4}	6.7×10^{-4}	1.4×10^{-3}	1.5×10^{-5}
HEPA Filter Failure	6.4×10^{-3}	Accident Scenario 3	7.0×10^{-5}	2.1×10^{-4}	3.1×10^{-4}	7.0×10^{-6}	2.6×10^{-4}	3.1×10^{-4}	6.2×10^{-4}	7.0×10^{-6}
Solid Waste Handling	1.8×10^{-3}	Accident Scenario 3	2.0×10^{-5}	6.0×10^{-5}	8.6×10^{-5}	2.0×10^{-6}	5.4×10^{-5}	8.6×10^{-5}	1.7×10^{-4}	2.0×10^{-6}
Chem. Decon. Waste Handling	7.0×10^{-6}	Accident Scenario 3	7.7×10^{-8}	2.3×10^{-7}	3.4×10^{-7}	7.7×10^{-8}	2.1×10^{-7}	3.4×10^{-7}	6.8×10^{-7}	7.7×10^{-8}

(a) For comparison, all releases are assumed to occur in a one-hour period.
(c) The radionuclide inventories used in the dose calculations are discussed in Appendix C.
(b) All releases are to the atmosphere except for the liquid release to the river.

J.2.2 Occupational Safety Aspects of Accident Cleanup

The occupational safety impacts of accident cleanup activities are discussed in the following subsections, including radiation doses to workers performing the accident cleanup tasks and potential industrial-accident (nonradiological) impacts to these workers. The information developed here is based on the detailed description of accident cleanup activities presented in Appendix E of this study.

J.2.2.1 Occupational Radiation Doses from Accident Cleanup Activities

The estimated occupational radiation doses accumulated by cleanup workers are based on postulated external gamma radiation dose rates in various areas of the reference PWR during accident cleanup and on estimated staff labor requirements for completing the accident cleanup tasks. Workers are assumed to use respiration equipment as appropriate to protect against inhalation of radioactive materials. The detailed analysis of occupational radiation doses from accident cleanup is presented in Appendix E because the results of the analysis are needed to adjust manpower requirements to ensure compliance with individual radiation dose limitations of 5 man-rem/man-year.[10] The results of these analyses are summarized here. These results do not include the doses to the transport workers engaged in shipping the accident cleanup wastes to offsite repositories; transport worker doses are included in Section J.2.3 with the other safety impacts of transportation activities.

Summaries of the estimated occupational radiation doses during accident cleanup following a scenario 1, scenario 2, or scenario 3 accident are given in Tables J.2-10, J.2-11, and J.2-12, respectively. No credit is taken for the decay of the radioactive materials that are the source of the radiation doses during accident cleanup because the dominant radionuclide is ^{137}Cs (with about a 30-year half-life) and, thus, the anticipated effect of this decay is minimal. As shown in the tables, the total occupational doses from accident cleanup following a scenario 1 accident are about a factor of 6 lower than those following a scenario 2 accident which, in turn, are about a factor of 3 lower than those following a scenario 3 accident. The doses for cleanup following a scenario 2 and scenario 3 accident include a contribution from cleanup activities postulated to be required in the auxiliary and fuel building.

TABLE J.2-10. Summary of Estimated Occupational Radiation Doses from Accident Cleanup of the Reference PWR Following a Scenario 1 Accident[a]

Activity Area	Estimated Occupational Doses (man-rem) Crew Leaders	Utility Operators	Laborers	Craftsmen	Health Physics Technicians	Totals
Processing of contaminated Liquids	--	28.80	14.40	--	--	43.20
Initial Decontamination of Containment Building	23.96	80.76	53.32	23.60	23.96	205.60
Defueling of the Reactor	16.00	87.20	32.80	32.00	28.80	196.80
Cleanup of the Primary Coolant System	4.40	12.00	6.00	5.20	4.40	32.00
Support Operations	28.20	36.00	37.20	57.60	30.00	189.00
Subtotals	72.56	244.76	143.72	118.40	87.16	666.60
Planning and Preparations						3.6
Total						670.2

(a) Summarized from Tables E.2-1 and E.4-4 of Appendix E.

TABLE J.2-11. Summary of Estimated Occupational Radiation Doses from Accident Cleanup of the Reference PWR Following a Scenario 2 Accident[a]

Activity Area	Estimated Occupational Doses (man-rem) Crew Leaders	Utility Operators	Laborers	Craftsmen	Health Physics Technicians	Totals
Cleanup of Auxiliary and Fuel Buildings	210.40	314.50	316.90	559.00	211.60	1 612.40
Processing of Contaminated Liquids	--	96.00	48.00	--	--	144.00
Initial Decontamination of Containment Building	87.00	290.40	193.20	90.80	88.80	750.20
Defueling of the Reactor	82.40	448.80	164.80	164.80	149.60	1 010.40
Cleanup of the Primary Coolant System	18.00	50.40	25.20	28.80	18.00	140.40
Support Operations	123.12	105.60	109.44	410.40	128.88	877.44
Subtotals	520.92	1 305.70	857.54	1 253.80	596.88	4 534.84
Planning and Preparations						45
Total						4 580

(a) Summarized from Tables E.2-1, E.3-2, and E.4-5 of Appendix E.

TABLE J.2-12. Summary of Estimated Occupational Radiation Doses from Accident Cleanup of the Reference PWR Following a Scenario 3 Accident[a]

Activity Area	Estimated Occupational Doses (man-rem) Crew Leaders	Utility Operators	Laborers	Craftsmen	Health Physics Technicians	Totals
Cleanup of Auxiliary and Fuel Buildings	210.40	314.50	316.90	559.00	211.60	1 612.40
Processing of Contaminated Liquids from Containment Building	--	192.00	96.00	--	--	288.00
Initial Decontamination of Containment Building	215.40	733.20	484.80	267.60	224.40	1 925.40
Defueling of the Reactor	297.60	1 602.80	595.20	889.60	528.00	3 913.20
Cleanup of the Primary Coolant System	55.20	187.20	93.60	148.80	55.20	540.00
Support Operations	406.40	288.00	304.00	2 035.20	430.40	3 464.00
Subtotals	1 185.00	3 317.70	1 890.50	3 900.20	1 449.60	11 743.00
Planning and Preparations						360
Total						12 103

(a) Summarized from Tables E.2-1, E.3-2, and E.4-6 of Appendix E.

J.2.2.2 Industrial Safety Aspects of Accident Cleanup

As for any industrial activity, injuries and fatalities can result to workers engaged in accident cleanup because of industrial accidents, but proper management and safety practices can minimize the occurrence of such accidents. Frequency estimates for injuries and fatalities during accident cleanup are based on data collected by the U.S. AEC for the period 1943-1970.[11]

The applicable staff man-hours used to estimate the potential injuries and fatalities are assumed to be the exposure hours given in Appendix E for the various accident cleanup tasks and are divided into three categories of accident potential.[12] The category with the highest potential impact, heavy construction, is not applicable to accident cleanup. The next category, light construction, primarily involves reactor defueling, installation of equipment, and other miscellaneous construction and maintenance tasks. The remainder of the accident cleanup activities are categorized as equivalent to operational support.

Table J.2-13 contains the estimated worker injuries and fatalities resulting from accident cleanup following each of the three reactor accident scenarios considered in this study. As shown in the table, about 2 lost-time injuries could result during cleanup following a scenario 3 accident, about 1 injury following a scenario 2 accident, and less than 1 injury following a scenario 1 accident. Fatalities appear to be unlikely during accident cleanup.

J.2.3 Transportation Safety Aspects of Accident Cleanup

Radioactive waste materials that result from accident cleanup are assumed to be shipped offsite to appropriate repositories as part the planned accident cleanup activities. The potential safety impacts from the transportation of this material are as follows:

- radiation doses from the radioactive materials to transport workers and to members of the public along the transportation routes
- radiation doses to the maximum-exposed individual from accidental atmospheric releases during transportation accidents
- injuries and fatalities resulting from transportation accidents.

The safety impacts of radioactive material transportation during accident cleanup at the reference PWR are discussed in the following subsections.

J.2.3.1 Radiation Doses from Routine Transportation Activities During Accident Cleanup

Radioactive waste materials resulting from accident cleanup are assumed to be shipped by truck to either a shallow-land burial site or a federal repository, either of which is assumed to be located 1600 km from the reference PWR. The method used to estimate radiation doses to transportation workers and to members of the public along the transportation route is based

TABLE J.2-13. Estimated Occupational Lost-Time Injuries and Fatalities Resulting from Accident Cleanup Activities

Accident-Potential Category	Frequency (Accidents/man-hr) Lost-Time Injuries	Frequency (Accidents/man-hr) Fatalities	Accident Scenario 1 man-hrs[a]	Accident Scenario 1 Lost-Time Injuries	Accident Scenario 1 Fatalities	Accident Scenario 2 man-hrs[b]	Accident Scenario 2 Lost-Time Injuries	Accident Scenario 2 Fatalities	Accident Scenario 3 man-hrs[c]	Accident Scenario 3 Lost-Time Injuries	Accident Scenario 3 Fatalities
Heavy Construction	10×10^{-6}	4.2×10^{-8}	NA[d]	--	--	NA	--	--	NA	--	--
Light Construction	5.4×10^{-6}	3.0×10^{-8}	3.6×10^{4}	0.19	1.1×10^{-3}	9.6×10^{4}	0.52	2.9×10^{-3}	2.9×10^{5}	1.5	8.7×10^{-3}
Operational Support	2.1×10^{-6}	2.3×10^{-8}	5.4×10^{4}	0.11	1.2×10^{-3}	9.5×10^{4}	0.20	2.2×10^{-3}	2.7×10^{5}	0.57	6.2×10^{-3}
Totals			9.0×10^{4}	0.30	2.3×10^{-3}	1.9×10^{5}	0.72	5.1×10^{-3}	5.6×10^{5}	2.1	1.5×10^{-2}

(a) Summarized from Table E.4-4 of Appendix E.
(b) Summarized from Table E.4-5 of Appendix E.
(c) Summarized from Tables E.3-1 and E.4-6 of Appendix E.
(d) Heavy construction is not applicable to accident cleanup activities.

on information in Reference 13. Radiation doses received by workers unloading the radioactive materials at the repository or disposal site are not considered in this study since these doses are assumed to occur at separate licensed facilities.

The following assumptions are made about truck shipments of radioactive materials:

1. Each shipment contains enough radioactive material to result in the maximum radiation exposure rates allowable by regulations. Department of Transportation (DOT) regulations[(14)] set the following exposure limits:
   - 1000 mR/hr at 1 m from the external surface of any package transported in a closed vehicle
   - 200 mR/hr at the external surface of the vehicle
   - 10 mR/hr at any point 2 m from the vehicle
   - 2 mR/hr at any normally occupied position in the vehicle.
2. For each shipment of radioactive waste, two truck drivers spend 24 hours inside the cab (with an exposure rate of 2 mR/hr) and 2 hours outside of the cab at a distance of 2 m from the cargo (with an exposure rate of 10 mR/hr).
3. For each shipment of radioactive waste, two garagemen each spend 20 minutes at an average distance of 2 m from the truck payload (at an exposure rate of 10 mR/hr).
4. For each shipment, 20 onlookers from the general public each spend 3 minutes at an average distance of 2 m from the payload (at an exposure rate of 10 mR/hr).
5. The population density along the transport corridors is 120 persons/km^2.
6. All shipments maintain an average speed of 65 km/hr; thus, the cumulative dose to the public is 2.3×10^{-6} man-rem/km.

In addition to the other radioactive materials considered, spent fuel removed from the reactor during accident cleanup requires shipment to an ISFSI or to a federal repository, both of which are assumed to be located 1600 km from the reference PWR. Spent fuel shipments are assumed to be made by rail. Two train brakemen are assumed to spend 10 minutes during each of 10 stops (one every 160 km) at an average distance of 1 m from the shipping cask (at an assumed exposure rate of 25 mR/hr).

The number of radioactive materials shipments by truck during accident cleanup are:

- 90 following a scenario 1 accident (from Table F.3-3 of Appendix F)
- 289 following a scenario 2 accident (from Table F.2-3 and F.3-4 of Appendix F)
- 652 following a scenario 3 accident (from Table F.2-3 and F.3-5 of Appendix F)

Using these numbers of shipments and the foregoing assumptions, radiation doses to transport workers and to the general public are calculated as shown in Table J.2-14 for the truck transport of radioactive wastes during accident cleanup after each of the three accident scenarios.

Rail shipments of spent fuel and fuel debris from the final reactor core are estimated to number 30 following a scenario 1 accident, 50 following a scenario 2 accident, and 52 following a scenario 3 accident, as shown in Tables F.3-3 through F.3-5 of Appendix F. The occupational and public doses from spent fuel shipment during accident cleanup are presented in Table J.2-15.

J.2.3.2 Radiation Doses from Postulated Transportation Accidents During Accident Cleanup

Transportation accidents during the offsite shipment of radioactive materials from accident cleanup at the reference PWR can potentially result in inadvertent releases of radioactivity and corresponding radiation doses to individuals near the accident location.

TABLE J.2-14. Estimated Radiation Doses from Truck Shipments of Radioactive Materials During Accident Cleanup

Group	Radiation Dose per Shipment (man-rem)[a]	Number of Shipments	Total Radiation Dose (man-rem)[b]
Accident Scenario 1			
Truck Drivers	1.4×10^{-1}	90	13
Garagemen	6.7×10^{-3}	90	0.60
Total Transport Worker Dose			14
Onlookers	1.0×10^{-2}	90	0.90
General Public	3.7×10^{-3}	90	0.33
Total Public Dose			1.2
Accident Scenario 2			
Truck Drivers	1.4×10^{-1}	289	40
Garagemen	6.7×10^{-3}	289	1.9
Total Transport Worker Dose			42
Onlookers	1.0×10^{-2}	289	2.9
General Public	3.7×10^{-3}	289	1.1
Total Public Dose			4.0
Accident Scenario 3			
Truck Drivers	1.4×10^{-1}	652	91
Garagemen	6.7×10^{-3}	652	4.4
Total Transport Worker Dose			95
Onlookers	1.0×10^{-2}	652	6.5
General Public	3.7×10^{-3}	652	2.4
Total Public Dose			8.9

(a) Based on one-way trips of 1600 km.
(b) All doses are rounded to two significant figures.

TABLE J.2-15. Estimated Radiation Dose from Rail Transport of Spent Fuel and Fuel Debris During Accident Cleanup

Group	Radiation Dose per Shipment (man-rem)[a]	Number of Shipments	Total Radiation Dose (man-rem)[b]
Accident Scenario 1			
Train Brakemen	8.3×10^{-2}	30	2.5
Total Occupational Dose			2.5
Onlookers	1.0×10^{-2}	30	0.30
General Public	3.7×10^{-3}	30	0.11
Total Public Dose			0.41
Accident Scenario 2			
Train Brakemen	8.3×10^{-2}	50	4.2
Total Occupational Dose			4.2
Onlookers	1.0×10^{-2}	50	0.50
General Public	3.7×10^{-3}	50	0.19
Total Public Dose			0.69
Accident Scenario 3			
Train Brakemen	8.3×10^{-2}	52	4.3
Total Occupational Dose			4.3
Onlookers	1.0×10^{-2}	52	0.52
General Public	3.7×10^{-3}	52	0.19
Total Public Dose			0.71

(a) Based on one-way trips of 1600 km.
(b) All doses are rounded to two significant figures.

A realistic "worst-case" accident involving truck transport can be analyzed based on information in Section 10.4 of Reference 1, with appropriate adjustments to account for the differences between the post-accident radionuclide mixtures 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. A release fraction of 10^{-5} is assumed for these conditions. The releases and resulting doses from this accident following a scenario 1, scenario 2, or scenario 3 reactor accident are shown in Tables J.2-7 through J.2-9, presented previously in this appendix. The first-year total-body dose to the maximum-exposed individual from such an accident, even following a scenario 3 reactor accident, is less than 100 mrem. The assumed meteorological conditions used in the estimates are considered to be upper bound conditions. Less severe impacts would result from an accident involving a Type A package; these impacts would be similar to those for such an accident during decommissioning following accident cleanup, as discussed in Section J.3.3.2 of this appendix.

Rail transport of spent fuel and fuel debris from the final reactor core is assumed to employ the IF-300 cask, which can hold 7 PWR fuel assemblies or 4 fuel-assembly canisters. The IF-300 cask is a thick-walled, water-filled container designed to provide safe transport of spent fuel, with design integrity to withstand most transport accident situations. The cask is licensed to withstand Type B package tests. Results of cask test programs show that this cask can withstand all but the most severe, highly unusual types of accidents.[15,16,17] For a release to occur during spent fuel transport, radioactive material must leave both the fuel cladding (or, for damaged fuel, the fuel-assembly canister) and the cask containment. Since the transportation of spent fuel is not unique to accident cleanup and decommissioning, and since the probabilities of accidents that lead to atmospheric releases of radionuclides during spent fuel transport are so low, no further analysis or dose calculations are presented in this study. A more complete discussion of the impact of spent fuel transportation accidents on public safety is given in Reference 18.

J.2.3.3 Nonradiological Safety Aspects of Transportation Activities During Accident Cleanup

As with any transportation task, a certain potential for accidental injury or death exits from transportation accidents during accident cleanup activities at the reference PWR. Estimates are made here based on accident frequency data presented in Reference 13, and the results are shown in Table J.2-16. 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 2 or scenario 1 accident are estimated to be lower by factors of about 2 or 6, respectively. In all cases, casualties from truck transport are estimated to be much greater than those from rail transport because of the greater number of truck shipments and the higher incidence of truck accidents per vehicle-kilometer.

J.3 DECOMMISSIONING

Decommissioning activities at the reference PWR follow completion of accident cleanup activities at the plant. Accident cleanup efforts contribute to the total decommissioning effort, but in this study, accident cleanup and decommissioning are addressed separately.

This section contains the details of the analysis of the safety impacts resulting from the post-accident decommissioning activities at the reference PWR. Radiological and nonradiological impacts of both routine activities and selected generic industrial and transportation accidents are considered. Radiological safety impacts to the public are assessed in Section J.3.1. Occupational safety impacts from post-accident decommissioning activities are discussed in Section J.3.2. Transportation safety impacts, both public and occupational, are addressed in Section J.3.3.

J.3.1 Public Safety Aspects of Post-Accident Decommissioning Activities

The public safety impacts from onsite activities during post-accident decommissioning (following completion of accident cleanup activities) are

TABLE J.2-16. Estimated Casualties from Transportation Accidents During Accident Cleanup at the Reference PWR

Transportation Category	Accident Frequency Data[a]			Number of Shipments	Total Round Trip Distances (km)[b]	Transportation Casualties	
	Accidents per Vehicle km	Injuries per Accident	Fatalities per Accident			Injuries	Fatalities
Accident Scenario 1							
Rail Transport	8.7×10^{-8}	2.7	0.2	30	9.6×10^{4}	0.023	0.0017
Truck Transport	1.0×10^{-6}	0.51	0.03	90	2.9×10^{5}	0.15	0.0087
Totals				120	3.8×10^{5}	0.17	0.010
Accident Scenario 2							
Rail Transport	8.7×10^{-8}	2.7	0.2	50	1.6×10^{5}	0.038	0.0028
Truck Transport	1.0×10^{-6}	0.51	0.03	289	9.2×10^{5}	0.47	0.028
Totals				339	1.1×10^{6}	0.51	0.030
Accident Scenario 3							
Rail Transport	8.7×10^{-8}	2.7	0.2	52	1.7×10^{5}	0.040	0.0030
Truck Transport	1.0×10^{-6}	0.51	0.03	652	2.1×10^{6}	1.1	0.063
Totals				704	2.3×10^{6}	1.1	0.066

(a) Based on data presented in Reference 13.
(b) Assuming 3200-km round-trip distance.
(c) Estimates are rounded to two significant figures.

discussed in the following subsections. Public radiation doses from atmospheric releases that result from routine decommissioning tasks and postulated industrial accidents during decommissioning are considered. Nonradiological safety impacts to the public from onsite activities are judged to be negligible and are not considered further. Public safety impacts from offsite shipment of radioactive waste materials during decommissioning are included in the assessment of transportation safety impacts, presented in Section J.3.3.

During decommissioning, as during accident cleanup, the routine tasks and postulated industrial accidents can generate airborne radioactivity in the plant. Contamination control measures, where applied, and HEPA filters in plant ventilation systems reduce the levels of radioactivity in the air that leaves the plant. The radioactivity released depends on the specific task or industrial accident considered and on the corresponding radionuclide inventory at that particular location.

The radionuclide inventories used in this study for post-accident decommissioning are the same as those used in Reference 2 for decommissioning following normal reactor shutdown, except that the fission-product contamination inventory is adjusted to reflect the additional fission-product contamination in the plant resulting from the postulated reactor accident.

In the following subsections, the atmospheric releases and resulting radiation doses to the public are discussed. Analyses are performed for decommissioning following a scenario 2 reactor accident, and the effects of variations in the severity of the postulated reactor accident are discussed. The atmospheric releases are estimated by determining the realistic maximum atmospheric release for each situation and then using this value whenever similar conditions occur, even for areas with lower levels of radioactive contamination.

J.3.1.1 Public Radiation Doses from Routine Tasks During Post-Accident Decommissioning

A complete discussion of the tasks required for post-accident decommissioning of the reference PWR by either the DECON, SAFSTOR, or ENTOMB alternative is contained in Appendix G. To quantify the radiation doses to the public that result from these tasks, atmospheric releases of radioactivity are estimated and the resulting doses to the maximum-exposed individual and the population are calculated on the basis of the particular radionuclide inventory involved.

The atmospheric releases of radioactivity for post-accident decommissioning are based on estimated values for decommissioning the reference PWR following normal reactor shutdown, presented in Appendix J of Reference 2. The radionuclide inventory for fission-product contamination and, thus, the dose conversion factors for this inventory are adjusted to account for the increased fission-product contamination resulting from the postulated reactor accident. Each of the atmospheric releases is assumed to be a chronic release (i.e., one that occurs at a uniform rate over a one-year period) to allow direct comparisons of the impacts from individual decommissioning tasks. The first-year dose and the fifty-year committed dose equivalents to both the maximum-exposed individual and to the population residing within 80 km of the site are calculated. The dose calculation includes direct exposure, inhalation, and ingestion pathways.

The estimated atmospheric releases of radioactivity and the resulting doses to the maximum-exposed individual from routine tasks during DECON and during preparations for safe storage are shown in Tables J.3-1 and J.3-2, respectively. The releases and the resulting doses to the population residing within 80 km of the site from these routine tasks during DECON and during preparations for safe storage are shown in Tables J.3-3 and J.3.4, respectively. The reference radionuclide inventories designated in the table are those shown in Reference 2 for decommissioning following normal reactor shutdown, except for reference radionuclide inventory 5 which is assumed to be dominated by the accident-generated fission-product contamination and thus exhibits the same makeup as the inventory shown in Appendix C of this study following a scenario 2 accident. The values shown in the tables are for decommissioning following a scenario 2 accident and, for those tasks involving reference radionuclide inventory 5, would be increased or decreased somewhat following a scenario 3 or a scenario 1 accident, respectively.

TABLE J.3-1. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine DECON Tasks

Operation or Location	Reference Radionuclide Inventory Number[a]	Release to Atmosphere (uCi)	First-Year Dose (rem) Total-Body	First-Year Dose (rem) GI-LLI	First-Year Dose (rem) Bone	First-Year Dose (rem) Lung	Fifty-Year Committed Dose Equivalent (rem) Total-Body	Fifty-Year Committed Dose Equivalent (rem) GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Bone	Fifty-Year Committed Dose Equivalent (rem) Lung
Segmentation of Nonactivated Stainless										
Coolant Pumps and Primary Piping	4	7.5×10^{1}	1.7×10^{-8}	3.1×10^{-8}	1.6×10^{-8}	4.3×10^{-8}	1.7×10^{-8}	3.1×10^{-8}	1.6×10^{-8}	5.8×10^{-8}
Steam Generators	4	4.0×10^{-1}	9.1×10^{-11}	1.6×10^{-10}	8.6×10^{-11}	2.3×10^{-10}	9.2×10^{-11}	1.6×10^{-10}	8.6×10^{-11}	3.1×10^{-10}
Segmentation of Activated Reactor										
Internals	1	5.0×10^{-1}	1.1×10^{-10}	1.8×10^{-10}	1.1×10^{-10}	2.5×10^{-10}	1.1×10^{-10}	1.8×10^{-10}	1.4×10^{-10}	3.5×10^{-10}
Vessel	2	2.0×10^{-1}	1.2×10^{-11}	2.1×10^{-11}	1.2×10^{-11}	3.0×10^{-11}	1.2×10^{-11}	2.1×10^{-11}	1.6×10^{-11}	4.2×10^{-11}
Waste Handling Bioshield Concrete	3	1.8×10^{0}	4.7×10^{-11}	7.7×10^{-11}	5.1×10^{-10}	8.0×10^{-11}	1.1×10^{-9}	7.7×10^{-11}	1.1×10^{-9}	1.6×10^{-9}
Surface Cleaning Operations										
Hand Held Lance										
HX Room	5[b]	2.0×10^{1}	1.9×10^{-8}	5.6×10^{-9}	1.8×10^{-8}	9.6×10^{-9}	3.8×10^{-8}	5.6×10^{-9}	9.4×10^{-8}	2.0×10^{-8}
Evaporator Room	5	5.0×10^{0}	4.7×10^{-9}	1.4×10^{-9}	4.5×10^{-9}	2.4×10^{-9}	9.5×10^{-9}	1.4×10^{-9}	2.4×10^{-8}	5.0×10^{-9}
Reactor Cavity	5	1.0×10^{0}	9.4×10^{-10}	2.8×10^{-10}	8.9×10^{-10}	4.8×10^{-10}	1.9×10^{-9}	2.8×10^{-10}	4.7×10^{-9}	1.0×10^{-9}
Steam Generator Area	5	1.0×10^{0}	9.4×10^{-10}	2.8×10^{-10}	8.9×10^{-10}	4.8×10^{-10}	1.9×10^{-9}	2.8×10^{-10}	4.7×10^{-9}	1.0×10^{-9}
Final Chemical Decontamination	4	7.8×10^{-1}	1.8×10^{-10}	3.2×10^{-10}	1.7×10^{-10}	4.5×10^{-10}	1.8×10^{-10}	3.2×10^{-10}	1.7×10^{-10}	6.0×10^{-10}
Ion Exchanger Vault	5	2.5×10^{-3}	2.4×10^{-12}	7.0×10^{-13}	2.2×10^{-12}	1.2×10^{-12}	4.8×10^{-12}	7.0×10^{-13}	1.2×10^{-11}	2.5×10^{-12}
Laundry Room	5	9.5×10^{-6}	8.9×10^{-15}	2.7×10^{-15}	8.5×10^{-15}	4.6×10^{-15}	1.8×10^{-14}	2.7×10^{-15}	4.5×10^{-14}	9.5×10^{-15}
In Situ Chemical Decontamination										
Spray Leak	5	3.5×10^{0}	3.3×10^{-9}	9.8×10^{-10}	3.1×10^{-9}	1.7×10^{-9}	6.7×10^{-9}	9.8×10^{-10}	1.6×10^{-8}	3.5×10^{-9}
Liquid Leak	5	1.2×10^{-2}	1.1×10^{-11}	3.4×10^{-12}	1.1×10^{-11}	5.8×10^{-12}	2.3×10^{-11}	3.4×10^{-12}	5.6×10^{-11}	1.2×10^{-11}
Removal of Bioshield										
Explosive	3	5.3×10^{-3}	1.4×10^{-13}	2.3×10^{-13}	1.5×10^{-12}	2.4×10^{-13}	3.4×10^{-12}	2.3×10^{-13}	3.2×10^{-12}	4.7×10^{-12}
Drilling	3	6.0×10^{-5}	1.6×10^{-15}	2.6×10^{-15}	1.7×10^{-14}	2.7×10^{-15}	3.8×10^{-14}	2.6×10^{-15}	3.6×10^{-14}	5.3×10^{-14}
Radiation Survey										
HX Room	5	1.0×10^{0}	9.4×10^{-10}	2.8×10^{-10}	8.9×10^{-10}	4.8×10^{-10}	1.9×10^{-9}	2.8×10^{-10}	4.7×10^{-9}	1.0×10^{-9}
Boric Acid Evap. Room	5	2.5×10^{-1}	2.4×10^{-10}	7.0×10^{-11}	2.2×10^{-10}	1.2×10^{-10}	4.8×10^{-10}	7.0×10^{-10}	1.2×10^{-9}	2.5×10^{-10}
Steam Generator Area	5	9.5×10^{-2}	8.9×10^{-11}	2.7×10^{-11}	8.5×10^{-11}	4.6×10^{-11}	1.8×10^{-10}	2.7×10^{-11}	4.5×10^{-10}	9.5×10^{-11}
Laundry Room	5	5.0×10^{-7}	4.7×10^{-16}	1.4×10^{-16}	4.5×10^{-16}	2.4×10^{-16}	9.5×10^{-16}	1.4×10^{-16}	2.4×10^{-15}	5.0×10^{-16}
Removal of Concrete Areas										
Explosives	5	5.0×10^{-5}	4.7×10^{-14}	1.4×10^{-14}	4.5×10^{-14}	2.4×10^{-14}	9.5×10^{-13}	1.4×10^{-14}	2.4×10^{-13}	5.0×10^{-14}
Drilling	5	1.0×10^{-7}	9.4×10^{-15}	2.8×10^{-15}	8.9×10^{-15}	4.8×10^{-15}	1.9×10^{-13}	2.8×10^{-15}	4.7×10^{-14}	1.0×10^{-14}
Pneumatic Jack Hammer	5	-[c]								
Rock Splitters	5	-[c]								

(a) From Appendix J of Reference 2.
(b) Reference radionuclide inventory 5 same as inventory following scenario 2 accident, see Appendix C.
(c) A dash means that the atmospheric release value is less than 1×10^{-7} uCi.

TABLE J.3-2. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine Preparations for Safe Storage Tasks

Operation or Location	Reference Radionuclide Inventory Number[a]	Release to Atmosphere (uCi)	First-Year Dose (rem) Total Body	First-Year Dose (rem) GI-LLI	First-Year Dose (rem) Bone	First-Year Dose (rem) Lung	Fifty-Year Committed Dose Equivalent (rem) Total-Body	Fifty-Year Committed Dose Equivalent (rem) GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Bone	Fifty-Year Committed Dose Equivalent (rem) Lung
Surface Cleaning Operations										
Hand Held Lance										
Primary System	5[b]	5.0×10^{0}	4.7×10^{-9}	1.4×10^{-9}	4.5×10^{-9}	2.4×10^{-9}	9.5×10^{-9}	1.4×10^{-9}	2.4×10^{-8}	5.0×10^{-9}
Evaporator Room	5	5.0×10^{0}	4.7×10^{-9}	1.4×10^{-9}	4.5×10^{-9}	2.4×10^{-9}	9.5×10^{-9}	1.4×10^{-9}	2.4×10^{-8}	5.0×10^{-9}
Reactor Cavity	5	1.0×10^{0}	9.4×10^{-10}	2.8×10^{-10}	8.9×10^{-10}	4.8×10^{-10}	1.9×10^{-9}	2.8×10^{-10}	4.7×10^{-9}	1.0×10^{-9}
Steam Generator Area	5	1.0×10^{0}	9.4×10^{-10}	2.8×10^{-10}	8.9×10^{-10}	4.8×10^{-10}	1.9×10^{-9}	2.8×10^{-10}	4.7×10^{-9}	1.0×10^{-9}
Laundry Room	5	9.5×10^{-6}	8.9×10^{-15}	2.7×10^{-15}	8.5×10^{-15}	4.6×10^{-15}	1.8×10^{-14}	2.7×10^{-15}	4.5×10^{-14}	9.5×10^{-15}
Sweeping	5	2.0×10^{-1}	1.9×10^{-10}	5.6×10^{-11}	1.8×10^{-10}	9.6×10^{-10}	3.8×10^{-10}	5.6×10^{-11}	9.4×10^{-10}	2.0×10^{-10}
Vacuum Alternate	5	2.5×10^{-3}	2.4×10^{-12}	7.0×10^{-13}	2.2×10^{-12}	1.2×10^{-12}	4.8×10^{-12}	7.0×10^{-13}	1.2×10^{-11}	2.5×10^{-12}
In Situ Chemical Decontamination										
Spray Leak	5	3.5×10^{0}[c]	3.3×10^{-9}	9.8×10^{-10}	3.1×10^{-9}	1.7×10^{-9}	6.7×10^{-9}	9.8×10^{-10}	1.6×10^{-8}	3.5×10^{-9}
Liquid Leak	5	1.2×10^{-2}[c]	1.1×10^{-11}	3.4×10^{-12}	1.1×10^{-11}	5.8×10^{-12}	2.3×10^{-11}	3.4×10^{-12}	5.6×10^{-11}	1.2×10^{-11}
Radiation Survey										
HX Room	5	1.0×10^{0}	9.4×10^{-10}	2.8×10^{-10}	8.9×10^{-10}	4.8×10^{-10}	1.9×10^{-9}	2.8×10^{-10}	4.7×10^{-9}	1.0×10^{-9}
Boric Acid Evap. Room	5	2.5×10^{-1}	2.3×10^{-10}	7.0×10^{-11}	2.2×10^{-10}	1.2×10^{-10}	4.8×10^{-10}	7.0×10^{-11}	1.1×10^{-9}	2.5×10^{-10}
Steam Generator Area	5	9.5×10^{-2}	8.9×10^{-11}	2.7×10^{-11}	8.5×10^{-11}	4.6×10^{-11}	1.8×10^{-10}	2.7×10^{-11}	4.5×10^{-10}	9.5×10^{-11}
Laundry Room	5	5.0×10^{-7}	4.7×10^{-16}	1.4×10^{-16}	4.5×10^{-16}	2.4×10^{-16}	9.5×10^{-16}	1.4×10^{-16}	2.4×10^{-15}	5.0×10^{-16}
Removal of Concrete Areas										
Drilling	5	-[d]								
Pneumatic Jack Hammer	5	-[d]								
Rock Splitters	5	-[d]								
Onsite Retrievable Waste Storage	5	7.5×10^{1}	7.1×10^{-8}	2.1×10^{-8}	6.7×10^{-8}	3.4×10^{-8}	1.4×10^{-7}	2.1×10^{-8}	3.6×10^{-7}	7.5×10^{-8}

(a) From Appendix J of Reference 2.
(b) Reference radionuclide inventory 5 same as inventory following scenario 2 accident, see Appendix C.
(c) In situ decontamination not necessarily used.
(d) A dash means that the atmospheric release value is less than 1×10^{-7} uCi.

TABLE J.3-3. Radiation Doses to the Population from Atmospheric Releases During Routine DECON Tasks[a]

Operation or Location	Reference Radionuclide Inventory Number[b]	Release to Atmosphere (uCi)	First-Year Dose (man-rem) Total-Body	GI-LLI	Bone	Lung	Fifty-Year Committed Dose Equivalent (man-rem) Total-Body	GI-LLI	Bone	Lung
Segmentation of Nonactivated Stainless										
Coolant Pumps and Primary Piping	4	7.5×10^{1}	1×10^{-5}	2×10^{-5}	9×10^{-6}	4×10^{-5}	1×10^{-5}	2×10^{-5}	9×10^{-6}	6×10^{-5}
Steam Generators	4	4.0×10^{-1}	5×10^{-8}	1×10^{-7}	5×10^{-8}	2×10^{-7}	5×10^{-8}	1×10^{-7}	5×10^{-8}	3×10^{-7}
Segmentation of Activated Reactor										
Internals	1	5.0×10^{-1}	6×10^{-8}	1×10^{-7}	6×10^{-8}	2×10^{-7}	6×10^{-8}	1×10^{-7}	8×10^{-8}	3×10^{-7}
Vessel	2	2.0×10^{-1}	7×10^{-9}	1×10^{-8}	7×10^{-9}	3×10^{-8}	7×10^{-9}	1×10^{-8}	1×10^{-8}	4×10^{-8}
Waste Handling Bioshield Concrete	3	1.8×10^{0}	2×10^{-7}	3×10^{-7}	6×10^{-7}	3×10^{-7}	1×10^{-6}	3×10^{-7}	1×10^{-6}	2×10^{-6}
Surface Cleaning Operations										
Hand Held Lance										
HX Room	5[c]	2.0×10^{1}	1×10^{-5}	3×10^{-6}	1×10^{-5}	8×10^{-6}	3×10^{-5}	3×10^{-6}	8×10^{-5}	2×10^{-5}
Evaporator Room	5	5.0×10^{0}	3×10^{-6}	9×10^{-7}	3×10^{-6}	2×10^{-6}	7×10^{-6}	9×10^{-7}	2×10^{-5}	5×10^{-6}
Reactor Cavity	5	1.0×10^{0}	7×10^{-7}	2×10^{-7}	7×10^{-7}	4×10^{-7}	1×10^{-6}	2×10^{-7}	4×10^{-6}	1×10^{-6}
Steam Generator Area	5	1.0×10^{0}	7×10^{-7}	2×10^{-7}	7×10^{-7}	4×10^{-7}	1×10^{-6}	2×10^{-7}	4×10^{-6}	1×10^{-6}
Final Chemical Decontamination	4	7.8×10^{-1}	5×10^{-7}	1×10^{-7}	5×10^{-7}	3×10^{-7}	1×10^{-6}	1×10^{-7}	3×10^{-6}	7×10^{-7}
Ion Exchanger Vault	5	2.5×10^{-3}	2×10^{-9}	4×10^{-10}	2×10^{-9}	1×10^{-9}	4×10^{-9}	4×10^{-10}	1×10^{-8}	2×10^{-9}
Laundry Room	5	9.5×10^{-6}	7×10^{-12}	2×10^{-12}	6×10^{-12}	4×10^{-12}	1×10^{-11}	2×10^{-12}	4×10^{-11}	9×10^{-12}
In Situ Chemical Decontamination										
Spray Leak	5	3.5×10^{0}	2×10^{-6}	6×10^{-7}	2×10^{-6}	2×10^{-6}	5×10^{-6}	6×10^{-7}	1×10^{-5}	3×10^{-6}
Liquid Leak	5	1.2×10^{-2}	8×10^{-9}	2×10^{-8}	8×10^{-9}	5×10^{-9}	2×10^{-8}	2×10^{-8}	5×10^{-8}	1×10^{-8}
Removal of Bioshield										
Explosive	3	5.3×10^{-3}	7×10^{-10}	8×10^{-10}	2×10^{-9}	8×10^{-10}	3×10^{-9}	8×10^{-10}	3×10^{-9}	6×10^{-9}
Drilling	3	6.0×10^{-5}	8×10^{-12}	9×10^{-12}	2×10^{-11}	9×10^{-12}	3×10^{-11}	9×10^{-12}	3×10^{-11}	7×10^{-11}
Radiation Survey										
HX Room	5	1.0×10^{0}	7×10^{-7}	2×10^{-7}	7×10^{-7}	4×10^{-7}	1×10^{-6}	2×10^{-7}	4×10^{-6}	1×10^{-6}
Boric Acid Evap. Room	5	2.5×10^{-1}	2×10^{-7}	4×10^{-8}	2×10^{-7}	1×10^{-7}	4×10^{-7}	4×10^{-8}	1×10^{-6}	2×10^{-7}
Steam Generator Area	5	9.5×10^{-2}	7×10^{-8}	2×10^{-8}	6×10^{-8}	4×10^{-8}	1×10^{-7}	2×10^{-8}	4×10^{-7}	9×10^{-8}
Laundry Room	5	5.0×10^{-7}	3×10^{-13}	9×10^{-14}	3×10^{-13}	2×10^{-13}	7×10^{-13}	9×10^{-14}	2×10^{-12}	5×10^{-13}
Removal of Concrete Areas										
Explosives	5	5.0×10^{-5}	3×10^{-11}	9×10^{-12}	3×10^{-11}	2×10^{-11}	7×10^{-11}	9×10^{-12}	2×10^{-10}	5×10^{-11}
Drilling	5	1.0×10^{-5}	7×10^{-12}	2×10^{-12}	7×10^{-12}	4×10^{-12}	1×10^{-11}	2×10^{-12}	4×10^{-11}	1×10^{-11}
Pneumatic Jack Hammer	5	-[d]								
Rock Splitters	5	-[d]								

(a) The calculated radiation doses are shown to one significant figure.
(b) From Appendix J of Reference 2.
(c) A dash means that the atmospheric release value is less than 1×10^{-7} uCi.
(d) Reference radionuclide inventory 5 same as inventory following scenario 2 accident, see Appendix C.

TABLE J.3-4. Radiation Doses to the Population from Atmospheric Releases During Routine Preparations for Safe Storage Tasks[a]

Operation or Location[b]	Reference Radionuclide Inventory Number[b]	Release to Atmosphere (μCi)	First-Year Dose (man-rem) Total-Body	First-Year Dose (man-rem) GI-LLI	First-Year Dose (man-rem) Bone	First-Year Dose (man-rem) Lung	Fifty-Year Committed Dose Equivalent (man-rem) Total-Body	Fifty-Year Committed Dose Equivalent (man-rem) GI-LLI	Fifty-Year Committed Dose Equivalent (man-rem) Bone	Fifty-Year Committed Dose Equivalent (man-rem) Lung
Surface Cleaning Operations										
Hand Held Lance										
Primary System	5[c]	5.0×10^{0}	3×10^{-6}	9×10^{-7}	3×10^{-6}	2×10^{-6}	7×10^{-6}	9×10^{-7}	2×10^{-5}	5×10^{-6}
Reactor Cavity	5	1.0×10^{0}	7×10^{-7}	2×10^{-7}	7×10^{-7}	4×10^{-7}	1×10^{-6}	2×10^{-7}	4×10^{-6}	1×10^{-6}
Steam Generator Area	5	1.0×10^{0}	7×10^{-7}	2×10^{-7}	7×10^{-7}	4×10^{-7}	1×10^{-6}	2×10^{-7}	4×10^{-6}	1×10^{-6}
Laundry Room	5	9.5×10^{-6}	7×10^{-12}	2×10^{-12}	6×10^{-12}	4×10^{-12}	1×10^{-11}	2×10^{-12}	4×10^{-11}	9×10^{-12}
Sweeping	5	2.0×10^{-1}	1×10^{-7}	3×10^{-3}	1×10^{-7}	8×10^{-8}	3×10^{-7}	3×10^{-8}	8×10^{-7}	2×10^{-7}
Vacuum Alternate	5	2.5×10^{-3}	2×10^{-9}	4×10^{-10}	2×10^{-9}	1×10^{-9}	4×10^{-9}	4×10^{-10}	1×10^{-8}	2×10^{-9}
In Situ Decontamination										
Spray Leak	5	3.5×10^{0}(d)	2×10^{-6}	6×10^{-7}	2×10^{-6}	1×10^{-6}	5×10^{-6}	6×10^{-7}	1×10^{-5}	3×10^{-6}
Liquid Leak	5	1.2×10^{-2}(d)	8×10^{-9}	2×10^{-9}	8×10^{-9}	5×10^{-9}	2×10^{-8}	2×10^{-9}	5×10^{-8}	1×10^{-8}
Radiation Survey										
HX Room	5	1.0×10^{0}	7×10^{-7}	2×10^{-7}	7×10^{-7}	4×10^{-7}	1×10^{-6}	2×10^{-7}	4×10^{-6}	1×10^{-6}
Boric Acid Evap. Room	5	2.5×10^{-1}	2×10^{-7}	4×10^{-8}	2×10^{-7}	1×10^{-7}	4×10^{-7}	4×10^{-8}	1×10^{-6}	2×10^{-7}
Steam Generator Area	5	9.5×10^{-2}	7×10^{-8}	1×10^{-8}	6×10^{-8}	4×10^{-8}	1×10^{-7}	2×10^{-8}	4×10^{-7}	9×10^{-8}
Laundry Room	5	5.0×10^{-7}	3×10^{-13}	9×10^{-14}	3×10^{-13}	2×10^{-13}	7×10^{-13}	9×10^{-14}	2×10^{-12}	5×10^{-13}
Removal of Concrete Areas										
Drilling	5	-(e)								
Pneumatic Jack Hammer	5	-(e)								
Rock Splitters	5	-(e)								
Onsite Retrievable Waste Storage	5	7.5×10^{1}	5×10^{-5}	1×10^{-5}	5×10^{-5}	3×10^{-5}	1×10^{-4}	1×10^{-5}	3×10^{-4}	7×10^{-5}

(a) The calculated radiation doses are shown to one significant figure.
(b) From Appendix J of Reference 2.
(c) Reference radionuclide inventory 5 same as inventory following scenario 2 accident. see Appendix C.
(d) In situ chemical decontamination not necessarily used.
(e) A dash means that the atmospheric release value is less than 1×10^{-7} μCi.

No detailed safety analysis of ENTOMB is presented in Reference 19 and, thus, no detailed results are presented here. However, the releases and public doses for ENTOMB should be slightly less than those for DECON and significantly greater than those for preparations for safe storage, because of the similarities between the activities for DECON and those for ENTOMB.

J.3.1.2 Public Radiation Doses from Releases Due to Postulated Industrial Accidents During Post-Accident Decommissioning

During decommissioning, unexpected situations may arise that lead to the accidental atmospheric release of radioactivity from the plant. The industrial accident situations considered for post-accident decommissioning are the same as those considered in Reference 2 for decommissioning following normal reactor shutdown. As for the releases from routine tasks discussed in the previous subsection, the reference radionuclide inventories are the same as those in Reference 2 except for inventory 5 which exhibits the same makeup as the inventory following a scenario 2 accident, shown in Appendix C of this study.

Estimates of the releases of radioactivity due to postulated industrial accidents during DECON and preparations for safe storage following a scenario 2 reactor accident, together with the resulting first-year doses and fifty-year committed dose equivalents to the maximum-exposed individual, are shown in Tables J.3-5 and J.3-6. The values for releases involving reference radionuclide inventory 5 would be increased or decreased somewhat following a scenario 3 or a scenario 1 accident, respectively. The releases and corresponding doses for ENTOMB are assumed to be the same as those shown for DECON (Table J.3-5), with the deletion of those situations that arise from activities not undertaken during ENTOMB (e.g., blasting, segmenting of reactor pressure vessel).

It is beyond the scope of this study to evaluate every potential industrial accident situation that could lead to the release of radioactivity during decommissioning. However, 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.

TABLE J.3-5. Radiation Doses to the Maximum-Exposed Individual from Releases Due to Industrial Accidents During Post-Accident DECON

Incident	Reference Radionuclide Inventory Number[a]	Release to Atmosphere (uCi)	First-Year Dose (rem) Total-Body	First-Year Dose (rem) GI-LLI	First-Year Dose (rem) Bone	First-Year Dose (rem) Lung	Fifty-Year Committed Dose Equivalent (rem) Total-Body	Fifty-Year Committed Dose Equivalent (rem) GI-LLI	Fifty-Year Committed Dose Equivalent (rem) Bone	Fifty-Year Committed Dose Equivalent (rem) Lung	Expected Frequency of Occurrence[b]
Explosion of LPG Leaked from Front End Loader	5[c]	1.8×10^{4}	2.0×10^{-4}	1.4×10^{-5}	4.5×10^{-4}	6.1×10^{-4}	4.3×10^{-4}	1.4×10^{-5}	3.8×10^{-3}	1.2×10^{-3}	Low
Explosion of Oxyacetylene during Segmenting of Vessel Shell	2	3.6×10^{2}	4.3×10^{-8}	4.1×10^{-7}	6.6×10^{-8}	6.1×10^{-6}	6.9×10^{-6}	4.1×10^{7}	2.4×10	6.9×10^{-6}	Medium
Explosion and/or Fire of Ion Exchange Resin	5	1.9×10^{2}	2.1×10^{-6}	1.5×10^{-7}	4.8×10^{-6}	6.5×10^{-6}	4.6×10^{-6}	1.5×10^{-7}	4.0×10^{-5}	1.3×10^{-5}	Medium
Gross Leak during In Situ Chemical Decontamination											
Spray Leak	5	1.1×10^{2}	1.2×10^{-6}	8.6×10^{-8}	2.8×10^{-6}	3.8×10^{-6}	2.6×10^{-6}	8.6×10^{-8}	2.3×10^{-5}	7.5×10^{-6}	Medium
Liquid Leak	5	3.5×10^{-1}	3.9×10^{-9}	2.7×10^{-10}	8.8×10^{-9}	1.2×10^{-8}	8.4×10^{-9}	2.7×10^{-10}	7.4×10^{-8}	2.4×10^{-8}	Medium
Segmentation of RCS Piping with Unremoved Contamination	4	1.1×10^{1}	4.6×10^{-9}	5.5×10^{-8}	5.0×10^{-9}	7.3×10^{-7}	4.8×10^{-9}	5.5×10^{-8}	5.1×10^{-9}	7.9×10^{-7}	High
Loss of Contamination Control Envelope during Oxyacetylene Cutting of Vessel Shell	2	2.3×10^{0}	2.8×10^{-10}	2.6×10^{-9}	4.2×10^{-10}	3.9×10^{-10}	4.5×10^{-10}	2.6×10^{-9}	1.6×10^{-9}	4.4×10^{-8}	Medium
Pressure Surge Damage to Filters during Blasting of Activated Concrete Bioshield	3	3.0×10^{-1}	1.3×10^{-11}	1.4×10^{-10}	4.5×10^{-11}	2.0×10^{-9}	3.6×10^{-11}	1.4×10^{-10}	2.0×10^{-10}	2.2×10^{-9}	Low
Loss of Integrity of Portable Filtered Ventilation Enclosure	5	1.5×10^{-1}	1.7×10^{-9}	1.2×10^{-10}	3.8×10^{-9}	5.1×10^{-9}	3.6×10^{-9}	1.2×10^{-10}	3.2×10^{-8}	1.0×10^{-8}	Medium
Fire Involving Contaminated Clothing or Combustible Waste	5	3.0×10^{-2}	3.3×10^{-10}	2.3×10^{-11}	7.5×10^{-10}	1.0×10^{-9}	7.2×10^{-10}	2.3×10^{-11}	6.3×10^{-9}	2.0×10^{-9}	Medium
Loss of Blasting Mat during Removal of Activated Concrete	3	2.7×10^{-3}	1.2×10^{-13}	1.3×10^{-12}	4.1×10^{-13}	1.8×10^{-11}	3.2×10^{-13}	1.3×10^{-12}	1.8×10^{-12}	2.0×10^{-11}	Medium
Detonation of Unused Explosives in Reactor Cavity	3	2.7×10^{-3}	1.2×10^{-13}	1.3×10^{-12}	4.1×10^{-13}	1.8×10^{-11}	3.2×10^{-13}	1.3×10^{-12}	1.8×10^{-12}	2.0×10^{-11}	Medium
Temporary Loss of Local Airborne Contamination Control During Blasting	3	1.4×10^{-6}	6.2×10^{-17}	6.7×10^{-16}	2.1×10^{-16}	9.2×10^{-15}	1.7×10^{-16}	6.7×10^{-16}	9.4×10^{-16}	1.0×10^{-14}	Low

(a) From Appendix J of Reference 2.
(b) Frequence of occurrence: High $>1 \times 10^{-2}$; Medium 1×10^{-2} to 1×10^{-5}; Low $<1 \times 10^{-5}$ per year.
(c) Reference radionuclide inventory 5 same as inventory following scenario 2 accident, see Appendix C.

TABLE J.3-6. Radiation Doses to the Maximum-Exposed Individual from Releases Due to Industrial Accidents During Post-Accident Preparations for Safe Storage

Incident	Reference Radionuclide Inventory Number[a]	Release to Atmosphere (μCi)	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)				Expected Frequency of Occurrence[b]
			Total-Body	GI-LLI	Bone	Lung	Total-Body	GI-LLI	Bone	Lung	
Gross Leak during In Situ Chemical Decontamination											
Spray Leak	5[c]	1.1×10^{2}	1.2×10^{-6}	8.6×10^{-8}	2.8×10^{-6}	3.8×10^{-6}	2.6×10^{-6}	8.6×10^{-8}	2.3×10^{-5}	7.5×10^{-6}	Low
Liquid Leak	5	3.5×10^{-1}	3.9×10^{-9}	2.7×10^{-10}	8.8×10^{-9}	1.2×10^{-8}	8.4×10^{-9}	2.7×10^{-10}	7.4×10^{-8}	2.4×10^{-8}	Low
Vacuum Bag Rupture	5	5.0×10^{0}	5.5×10^{-8}	3.9×10^{-9}	1.3×10^{-7}	1.7×10^{-7}	1.2×10^{-7}	3.9×10^{-9}	1.1×10^{-6}	3.4×10^{-7}	Medium
Accidental Spraying of Concentrated Contamination with High Pressure Spray	5	6.0×10^{-1}	6.6×10^{-9}	4.7×10^{-10}	1.5×10^{-8}	2.0×10^{-8}	1.4×10^{-8}	4.7×10^{-10}	1.3×10^{-7}	4.1×10^{-8}	High
Accidental Cutting of Contaminated Piping	4	1.8×10^{-1}	7.6×10^{-11}	9.0×10^{-10}	8.1×10^{-11}	1.2×10^{-8}	7.9×10^{-11}	9.0×10^{-10}	8.3×10^{-11}	1.3×10^{-8}	High
Accidental Break of Contaminated Piping during Inspection	4	1.1×10^{-1}	4.6×10^{-11}	5.5×10^{-10}	5.0×10^{-11}	7.3×10^{-9}	4.8×10^{-11}	5.5×10^{-10}	5.1×10^{-11}	7.9×10^{-9}	Low
Fire Involving Contaminated Clothing or Combustible Waste	5	3.0×10^{-2}	3.3×10^{-10}	2.3×10^{-11}	7.5×10^{-10}	1.0×10^{-9}	7.2×10^{-10}	2.3×10^{-11}	6.3×10^{-9}	2.0×10^{-9}	Medium
Fire in Contaminated Sweeping Compound	3	3.8×10^{-4}	4.2×10^{-12}	3.0×10^{-13}	9.5×10^{-12}	1.3×10^{-11}	9.1×10^{-12}	3.0×10^{-13}	8.0×10^{-11}	2.6×10^{-11}	Medium

(a) From Appendix J of Reference 2.
(b) Frequency of occurrence: High $>1 \times 10^{-2}$; Medium 1×10^{-2} to 1×10^{-5}; Low $<1 \times 10^{-5}$ per year.
(c) Reference radionuclide inventory 5 same as inventory following scenario 2 accident, see Appendix C.

J.3.2 Occupational Safety Aspects of Post-Accident Decommissioning

The occupational safety impacts of post-accident decommissioning activities at the reference PWR are discussed in the following subsections. Included are occupational radiation doses and potential industrial-accident (nonradiological) impacts to the decommissioning workers. The information developed here is based on the detailed description of decommissioning activities presented in Appendix G of this study.

J.3.2.1 Occupational Radiation Doses from Decommissioning Activities

Estimates of occupational radiation doses to decommissioning workers are developed in Appendix G and summarized here. These doses are estimated in the same manner as those for accident cleanup, discussed previously in Section J.2.2.1. Doses to transportation workers are presented in Section J.3.3.

Summaries of the estimated occupational radiation doses during DECON, preparations for safe storage, and ENTOMB following a scenario 2 accident are given in Tables J.3-7, J.3-8, and J.3-9, respectively. The doses shown could be increased by a factor of 2 to 3 following a scenario 3 accident or reduced by a similar factor following a scenario 1 accident. As shown in the tables, the total occupational doses from DECON are the largest and are about 7 times those from preparations for safe storage. The doses from ENTOMB are about 80% of those from DECON. In general, most of the occupational radiation dose from decommissioning results from activities in the containment building.

The estimated occupational radiation doses from the continuing care and deferred decontamination phases of SAFSTOR are shown in Table J.3-10, for safe storage periods of 30 and 100 years. The total doses from all phases of SAFSTOR are estimated by summing the doses for preparations for safe storage, continuing care, and deferred decontamination: total SAFSTOR doses following a scenario 2 accident are about 2050 man-rem with 30-year safe storage and about 950 man-rem with 100-year safe storage.

TABLE J.3-7. Summary of Estimated Occupational Radiation Doses from DECON at the Reference PWR Following a Scenario 2 Accident[a]

Activity Area	Estimated Occupational Doses (man-rem)				
	Supervisors	Utility Operators and Laborers	Craftsmen	Health Physics Technicians	Totals
Containment Building	317.16	1260.72	647.73	354.54	2580.15
Fuel Building	3.29	11.92	6.52	4.13	25.86
Auxiliary Building	16.87	23.95	37.76	17.47	96.05
Ancillaries	35.49	218.40	11.42	95.69	361.00
Totals	372.81	1514.99	703.43	471.83	3063.06

(a) Summarized from Table G.2-2 of Appendix G.

TABLE J.3-8. Summary of Estimated Occupational Radiation Doses from Preparations for Safe Storage at the Reference PWR Following a Scenario 2 Accident[a]

Activity Area	Estimated Occupational Doses (man-rem)				
	Supervisors	Utility Operators and Laborers	Craftsmen	Health Physics Technicians	Totals
Containment Building	30.92	63.11	36.26	41.16	171.45
Fuel Building	0.50	4.53	2.70	2.65	10.38
Auxiliary Building	2.29	3.97	2.79	10.59	19.64
Ancillaries	21.63	157.92	2.52	45.03	227.10
Totals	55.34	229.53	44.27	99.43	428.57

(a) Summarized from Table G.3-1 of Appendix G.

TABLE J.3-9. Summary of Estimated Occupational Radiation Doses from ENTOMB at the Reference PWR Following a Scenario 2 Accident[a]

Activity Area	Estimated Occupational Doses (man-rem)				
	Supervisors	Utility Operators and Laborers	Craftsmen	Health Physics Technicians	Totals
Containment Building	254.16	997.73	512.98	284.40	2049.27
Fuel Building	3.29	11.92	6.52	4.13	25.86
Auxiliary Building	16.87	23.95	37.76	17.47	96.05
Ancillaries	33.44	210.20	9.37	93.64	346.65
Totals	307.76	1243.80	566.63	399.64	2517.83

(a) Summarized from Table G.4-1 of Appendix G.

TABLE J.3-10. Summary of Estimated Occupational Radiation Doses During All Phases of SAFSTOR Following a Scenario 2 Accident

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 J.3-8.
(b) From Section G.3.5 of Appendix G.

J.3.2.2 Industrial Safety Aspects of Post-Accident Decommissioning

As discussed previously for accident cleanup (see Section J.2.2.2), estimates are made of injuries and fatalities resulting from industrial accidents during decommissioning, based on data given in Reference 11. The man-hours used to estimate these impacts are assumed to be the same as the exposure hours given in Appendix G, and are divided into three categories of accident potential:[12]

- heavy construction - primarily includes large scale removal of piping, equipment, and concrete
- light construction - includes minor removal tasks and some waste handling activities
- operational support - miscellaneous support activities required to complete the decommissioning.

For DECON and for preparations for safe storage, the percentages of time applicable to each category are assumed to be the same as for decommissioning following normal reactor shutdown, as shown in Table 11.3-6 of Reference 2. The percentage breakout for ENTOMB is assumed to be approximately the same as that for DECON.

The estimated worker injuries and fatalities for each of the three decommissioning alternatives following a scenario 2 reactor accident are presented in Table J.3-11. As shown in the table, less than 1 lost-time

TABLE J.3-11. Estimated Occupational Lost-Time Injuries and Fatalities During Post-Accident Decommissioning

Accident-Potential Category	Frequency (Accidents/man-hr)		DECON			Preparations for Safe Storage			ENTOMB		
	Lost-Time Injuries	Fatalities	man-hr[a]	Lost-Time Injuries	Fatalities	man-hrs[b]	Lost-Time Injuries	Fatalities	man-hrs[c]	Lost-Time Injuries	Fatalities
Heavy Construction	10×10^{-6}	4.2×10^{-8}	3.1×10^{4}	0.31	1.3×10^{-3}	NA[d]	--	--	3.1×10^{4}	0.31	1.3×10^{-3}
Light Construction	5.4×10^{-6}	3.0×10^{-8}	6.7×10^{4}	0.36	2.0×10^{-3}	1.9×10^{4}	0.10	5.7×10^{-4}	6.6×10^{4}	0.36	2.0×10^{-3}
Operational Support	2.1×10^{-6}	2.3×10^{-8}	5.5×10^{4}	0.12	1.3×10^{-3}	1.6×10^{4}	0.034	3.7×10^{-4}	5.4×10^{4}	0.11	1.2×10^{-3}
Totals			1.5×10^{5}	0.79	4.6×10^{-3}	3.5×10^{4}	0.13	9.4×10^{-4}	1.5×10^{5}	0.78	4.5×10^{-3}

(a) Summarized from Table G.2-2 of Appendix G.
(b) Summarized from Table G.3-1 of Appendix G.
(c) Summarized from Table G.4-1 of Appendix G.
(d) Heavy construction is not applicable to preparations for safe storage.

injury and less than 5 x 10^{-3} fatalities are estimated for decommissioning by any of the three alternatives, with the lowest accident potential for preparations for safe storage. The severity of the postulated reactor accident at the plant has little effect on the estimates of injuries and fatalities from industrial accidents because these estimates are based on exposure hours during decommissioning and, as discussed in Appendix G, exposure-hours for any specific decommissioning alternative do not vary significantly from accident scenario to accident scenario.

J.3.3 Transportation Safety Aspects of Post-Accident Decommissioning

Radioactive waste materials that result from decommissioning activities are assumed to be shipped offsite to a shallow-land burial site for disposal. The potential safety impacts of transportation activities are: 1) radiation doses to transport workers and to members of the public along the route, 2) radiation doses to the maximum-exposed individual from radiation releases during transport accidents, and 3) injuries and fatalities resulting from transportation accidents. The safety impacts of radioactive waste transportation during the post-accident decommissioning of the reference PWR are discussed in the following subsections. Estimates are based on the same assumptions used to calculate corresponding safety impacts of accident cleanup as presented in Section J.2.3.

J.3.3.1 Radiation Doses from Routine Transportation Activities During Decommissioning

Radioactive waste materials resulting from decommissioning activities are assumed to be shipped by exclusive-use truck to a shallow-land burial site 1600 km from the reference PWR. The method and the assumptions used to estimate radiation doses to transportation workers and to members of the public along the transportation route during decommissioning are the same as those used to estimate these doses during accident cleanup, as discussed in Section J.2.3.1 of this appendix.

Using these assumptions and the numbers of shipments during the various decommissioning alternatives (from Appendix H), radiation doses to transport workers and to the general public during decommissioning following a scenario 2 reactor accident are shown in Table J.3-12. As shown in the table,

TABLE J.3-12. Estimated Radiation Doses from Truck Shipments of Radioactive Materials During Post-Accident Decommissioning of the Reference PWR

Group	Radiation Dose per Shipment (man-rem)[a]	Number of Shipments	Total Radiation Dose (man-rem)[b]
DECON			
Truck Drivers	1.4×10^{-1}	1352[c]	190
Garagemen	6.7×10^{-3}	1352	9.1
Total Transportation Worker Dose			200
Onlookers	1.0×10^{-2}	1352	14
General Public	3.7×10^{-3}	1352	5.0
Total Public Dose			19
Preparations for Safe Storage			
Truck Drivers	1.4×10^{-1}	86[d]	12
Garagemen	6.7×10^{-3}	86	0.58
Total Transportation Worker Dose			13
Onlookers	1.0×10^{-2}	86	0.86
General Public	3.7×10^{-3}	86	0.32
Total Public Dose			1.2
ENTOMB			
Truck Drivers	1.4×10^{-1}	613[e]	86
Garagemen	6.7×10^{-3}	613	4.1
Total Transportation Worker Dose			90
Onlookers	1.0×10^{-2}	613	6.1
General Public	3.7×10^{-3}	613	2.3
Total Public Dose			8.4

(a) Based on one-way trips of 1600 km.
(b) All doses are rounded to two significant figures.
(c) From Table H.1-3 of Appendix H.
(d) From Table H.2-3 of Appendix H.
(e) From Table H.3-3 of Appendix H.

the doses during DECON are about 2 times those during ENTOMB and about 15 times those during preparations for safe storage. Changes in accident severity are estimated to result in only minor changes (± 10%) in the number of shipments required and, thus, only minor variations in these doses are estimated with changes in accident severity.

J.3.3.2 Radiation Doses from Postulated Transportation Accidents During Decommissioning

Transportation accidents during the offsite shipment of radioactive waste materials from decommissioning can potentially result in inadvertent releases of radioactivity and corresponding radiation doses to individuals near the accident location.

The method used to estimate doses resulting from transportation accidents is described in Section N.5 of Reference 20, which also discusses the probabilities of such accidents. The radioactive materials that are transported in Type B packages (highly activated reactor vessel internals and pressure vessel segments) are in solid, noncombustible forms that are not likely to become airborne in an accident. Therefore, no accident analysis of Type B packages is considered. Instead, two more realistic accidents involving combustible radioactive wastes in Type A packages are defined, both of which are judged to have a low frequency of occurrence. Both accidents are assumed to involve radioactive material dominated by accident-generated fission products and characterized in Appendix C of this study. For the minor accident, one waste package containing 1 curie of this radioactive inventory is assumed to rupture and burn while, for the severe accident, forty such packages are assumed to be involved. The assumed release fraction of the radioactivity is 5×10^{-4}.[21] The estimated resulting first-year radiation doses and fifty-year committed radiation dose equivalents to the maximum-exposed individual, assumed to be located 100 m downwind from the accident, are shown in Table J.3-13. As shown in the table, the fifty-year committed dose equivalent to the bone of the maximum-exposed individual is about 5 mrem from the minor accident and about 190 mrem from the severe accident. Assuming the same curie content per package for decommissioning following a scenario 1 or scenario 3 reactor accident, the doses from a

TABLE J.3-13. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Truck Transportation Accidents

Truck Transportation Accident	Total Release (Ci/hr)[a]	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
		Total-Body	Bone	Lung	GI-LLI	Total-Body	Bone	Lung	GI-LLI
Minor	5×10^{-4}	2.5×10^{-4}	6.0×10^{-4}	8.0×10^{-4}	1.8×10^{-5}	5.5×10^{-4}	4.8×10^{-3}	1.6×10^{-3}	1.8×10^{-5}
Severe	2×10^{-2}	1.0×10^{-2}	2.4×10^{-2}	3.2×10^{-2}	7.2×10^{-4}	2.2×10^{-2}	1.9×10^{-1}	6.4×10^{-2}	7.2×10^{-4}

(a) Released assumed to occur in a 1-hour period for comparison purposes.

transportation accident during decommissioning following either of these accidents are estimated to be within the same order of magnitude as those shown in Table J.3-13 for decommissioning following a scenario 2 reactor accident.

J.3.3.3 Nonradiological Safety Aspects of Transportation Activities During Decommissioning

As discussed previously for accident cleanup, any transportation task has a certain potential for accidents that could result in accidental injury or death. Estimates of casualties during decommissioning transportation activities are made here, based on accident frequency data from Reference 13, and the results are shown in Table J.3-14. As shown in the table, about 2.2 injuries and 0.13 fatalities are estimated for transportation accidents during DECON following a scenario 2 reactor accident. The corresponding values for preparations for safe storage and for ENTOMB are estimated to be lower by factors of about 15 and 2, respectively.

TABLE J.3-14. Estimated Casualties from Transportation Accidents During Post-Accident Decommissioning of the Reference PWR

Decommissioning Alternative	Accident Frequency Data[a]			No. of Shipments	Total Round-Trip Distances (km)[b]	Transportation Casualties[c]	
	Accidents per Vehicle km	Injuries per Accident	Fatalities per Accident			Injuries	Fatalities
DECON	1.0×10^{-6}	0.51	0.03	1352	4.3×10^{6}	2.2	0.13
Preparations for Safe Storage	1.0×10^{-6}	0.51	0.03	86	2.8×10^{5}	0.14	0.0084
ENTOMB	1.0×10^{-6}	0.51	0.03	613	2.0×10^{6}	1.0	0.060

(a) Based on data presented in Reference 13.
(b) Assuming 3200-km round-trip distance.
(c) Estimates are rounded to two significant figures.

REFERENCES

1. Final Programmatic Environmental Impact Statement Related to 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.

2. 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.

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

4. C. A. Burchsted, J. E. Kahn and A. B. Fuller, Nuclear Air Cleaning Handbook, ERDA 76-21, pp. 42-43, March 1976.

5. Final Elk River Reactor Program Report, COO-651-93, Revised, p. F-9, United Power Association, Elk River, Minnesota, November 1974.

6. Siting of Fuel Reprocessing and Waste Management Facilities, ORNL-4451, pp. 8-47, Oak Ridge National Laboratory, Oak Ridge, Tennessee, July 1970.

7. J. M. Selby, et al., Considerations in the Assessment of Consequences of Effluents from Mixed Oxide Fuel Fabrication Plants, BNWL-1697, Rev. 1, pp. 78-79, Pacific Northwest Laboratory, Richland, Washington, June 1975.

8. U.S. NRC, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, Regulatory Guide 1.109, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C., 1977.

9. U.S. Code of Federal Regulations, Title 10, Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," (10 CFR 50, App. I), U.S. Government Printing Office, Washington, D.C., 1980.

10. 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.

11. Operational Accidents and Radiation Exposure Experiences Within the U.S. AEC 1943-1970, WASH-1192, U.S. Atomic Energy Commission, Washington, D.C., 1971.

12. American National Standards Institute, Method of Recording and Measuring Work Injury Experience, ANSI 216.1, 1967.

13. Directorate of Regulatory Standards, Environmental Safety of Transportation of Radioactive Materials to and from Nuclear Power Plants, WASH-1238, U.S. Atomic Energy Commission, Washington, D.C., 1972.

14. U.S. Code of Federal Regulations, Title 49, Part 173.393, "General Packaging and Shipping Requirements," (49 CFR 173.393), U.S. Government Printing Office, Washington, D.C., 1978.

15. O. A. Kelly and W. C. Stoddart, Highway Vehicle Impact Studies: Tests and Mathematical Analyses of Vehicles, Package and Tiedown Systems Capable of Carrying Radioactive Material, ORNL-NSIC-61, Nuclear Safety Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, February 1970.

16. L. B. Shappert, et al., "The Full-Scale Cask Program at ORNL," International Seminar on the Design, Construction and Testing of Packaging for the Safe Transport of Radioactive Materials, IAEA-SR-10, International Atomic Energy Agency, Vienna, Austria, August 1976.

17. H. R. Yoshimura and M. Huerta, Full-Scale Tests of Spent Nuclear Fuel Shipping Systems, SAND-77-5707, Sandia Laboratories, Albuquerque, New Mexico, July 1976.

18. Technology for Commercial Radioactive Waste Management, DOE/ET-0028, Vol. 4, Chapter 6.2, Pacific Northwest Laboratory for the U.S. Department of Energy, Richland, Washington, May 1979.

19. 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.

20. 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.

21. J. Mishima and L. C. Schwendiman, Fractional Airborne Release of Uranium (Representing Plutonium) During the Burning of Contaminated Waste, BNWL-1730, Pacific Northwest Laboratory, Richland, Washington, April 1973.

APPENDIX K

DETAILS OF POST-ACCIDENT CLEANUP AND DECOMMISSIONING AT A REFERENCE BWR

This appendix provides detailed analyses of the technical requirements, estimated costs, and safety impacts of post-accident cleanup and decommissioning of a large boiling water reactor power station (BWR). Section K.1 contains a brief description of the reference BWR. Accident scenarios that provide the basis for the conceptual evaluation of accident cleanup and decommissioning are described in Section K.2. Details of activities, manpower requirements, and costs of accident cleanup are given in Sections K.3 and K.4. Details of activities, manpower requirements, and costs of decommissioning following accident cleanup are given in Sections K.5 and K.6. Safety assessment details are given in Section K.7.

K.1 REFERENCE BWR FACILITY DESCRIPTION

The reference BWR is the 3320-MWt (1155-MWe) nuclear power plant (WNP-2) being built by the Washington Public Power Supply System (WPPSS) at a site near Richland, Washington. The description of the WNP-2 reactor presented in this appendix is intended to provide the background for understanding the estimates of time and manpower requirements and costs for post-accident cleanup and decommissioning that are presented in other sections of this appendix and are summarized in Chapter 16 of Volume 1.

Additional details of the reference BWR nuclear power plant are given in Appendix C of Reference 1. The BWR facility description in Reference 1 is based on the Final Safety Analysis Report for WPPSS Nuclear Project No. 2 (WNP-2),(2) the WNP-2 Environmental Report,(3) and drawings and other data supplied by WPPSS personnel.

K.1.1 Plant Structures

The arrangement of the structures on the reference BWR plant site is shown in Figure K.1-1. The reactor building, the turbine generator building,

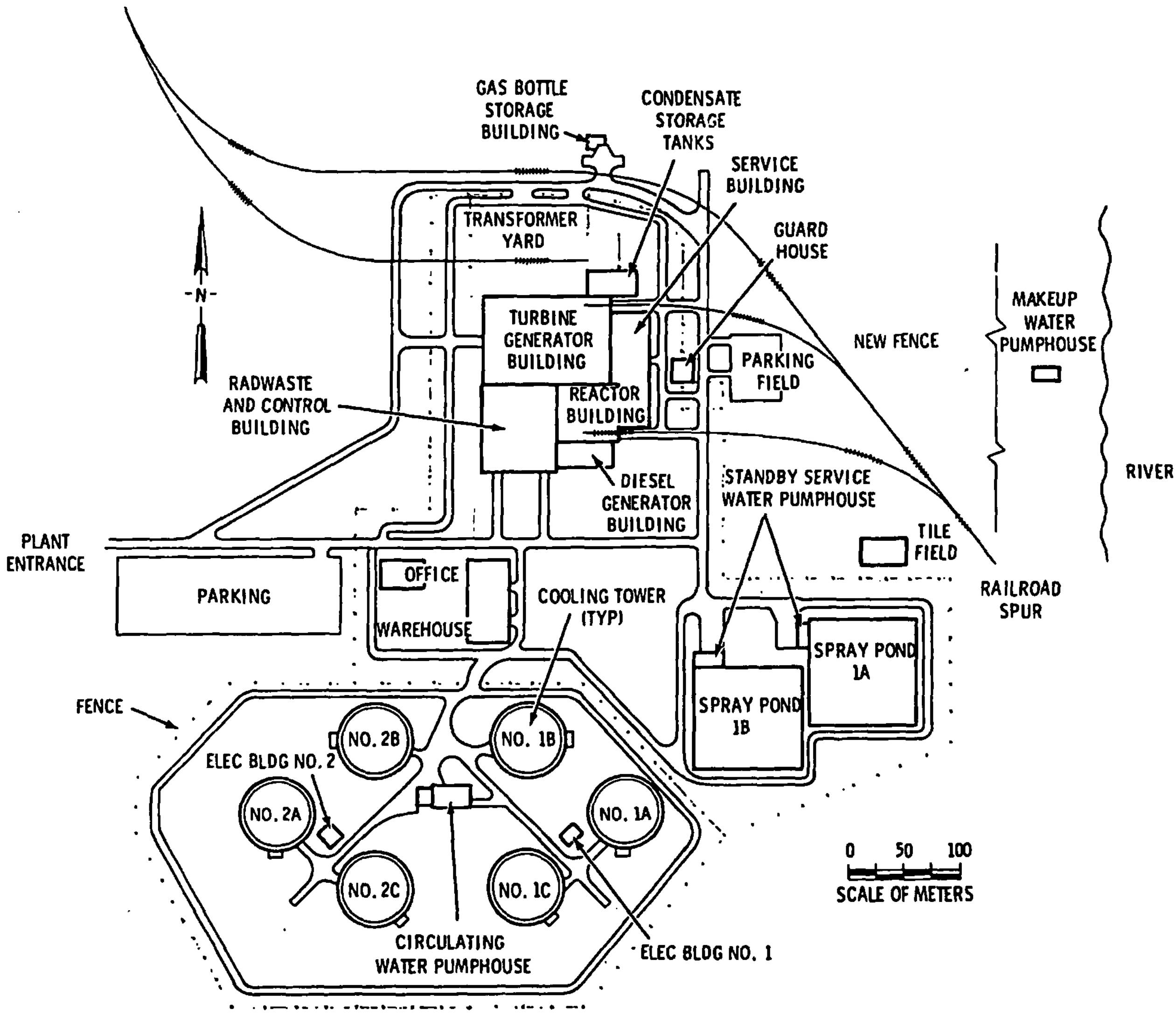


FIGURE K.1-1. Reference BWR Plant Layout

and the radwaste and control building are the buildings that, following a reactor accident, would require the major decontamination and decommissioning effort. Brief descriptions of these three buildings and of other structures on the site are given in this section.

K.1.1.1 Reactor Building

The reactor building is shown in cross section in Figures K.1-2 and K.1-3. Major building areas and structural components are identified in Figure K.1-2. The locations of major equipment items are shown in Figure K.1-3.

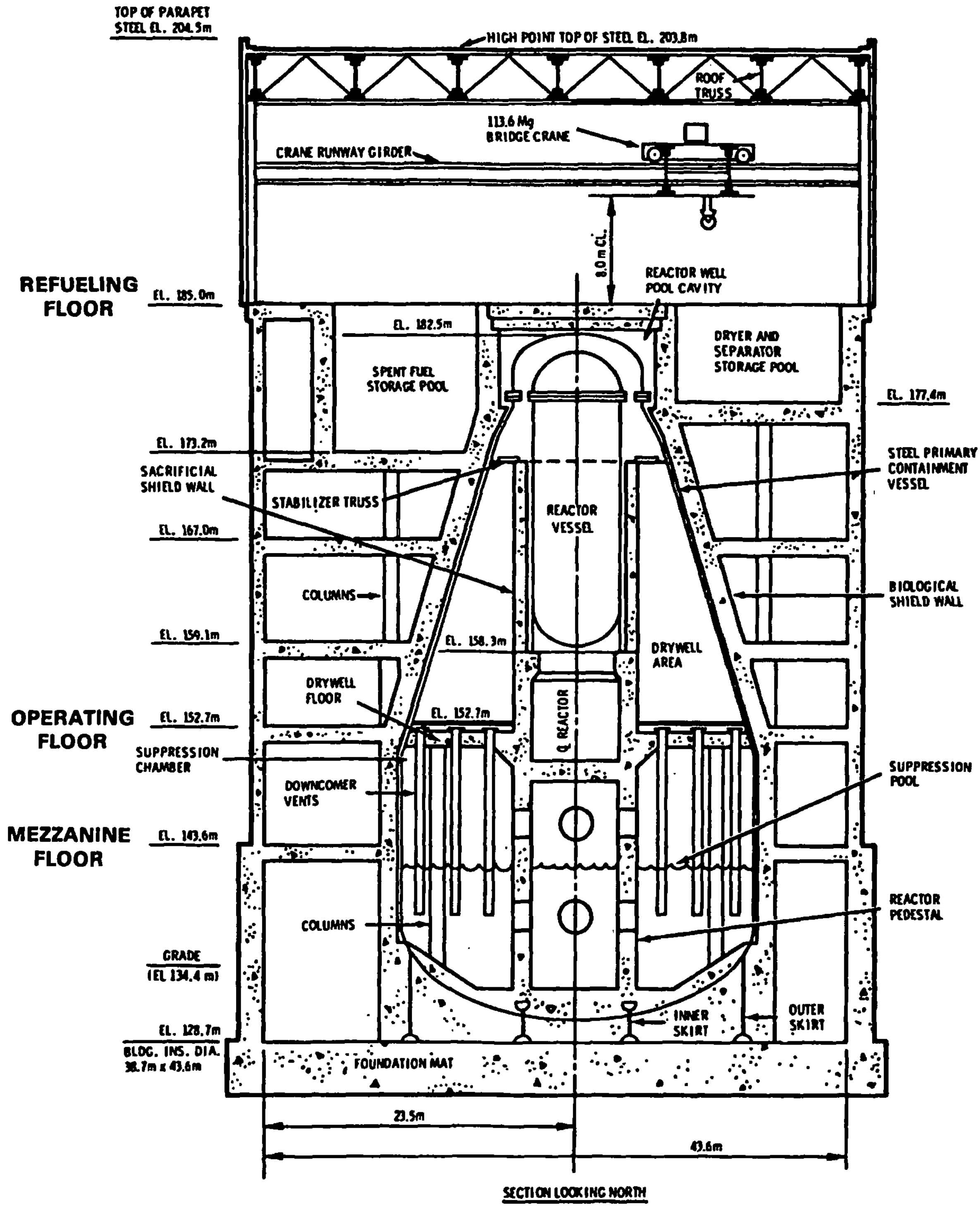


FIGURE K.1-2. Vertical Section of the BWR Reactor Building

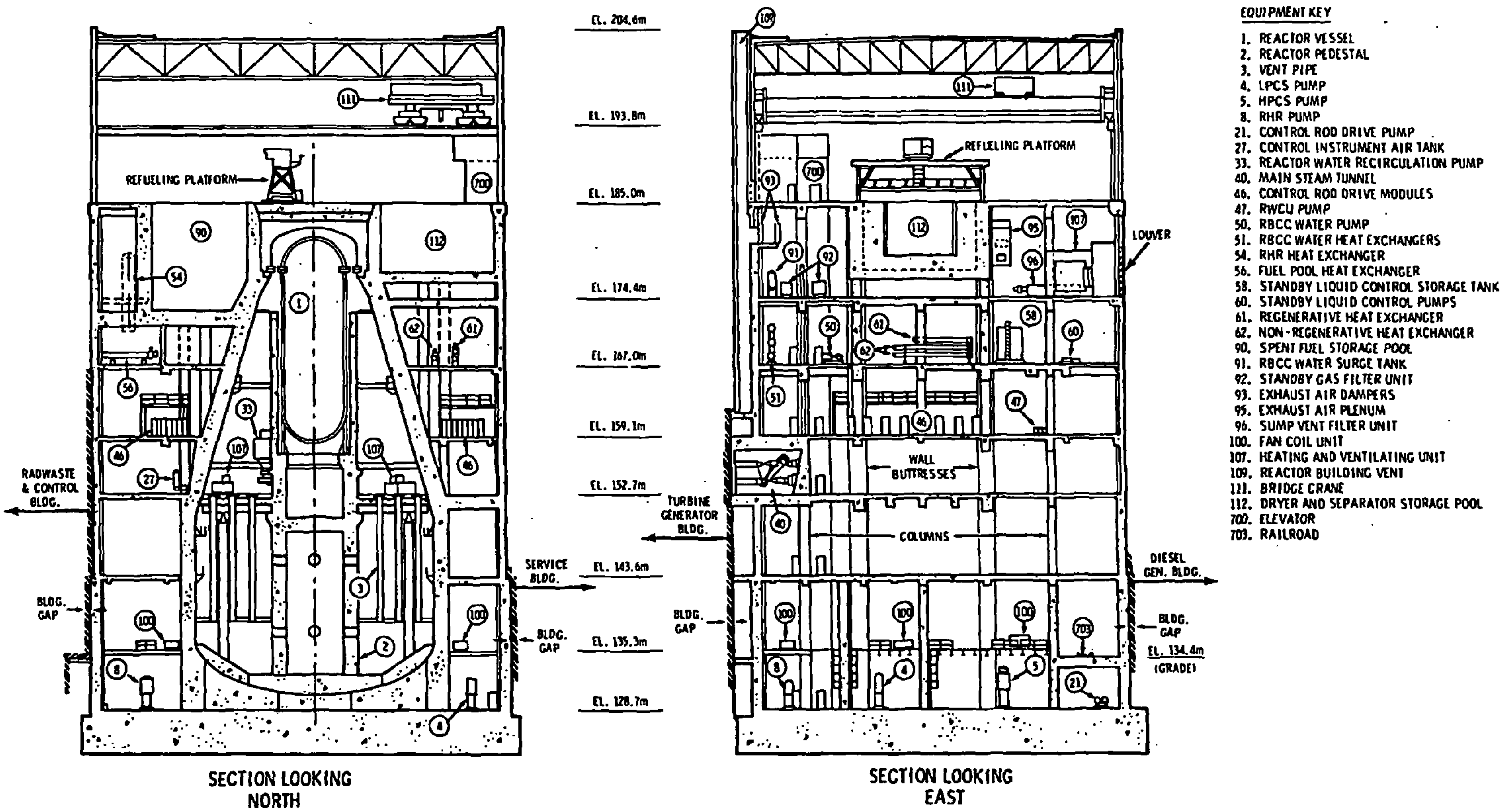


FIGURE K.1-3. Major Equipment Locations in the BWR Reactor Building

The reactor building houses the primary reactor system, reactor auxiliary and emergency cooling systems, and all the facilities required for refueling operations. It consists of two containment barriers: the steel primary containment vessel and the building itself, which provides secondary containment. The primary containment vessel is described in Section K.1.2. The secondary containment encloses the biological shield wall and houses refueling and reactor servicing equipment, fuel storage facilities, and other reactor safety and auxiliary systems.

The reactor building is rectangular in plan (about 39 m x 44 m) and in elevation (76 m high). The building is a cast-in-place, reinforced concrete structure up to and including the operating floor at elevation 185.0 m. Above the operating floor, the building is constructed of structural steel members with insulated metal siding and roof decking sealed to ensure leak-tightness. The steel superstructure supports the overhead bridge crane and houses refueling operations.

K.1.1.2 Turbine Generator Building

The turbine generator building houses the power conversion system equipment. The power conversion system converts the usable energy from the steam produced in the reactor vessel to electricity, condenses the steam, and heats the condensate and returns it to the reactor as feedwater. The system, shown in Figure K.1-4, consists of a large steam turbine and generator, moisture separator-reheaters, a single-pass condenser, motor-driven condensate and condensate booster pumps, a full-flow condensate demineralizer system, turbine-driven feedwater pumps, and six stages of feedwater heating.

The building exterior is reinforced concrete covered with metal siding from the foundation up to the operating floor level, elevation 152.7 m. Above the operating floor, the exterior walls are insulated metal siding supported by structural steel. The roof is insulated metal decking with built-up roofing. The building is rectangular in plan (∿58.8 m by ∿91.4 m) and in elevation (∿42.5 m high). The building structure and the reinforced concrete turbine generator pedestal are supported on a common foundation mat constructed of reinforced concrete 2.74 m to 3.66 m thick.

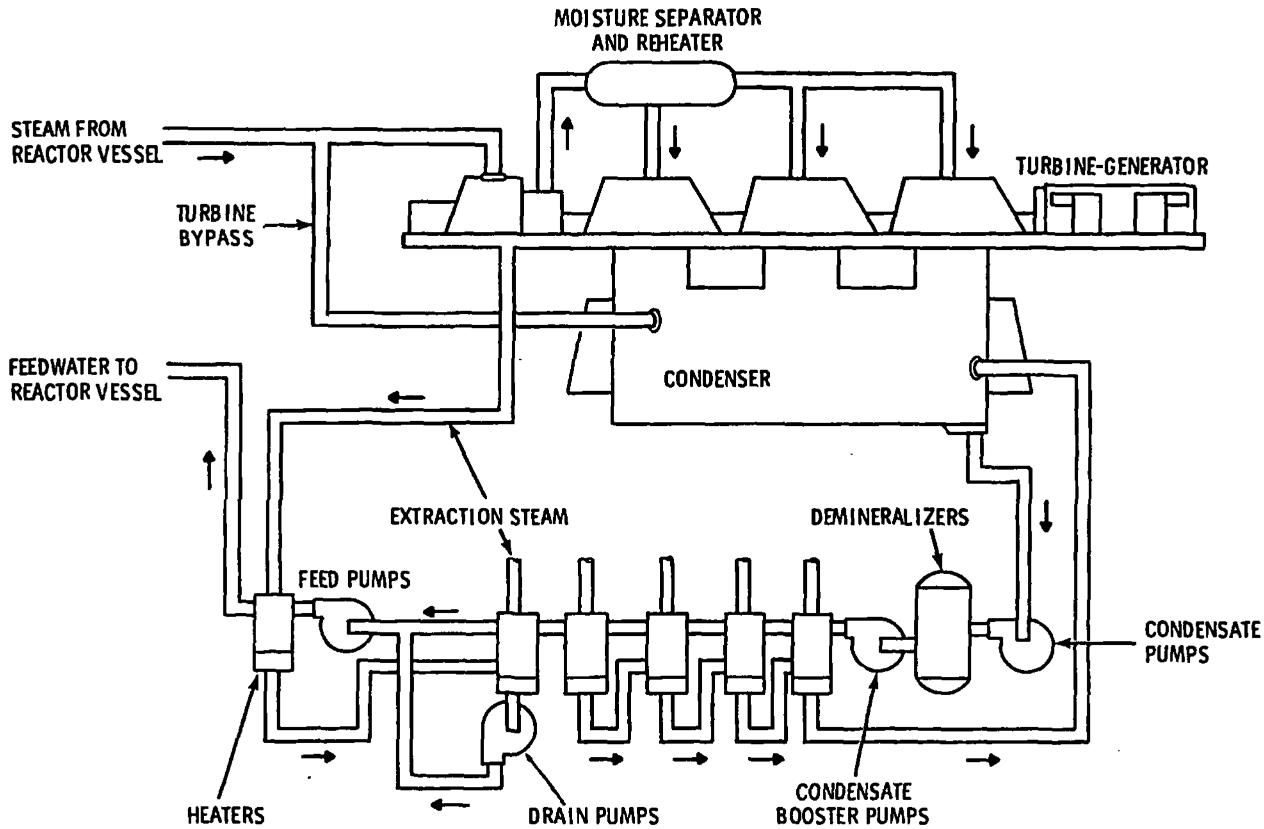


FIGURE K.1-4. BWR Power Conversion System

Internal walls of the turbine generator building include reinforced concrete walls, concrete block walls, and metal partitions. The turbine, the main condenser, the moisture separator reheaters, the feedwater heater system, and the condenser gas removal and handling equipment are surrounded by concrete walls at least 1.07 m thick for shielding purposes. The walls that surround the turbine generator area at elevation 152.7 m extend up to elevation 159.9 m.

A 182-Mg-capacity overhead crane services the building.

There are two steel tanks for condensate storage located adjacent to the turbine generator building, as shown in Figure K.1-1. The condensate storage tanks are in an enclosed, diked retaining area consisting of a structural slab on soil and four perimetral walls.

K.1.1.3 Radwaste and Control Building

The radwaste and control building houses the main control room, cable spreading room, emergency power equipment, radioactive liquid and solid waste treatment systems, condensate demineralizer system, reactor water cleanup demineralizer system, fuel pool cooling and cleanup demineralizer system, and offices and facilities for shift operating personnel.

The building is constructed of reinforced concrete and metal-sided and roofed structural steel, with two full floors and one partial floor above the ground floor. It is approximately 63.7 m by 48.8 m in plan and 32 m in overall height. Additional details of construction and diagrams showing the placement of equipment in the building are given in Appendix C of Reference 1.

K.1.1.4 Other Structures

Other structures on the reference BWR site include:

- a cooling tower complex with six cooling towers, a circulating water pumphouse, and two electrical buildings
- the diesel generator building
- the service building
- the makeup water pumphouse
- the gas bottle storage building
- two spray ponds that provide the ultimate heat sink for the facility.

These structures are expected to remain uncontaminated under both normal and accident conditions.

K.1.2 Primary Containment Vessel

The primary containment vessel (see Figure K.1-2) contains a drywell that houses the reactor vessel, a sacrificial shield, the reactor coolant recirculating loops, and connections of the reactor primary system. It also contains a pressure suppression chamber that stores 3160 m^3 (maximum) of water and a submerged vent system that connects the drywell and the suppression pool.

The primary containment vessel is a free-standing steel pressure vessel with a dividing floor forming an over-under configuration. Vessel shell plate thicknesses vary from 38.1 mm at the bottom to 19.1 mm at the conical section. The vessel is enclosed in the biological shield wall, but it is separated from the wall by an annulus of compressible insulation material approximately 50 mm thick.

Physical dimensions of the containment vessel are:

- diameter of cylindrical portion at base of cone ∿26.2 m
- diameter at top of cone ∿11.4 m
- overall shell height ∿52 m.

The vessel is reinforced with internal vertical and horizontal stiffeners. The top closure head of the drywell is bolted to a steel flange attached to the top of the containment vessel. The vessel is provided with two concentric circular skirts on the bottom ellipsoidal head intergral with the vessel. The skirts are anchor-bolted to the foundation mat and are backed up by concrete fill.

Major structural components inside the primary containment vessel are shown in Figure K.1-2 and include the following:

<u>Reactor Pedestal</u>

The reactor pedestal is a hollow, right-cylindrical reinforced concrete foundation that supports the reactor pressure vessel and the sacrificial shield wall. The bottom of the pedestal is keyed into the reinforced concrete liner inside the bottom head of the primary containment vessel. The inside and outside surfaces of the reactor pedestal are coated with a special decontaminable epoxy coating.

<u>Sacrificial Shield Wall</u>

The sacrificial shield wall is a vertical, cylindrical shell structure that surrounds the lower two-thirds of the reactor pressure vessel. It is constructed of layers of inner and outer steel rings welded together and filled with concrete.

Drywell Floor

The drywell floor is a leak-tight pressure barrier dividing the primary containment vessel into a drywell portion above the floor and a suppression chamber (wetwell) below the floor. Eighty-four 0.61-m-diameter and eighteen 0.71-m-diameter downcomer and vent pipes penetrate the drywell floor into the suppression chamber. The drywell floor is coated with a special decontaminable epoxy coating.

Stabilizer Truss

The stabilizer truss is a circular steel truss that connects the top of the sacrificial shield wall to the containment vessel.

K.1.2.1 Reactor Vessel and Internals

The reactor vessel, shown in Figure K.1-5, is a vertical, cylindrical pressure vessel of welded steel construction designed for a pressure of 8.72 MPa. The vessel is fabricated of carbon steel and is clad internally with stainless steel (except for the top head, nozzles, and nozzle weld zones which are unclad). The vessel top head is secured to the reactor vessel by 108 studs and nuts. Vessel flanges are sealed with two concentric metal seal-rings.

The design parameters of the reactor vessel are presented in Table K.1-1. 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.

The major reactor internal components are the core (fuel, flow channels, control rods, and instrumentation), the core support structure (including the core shroud, top fuel guide, and core support plate), the shroud head and steam separator assembly, the steam dryer assembly, the jet pumps, the feedwater spargers, and the core spray lines. The internals are made primarily of stainless steel and have a total mass of almost 200 Mg.

Two semicircular groups of 10 jet pumps are located in the outer annulus between the core shroud and the reactor vessel wall. These jet pumps are part of the reactor water recirculation system described in Section K.1.2.2.

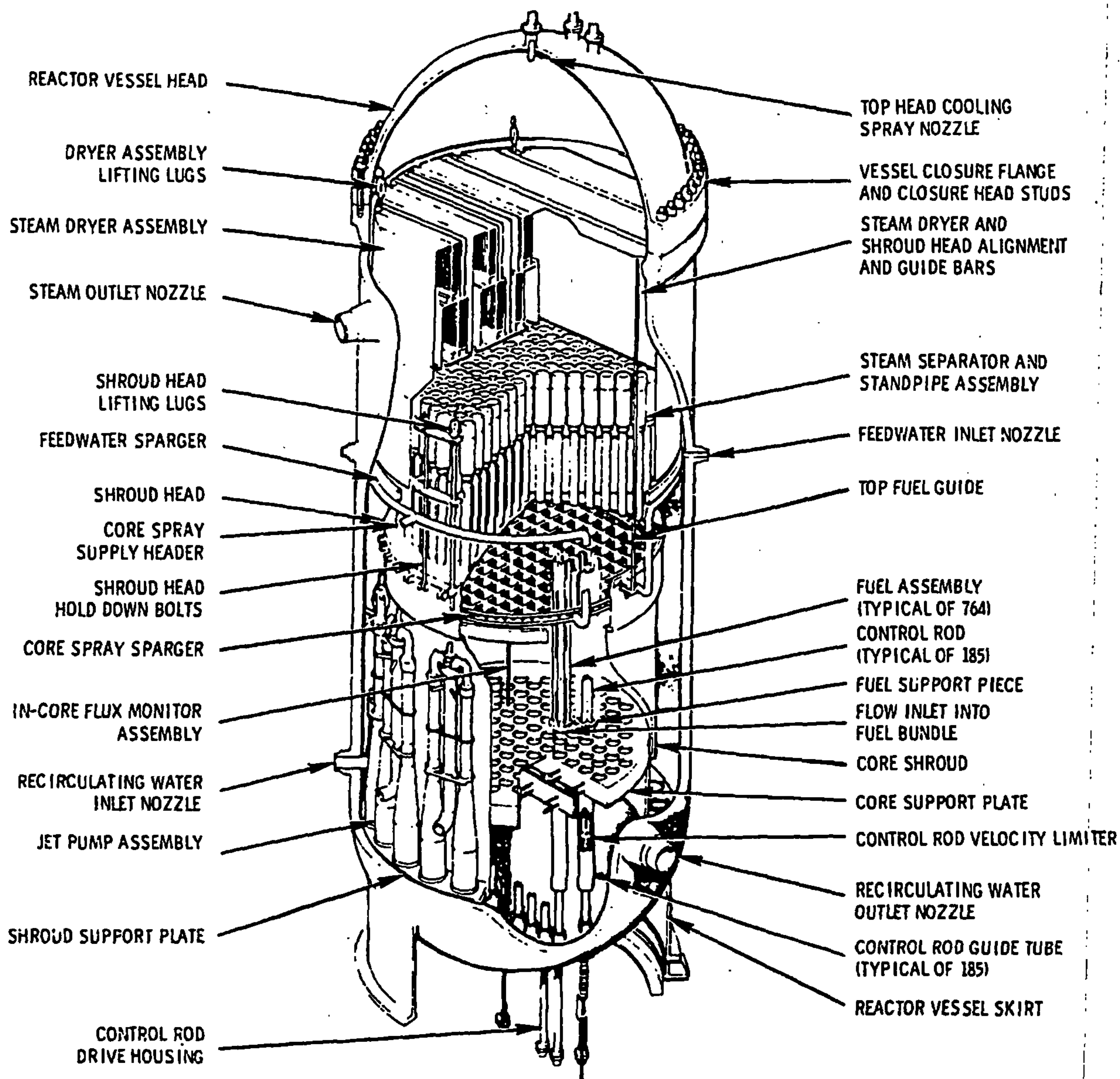


FIGURE K.1-5. BWR Reactor Vessel and Internals

K.1.2.2 Reactor Water Recirculation System

The reactor water recirculation system, shown in Figure K.1-6, has two identical loops external to the reactor vessel, but located in the drywell inside the primary containment vessel. Each loop contains one motor-driven recirculation pump, one hydraulically operated flow control valve, two motor-operated shutoff valves, and bypass around the discharge shutoff valve

TABLE K.1-1. BWR Reactor Vessel Design Parameters

Parameter of Interest	Design Specification
Internal Height from Inside Bottom Head to Inside Top Head	22.23 m
Shell Height to Top of Closure Flange	18.92 m
ID of Shell	6.375 m
Minimum Shell Wall Thickness	171 mm
Wall SS Cladding Thickness	3.18 mm
Bottom Head Thickness	203 mm
Top Head Thickness	114 mm
Number of Top Head Studs	108
Mass of Vessel with Top Head, Studs, Nuts, and Washers	748.5 Mg
Mass of Top Head	90.7 Mg
Mass of Vessel with Top Head and Internals	1 034.2 Mg

and the flow control valve. Each loop supplies reactor water to 10 jet pumps located inside the reactor vessel in the annular region between the core shroud and the reactor vessel wall.

Each reactor water recirculation pump is a single-suction, single-stage, double volute, motor-driven pump. The pump drive motor is mounted on a carbon steel stand above the pump, and torque is transmitted to the pump impeller by a vertical shaft. Besides being held in place by its placement in the recirculation loop piping, each recirculation pump motor is suspended from an overhead radial beam system by four spring-loaded hangers. Each pump case is semi-rigidly attached to the surrounding drywell structure by three mechanical snubbers.

K.1.3 Spent Fuel Handling and Storage

Defueling of the reactor is a major cleanup activity following the reference accidents analyzed in this study. Insofar as practical, post-accident defueling operations are performed by using the same equipment and procedures that are used for normal reactor refueling operations. The equipment and procedures used for normal reactor refueling at the reference BWR are described in this subsection. All spent fuel handling and storage activities take place in the reactor building.

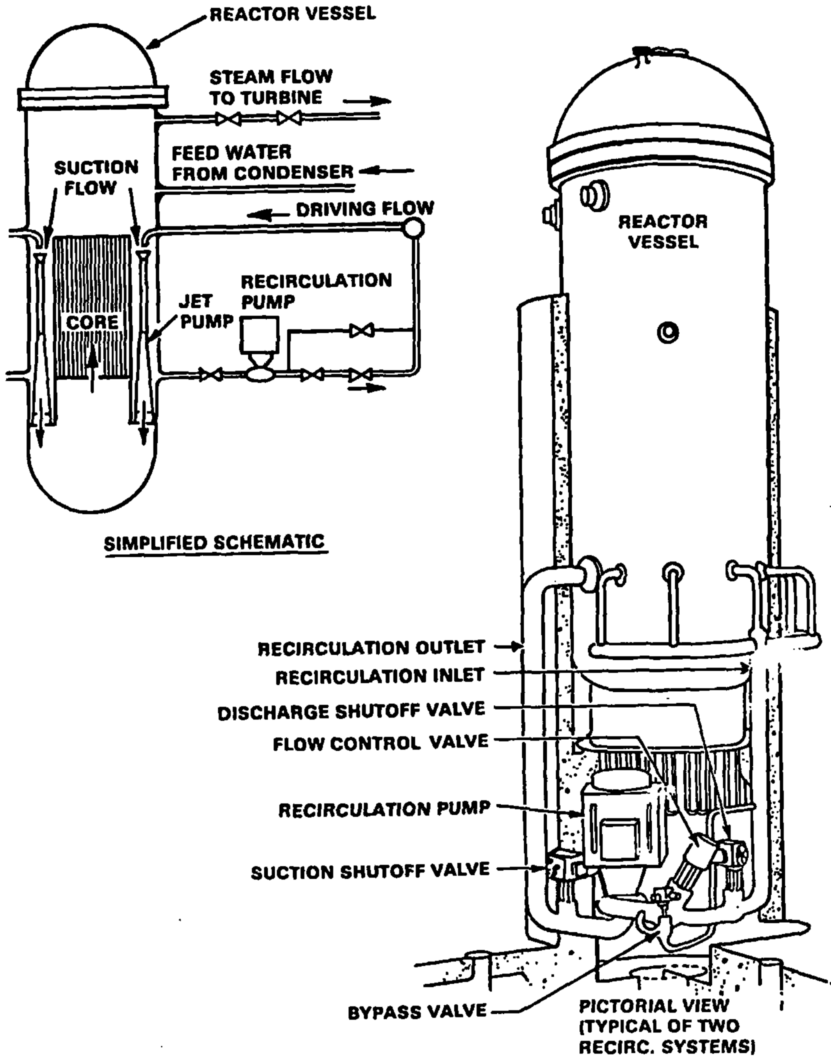


FIGURE K.1-6. BWR Reactor Water Recirculation System

The spent fuel storage pool is located in the reactor building, as shown previously in Figure K.1-2. It is a reinforced concrete structure completely lined with 6.4-mm stainless steel plates that are seamwelded together and are anchored to the surrounding concrete by reinforcing members welded to the liner plates. The pool contains fuel storage racks with enough storage locations for 2658 fuel assemblies (a normal core loading is 764 fuel assemblies). The pool is filled with water to a depth of approximately 11.5 m (water volume: 1325 m^3). The water supply is the condensate supply system, with emergency makeup water provided by the standby service water system.

K.1.3.1 Defueling Equipment

Two major equipment items used for normal spent fuel handling and storage operations at the reference BWR are the reactor building crane and the refueling platform.

Reactor Building Crane. The reactor building crane is a single-trolley top-running electric overhead traveling crane with a 114-Mg-capacity main hoist and a span of about 38.4 m. The crane can be operated either from the cab or from the refueling floor by radio control. The main purpose of the crane is to handle spent fuel casks; secondary purposes include servicing and refueling the reactor vessel and handling equipment and parts.

Refueling Platform. The refueling platform is a gantry crane used to transport fuel and reactor components to and from pool storage and the reactor vessel. The platform spans the fuel storage and vessel pools on rails bedded in the refueling floor. A telescoping mast and grapple suspended from a trolley system is used to transport and orient fuel bundles for placement. Control of the platform is from an operator station on the main trolley. Two 450-kg-capacity auxiliary hoists, one on the main trolley and the other on the auxiliary trolley, are provided for such activities as fuel support replacement, jet pump servicing, and control rod replacement.

K.1.3.2 Defueling Procedures

Defueling procedures are divided into three major phases: preparation, reactor vessel opening, and fuel handling. A general description of each of these phases is presented in the following paragraphs.

Preparation. Prior to plant shutdown, all equipment is thoroughly checked and tested. All necessary maintenance and interlock checks are performed to ensure against equipment failure. The reactor is then shut down according to a prescribed procedure. During cooldown, the reactor vessel is vented and filled to above flange level to equalize cooling. The reactor well shield plugs are removed using the reactor building cranes. The canal plugs and slot plugs are then removed to prepare for connecting the reactor well pool cavity with the dryer and separator pool and the spent fuel pool.

Reactor Vessel Opening. The reactor vessel head nuts are loosened using a stud tensioner supported from the reactor building crane and then removed using the vessel nut handling tool. Vessel stud protectors and vessel head guide caps are installed. The head strongback, transported by the reactor building crane, is attached to the vessel head, and the head is transported to the head-holding pedestals on the refueling floor. The dryer-separator sling is lowered by the reactor building crane and attached to the dryer lifting lugs. The dryer is then lifted from the reactor vessel and transported to its storage location in the dryer-separator storage pool adjacent to the reactor well. In preparation for separator removal, the service platform and service platform support are installed on the vessel flange. The steam line plugs are installed and the separator unbolted. The service platform is then removed, 6
and the dryer-separator unbolted. The service platform is then removed, and the dryer-separator sling is lowered into the vessel and attached to the separator lifting lugs. The water in the reactor well and in the dryer-separator pool is raised to fuel pool water level, and the separator is transferred under water to its allotted storage space. The remaining gate isolating the fuel pool from the reactor well is now removed, and fuel handling can commence.

Fuel Handling. Detailed procedures for defueling and fuel movement are specifically developed immediately prior to the defueling. The actual fuel handling is done with the fuel grapple, which is an integral part of the refueling platform. In addition to the fuel grapple, the refueling platform is equipped with two auxiliary hoists which can be used with various grapples to service other reactor internals. To move fuel, the fuel grapple is aligned over the fuel assembly, lowered, and attached to the fuel bundle bail. The fuel bundle is raised out of the core, moved through the refueling slot to the fuel pool, positioned over the storage rack, and lowered to storage. Fuel is shuffled within the storage pool in the same manner.

K.2 DETAILS OF REFERENCE BWR ACCIDENT SCENARIOS AND RESULTANT CONTAMINATION LEVELS

Values postulated for the parameters that characterize the reference BWR accident scenarios are given in Table K.2-1, including values for fission product contamination and for radiation exposure rates in both the primary containment vessel and the reactor building. Values listed in the table refer to conditions 1 year after the postulated accidents. Assumptions and models used to estimate radioactive contamination and exposure levels for the reference BWR accident scenarios are discussed in this section.

K.2.1 Fission Product Contamination

The BWR accident scenarios postulated for this study are basically similar to the PWR accident scenarios discussed in Chapter 8 and Appendix C. For both the BWR and the PWR accidents, the postulated accident consequences are believed to be credible in terms of initiating events. The three postulated accidents provide a spectrum of accident consequences that are the bases for cleanup and decommissioning analyses with different cost and safety requirements. Use of the same basic accident scenarios for both the BWR and the PWR facilitates comparisons between cost and safety requirements for cleanup and decommissioning for the two reference reactors.

As discussed in Section 16.3 of Volume 1, the same fission product source inventory and the same fuel cladding failure and fuel melting fractions are used for both the BWR and the PWR accident scenarios. Differences in contamination levels and exposure rates inside the reactor buildings of the BWR and PWR result from differences in containment design for the two reactor power plants. For the reference PWR, the reactor building is the primary containment structure. (Details of the reference PWR are given in Appendix B.) For the reference BWR, which uses the Mark II containment design, the reactor building provides a secondary level of containment. The primary containment is a steel pressure vessel situated inside a concrete biological shield within the reactor building (see Figure K.1-2). The primary containment encloses the reactor vessel, the reactor coolant recirculating loops, the drywell, and the pressure suppression pool. The most likely BWR

TABLE K.2-1. 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

reactor water recirculation system volume) of water to the reactor building in the form of contaminated steam. The radioactivity concentration in the steam released to the reactor building is the same as the radioactivity concentration in the steam released to the drywell.

Contamination of other onsite structures as a result of a reactor accident is generally not considered in this study. However, for the scenario 3 accident, some radioactive contamination of the radwaste building is postulated. This contamination is assumed to be limited to plateout on building and equipment surfaces and to internal contamination of the reactor water cleanup system. General area radiation exposure levels inside the radwaste building following the scenario 3 accident are assumed to be about 50 mR/hr. Higher readings of up to 10 R/hr occur in cubicles that contain the filters, demineralizers, and holdup tanks for the reactor water cleanup system.

K.2.2 Exposure Rate Calculations

Estimated average gamma radiation exposure rates from fission product contamination at the operating floor level inside the containment vessel (i.e., the drywell) and at the refueling, operating, and service floor levels inside the reactor building of the reference BWR 1 year after the postulated accidents are shown in Table K.2-1. Exposure rates are based on fission product contamination levels described in Section K.2.1.

Estimated average external exposure rates at the operating floor level inside the containment vessel range from 58 mR/hr for the scenario 1 accident to 5.1 R/hr for the scenario 3 accident. About 90% of the estimated gamma exposure rate inside the containment vessel is from surface contamination (plateout) with the remainder from suppression pool contamination.

Estimated average external exposure rates inside the reactor building following the scenario 3 accident range from 20 mR/hr at the refueling floor level to 30 R/hr at the service floor level. At the refueling and operating floor levels the estimated gamma exposure rate is almost entirely the result of surface contamination. At the service floor level the estimated gamma exposure rate is almost entirely from sump water contamination.

Exposure rate calculations employ the methodology for calculating photon fluxes from uniformly contaminated regular geometric sources described in the Reactor Shielding Design Manual.[4] Details of the methodology are given in Section C.3 of Appendix C. To calculate the external exposure rate from surface contamination, the postulated contamination on floors and walls is approximated by a uniformly contaminated infinite plane. To calculate the external exposure rate from the contaminated suppression pool or sump water, the contaminated liquid is modeled as a uniformly contaminated disk of finite thickness. Concrete shielding in the reactor building is assumed to reduce the exposure rate from the contaminated suppression pool or sump water. A simplified model of the BWR containment vessel and reactor building, used to define the geometrical parameters in the equations for exposure rate calculations, is shown in Figure K.2-1. Point P_1 represents the point at the operating floor level inside the containment vessel where gamma exposure rates are calculated. Points P_2, P_3, and P_4 represent points at the refueling, operating, and service floors, respectively, where gamma exposure rates in the reactor building are calculated.

The estimated external exposure rates from fission product contamination shown in Table K.2-1 are average values that provide a basis for estimating occupational radiation doses to workers engaged in cleanup and decommissioning operations inside the reference BWR. Actual exposure rates would vary from the average, depending on the location of the worker in the building. However, the average rates shown in the table are believed to be adequate for the estimates of occupational safety given in this study.

K.3 DETAILS OF ACTIVITIES AND MANPOWER REQUIREMENTS FOR ACCIDENT CLEANUP

This section provides details of the technical requirements and manpower needs for accident cleanup in the reference BWR following the postulated accidents described in Section K.2.

Details of the rationale for accident cleanup are presented in Section E.1 of Appendix E. The goals of BWR accident cleanup are the same as those of PWR accident cleanup discussed in Appendix E, namely:

TABLE K.2-1. (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.

accident scenarios would result in the released radioactivity being confined within the primary containment.

The assumptions used to estimate fission product contamination in the primary containment of the reference BWR for the postulated accidents are discussed 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 reactor coolant recirculation system. The remaining 50% is released to the drywell atmosphere.

2. Approximately 0.1% of the non-gaseous fission product radioactivity released to the drywell atmosphere plates out on surfaces inside the drywell. The remainder of the radioactivity concentrates in the suppression pool.

3. Radioactivity that plates out on building and equipment surfaces within the drywell is initially distributed uniformly over a total surface area of 5000 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.

These assumptions are similar to those made for the PWR accident scenarios. The rationale for these assumptions is presented in Section 8.3.1 of Volume 1.

BWR loss-of-coolant accidents would likely result in the retention of all or most of the released radioactivity within the drywell and in the suppression pool. To provide a basis for evaluating post-accident cleanup requirements in the BWR reactor building, should a severe accident result in contamination of this building, some leakage of radioactivity through penetrations in the primary containment is postulated for the scenario 2 and scenario 3 accidents. (There are 171 penetrations through the primary containment, ranging in diameter from 19.1 mm to 3.81 m.[1]) For the scenario 2 accident, leakage is postulated to result in the release of 10 Ci of radioactivity to the reactor building. For the scenario 3 accident, leakage is postulated to result in a release of 500 m^3 (approximately one

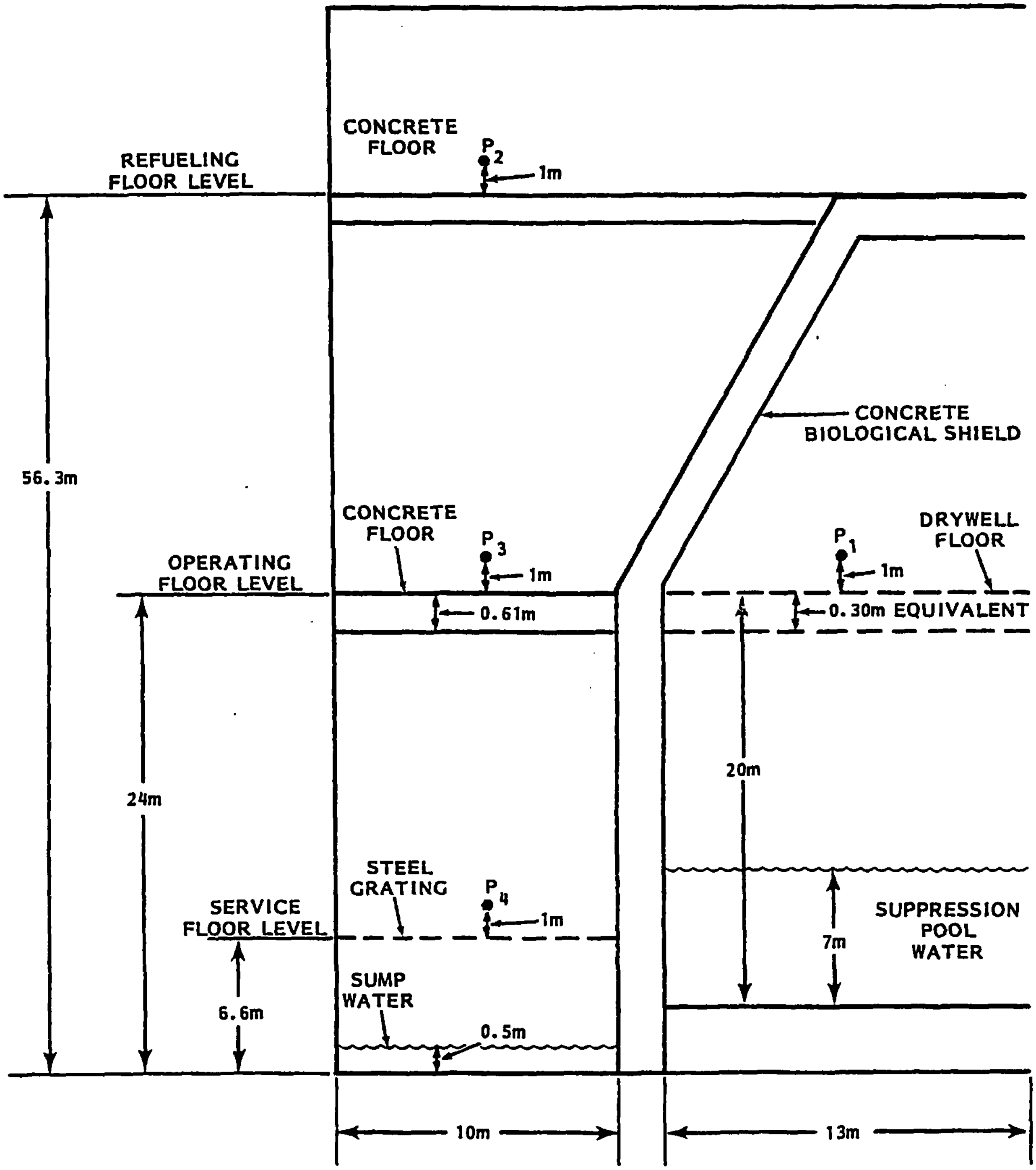


FIGURE K.2-1. Simplified Model of BWR Containment

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 BWR accident cleanup campaign is postulated to include tasks that are similar to those described in Appendix E for PWR accident cleanup, namely:

- 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 recirculation system
- solidification and packaging of wastes from accident cleanup operations.

The contaminated water generated by the accident includes the suppression pool water (which contains most of the radioactivity released in the accident) and water that collects in sumps, on floors, or in the basement of the reactor building.

Building surfaces include those in the reactor building (contamination of the reactor building is postulated for both the scenario 2 and the scenario 3 accidents) and those inside the containment vessel. Decontamination of the containment vessel is not a requirement for defueling the reactor since defueling operations are carried out from the defueling floor of the reactor building and do not require entry into the containment. However, in this study, decontamination of the containment vessel is considered part of accident cleanup for the following reasons:

1) Contamination of the containment vessel is a direct result of the accident, and decontamination costs may properly be charged to accident cleanup.

2) The first steps in accident cleanup of the containment vessel involve the washdown of containment surfaces with water from the containment spray system and with high-pressure hoses. The water from these cleanup operations drains into the suppression pool and is processed with the contaminated suppression pool water.

3) Containment cleanup is required to provide access to this area for workers engaged in either reactor decommissioning or reactor refurbishment and restart operations.

K.3.1 Details of Preparations for Accident Cleanup

A period of planning and preparation precedes the actual performance of accident cleanup operations in the accident-damaged BWR. Planning and preparation activities are similar to those for the PWR described in Section E.2.1 of Appendix E. They include:

- reactor building and containment entry and data acquisition
- venting of radioactive gases (e.g., krypton-85)
- 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.

As was postulated for PWR accident cleanup, the filter/demineralizer system used for processing accident water is assumed to be installed in the spent fuel pool prior to containment vessel cleanup. This requires that fuel stored in the pool from refuelings during normal operation be removed to make space available for the filter/demineralizer. Other provisions could be made for the filter/demineralizer system, such as locating it in a building specially constructed on the site. The requirements and costs for this option as it relates to PWR decommissioning are discussed in Chapter 15. Similar considerations would apply for the BWR if a separate building for the filter/demineralizer system were constructed prior to accident cleanup.

The 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, except that for the BWR more time is required to remove the accumulated spent fuel and fuel storage racks from the spent fuel storage pool. A minimum of 15 months is assumed to be required to discharge the accumulated spent fuel and ship it to an ISFSI (independent spent fuel storage installation), based on the assumptions that the pool contains 1-1/3 fuel cores at the time of the reactor accident and that two spent fuel rail casks are continuously available to transport 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 staff organization for preparations for BWR accident cleanup is the same as that for preparations for PWR accident cleanup shown in Figure E.2-2 of Appendix E. Estimated staff labor requirements for utility staff engaged in prepartions for BWR accident cleanup are shown in Table K.3-1. A total of 369 man-years of staff labor are required for cleanup preparations following the scenario 1 accident, 552 man-years following the scenario 2 accident, and 972 man-years following the scenario 3 accident. These staff labor requirements form the bases for the staff labor cost estimates for preparations for accident cleanup discussed in Section K.4.1.

Estimated occupational doses to workers who enter the reactor building during preparations for accident cleanup are shown in Table K.3-2. Total estimated occupational doses for this activity are 57 man-rem following the scenario 1 accident, 146 man-rem following the scenario 2 accident, and 426 man-rem following the scenario 3 accident. Operations that are major contributors to occupational dose during preparations for accident cleanup include radiation surveys inside the reactor building, the discharge of spent fuel from the spent fuel pool, and radiation surveys inside the containment

TABLE K.3-1. Estimated Utility Staff Labor Requirements for Preparations for Accident Cleanup at the Reference BWR

Position	Scenario 1 Accident man-years/year	Scenario 1 Accident Total man-years(a)	Scenario 2 Accident man-years/year	Scenario 2 Accident Total man-years(b)	Scenario 3 Accident man-years/year	Scenario 3 Accident Total man-years(c)
Plant Superintendent	1.0	1.5	1.0	2.0	1.0	3.0
Assistant Plant Superintendent	1.0	1.5	1.0	2.0	1.0	3.0
Consultants	4.0	6.0	6.0	12.0	10.0	30.0
Secretaries and Word Processors	8.0	12.0	10.0	20.0	12.0	36.0
Site Support Staff						
Health and Safety Supervisor	1.0	1.5	1.0	2.0	1.0	3.0
Health Physicist	1.0	1.5	1.0	2.0	1.0	3.0
Senior Health Physics Technician	8.0	12.0	8.0	16.0	12.0	36.0
Health Physics Technician	16.0	24.0	16.0	32.0	24.0	72.0
Protective Equipment Attendant	4.0	6.0	8.0	16.0	8.0	24.0
Industrial Safety Specialist	1.0	1.5	1.0	2.0	1.0	3.0
Industrial Safety Technician	2.0	3.0	2.0	4.0	2.0	6.0
Security Supervisor	1.0	1.5	1.0	2.0	1.0	3.0
Security Shift Supervisor	4.0	6.0	4.0	8.0	4.0	12.0
Security Patrolman	48.0	72.0	48.0	96.0	48.0	144.0
Contracts and Accounting Supervisor	1.0	1.5	1.0	2.0	1.0	3.0
Accountant	1.0	1.5	1.0	2.0	2.0	6.0
Contracts Specialist	1.0	1.5	1.0	2.0	1.0	3.0
Insurance Specialist	1.0	1.5	1.0	2.0	1.0	3.0
Procurement Specialist	1.0	1.5	1.0	2.0	1.0	3.0
Clerk	2.0	3.0	4.0	8.0	4.0	12.0
Quality Assurance Supervisor	1.0	1.5	1.0	2.0	1.0	3.0
Quality Assurance Engineer	1.0	1.5	2.0	4.0	2.0	6.0
Quality Assurance Technician	1.0	1.5	2.0	4.0	2.0	6.0
Subtotals		144.0		208.0		351.0
Plant Operations Staff						
Plant Operations Supervisor	1.0	1.5	1.0	2.0	1.0	3.0
Plant Chemist	1.0	1.5	1.0	2.0	1.0	3.0
Chemist	2.0	3.0	2.0	4.0	2.0	6.0
Reactor Operations Engineer	1.0	1.5	1.0	2.0	1.0	3.0
Engineer	2.0	3.0	2.0	4.0	2.0	6.0
Reactor Operations Shift Supervisor	4.0	6.0	4.0	8.0	4.0	12.0
Senior Reactor Operator	8.0	12.0	8.0	16.0	8.0	24.0
Reactor Operator	16.0	24.0	16.0	32.0	16.0	48.0
Utility Operator	16.0	24.0	16.0	32.0	16.0	48.0
Technician	12.0	18.0	12.0	24.0	12.0	36.0
Craft Supervisor	1.0	1.5	1.0	2.0	1.0	3.0
Crew Foreman	4.0	6.0	4.0	8.0	4.0	12.0
Maintenance Mechanic	16.0	24.0	16.0	32.0	16.0	48.0
Instrument Technician	16.0	24.0	16.0	32.0	16.0	48.0
Warehouseman	4.0	6.0	4.0	8.0	4.0	12.0
Tool Crib Attendant	4.0	6.0	4.0	8.0	4.0	12.0
Subtotals		162.0		216.0		324.0
Cleanup Planning Staff						
Cleanup Planning Supervisor	1.0	1.5	1.0	2.0	1.0	3.0
Engineering Supervisor	1.0	1.5	1.0	2.0	2.0	6.0
Engineer	6.0	9.0	8.0	16.0	12.0	36.0
Estimator	1.0	1.5	2.0	4.0	4.0	12.0
Draftsman	2.0	3.0	4.0	8.0	6.0	18.0
Crew Leader	1.0	1.5	2.0	4.0	2.0	6.0
Utility Operator	4.0	6.0	8.0	16.0	16.0	48.0
Craftsman	8.0	12.0	12.0	24.0	16.0	48.0
Laborer	4.0	6.0	8.0	16.0	16.0	48.0
Subtotals		42.0		92.0		225.0
Totals		369		552		972

(a) Based on an estimated preparations for cleanup time requirement of 1.5 years.
(b) Based on an estimated preparations for cleanup time requirement of 2.0 years.
(c) Based on an estimated preparations for cleanup time requirement of 3.0 years.

TABLE K.3-2. Estimated Occupational Radiation Doses to Workers Entering the Reactor Building During Preparations for Accident Cleanup at the Reference BWR

	Accident Scenario		
	Scenario No. 1	Scenario No. 2	Scenario No. 3
Discharge Accumulated Spent Fuel			
Number of Entries into Reactor Building	90	90	90
Average Time per Entry (hours)	6	6	6
Average Dose Rate (rem/hr)	0.010	0.015	0.030
Number of Workers per Entry	8	8	8
Total Accumulated Occupational Dose (man-rem)	43	65	130
Radiation Survey of Reactor Building			
Number of Entries into Reactor Building	20	20	30
Average Time per Entry (hours)	4	4	4
Average Dose Rate (rem/hr)	0.01	0.02	0.1
Number of Workers per Entry	8	8	8
Total Accumulated Occupational Dose (man-rem)	6	13	96
Radiation Survey of Containment Vessel			
Number of Entries into Containment Vessel	10	10	10
Average Time per Entry (hours)	1	1	0.5
Average Dose Rate (rem/hr)	0.1	0.85	5
Number of Workers per Entry	8	8	8
Total Accumulated Occupational Dose (man-rem)	8	68	200
Total Occupational Dose During Preparations for Cleanup	57	146	426

vessel. The containment is assumed to be vented for the removal of ^{85}Kr prior to the initial entry of workers into the vessel to perform radiation surveys. All personnel entering the containment vessel wear protective clothing and full-face respirators. Respirators are assumed not to be required for entry into the reactor building except for performing radiation surveys in the lowest levels of the building following the scenario 3 accident.

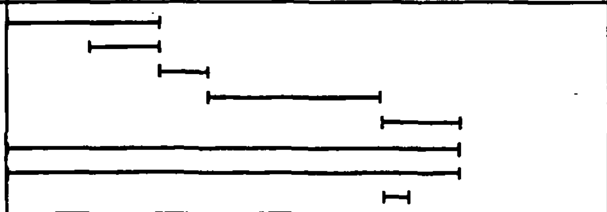
K.3.2 Details of Accident Cleanup in the Radwaste Building

Fission product contamination of the radwaste building is postulated for the scenario 3 accident. This contamination includes radioactive plateout on building and equipment surfaces and internal contamination of the reactor water cleanup system. The radwaste building contains many areas associated with daily plant operation as well as radioactive liquid and solid waste systems, the reactor water cleanup demineralizer system, the spent fuel pool cooling and cleanup demineralizer system, the fire protection system, and other safety-related and nonsafety-related systems. Reliable operation of the systems and equipment in this building is essential to maintaining the reactor in a safe shutdown condition until defueling and to the efficient performance of cleanup operations in the reactor building and the containment. Decontamination of the building is necessary to permit routine access by plant personnel to perform required operational and maintenance tasks without excessive occupational exposures or the need for elaborate protective clothing.

Accident cleanup in the radwaste building following the scenario 3 accident is postulated to take place during preparations for accident cleanup in the reactor building and the containment. The task sequence and schedule for radwaste building decontamination is shown in Figure K.3-1. Tasks in this cleanup operation 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 for processing contaminated liquids from the reactor water cleanup system

CLEANUP TASK (SHIFTS PER DAY[a]/DURATION IN MONTHS)		TIME (MONTHS) AFTER START OF RADWASTE BUILDING CLEANUP OPERATIONS 1–24	MAN-DAYS PER SHIFT[b] CREW LEADER	UTILITY OPERATOR	LABORER	CRAFTSMAN	HEALTH PHYS. TECHNICIAN
INITIAL DECONTAMINATION OF RADWASTE BUILDING	(2[c]/6)		2	4	4		2
INSTALL LIQUID WASTE PROCESSING SYSTEM	(2/3)		1	2	2	4	1
FLUSH RADWASTE SYSTEMS AND PROCESS CONTAMINATED LIQUIDS	(3/2)		1	2	2		1
ADDITIONAL DECONTAMINATION OF RADWASTE BUILDING	(2/7)		2	4	4		2
INITIAL DECONTAMINATION OF REACTOR BUILDING[d]	(2/3)		1	2	2	2	1
PROCESS AND PACKAGE WASTE FROM CLEANUP OPERATIONS	(2/18)		1	2	2		1
CONSTRUCTION AND MAINTENANCE SUPPORT	(2/18)		1			6	1
SURVEY RADWASTE BUILDING	(2/1)		1		2		4



LABOR CATEGORY	TOTAL MAN-MONTHS	MAN-MONTHS PER WORKING MONTH[e] 1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24
CREW LEADER	140	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	6	6						
UTILITY OPERATOR	204	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	8	8	8						
LABORER	208	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	8	8						
CRAFTSMAN	252	12	12	12	20	20	20	12	12	12	12	12	12	12	12	12	16	16	16						
HEALTH PHYSICS TECHNICIAN	146	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	14	6	6						

(a) TWO SHIFTS PER DAY OPERATE ON 5-DAY WEEK. THREE SHIFTS PER DAY OPERATE ON 7-DAY WEEK.

(b) ASSUME 20 MAN-DAYS PER MAN-MONTH.

(c) ONE SHIFT PER DAY DURING MONTHS 4 THROUGH 6.

(d) DECONTAMINATION OF SURFACES 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.

(e) MANPOWER REQUIRMENTS ARE SHOWN ROUNDED TO THE NEAREST MAN-MONTH. REQUIREMENTS ARE BASED ON NECESSARY LABOR TO COMPLETE TASKS AND DO NOT INCLUDE EXTRA MANPOWER NEEDED TO COMPLY WITH OCCUPATIONAL DOSE LIMITS.

FIGURE K.3-1. Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident

- 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 K.3-1 is some initial decontamination of the reactor building, primarily at the refueling floor level, to allow worker access for discharging the accumulated fuel from the spent fuel pool and the subsequent installation in the pool of specially fabricated racks for canistered fuel and of the filter/demineralizer system postulated to be installed there. The filter/demineralizer system is similar to that described in Section E.4.1 and is used to process suppression pool water and other contaminated liquids generated by the accident and by cleanup operations in the reactor building and the containment. (Time and manpower requirements for the installation of this equipment in the spent fuel pool are given in Figure K.3-4 for the scenario 3 accident.)

Accident cleanup in the radwaste building following the scenario 3 accident is postulated to require approximately 1.5 years. Cleanup is accomplished by crews that perform decontamination, construction and maintenance support, and waste processing and waste packaging functions. This cleanup staff is added to the staff for preparations for cleanup shown in Figure E.2-2 of Appendix E. The numbers of personnel required to actually complete the cleanup tasks in the radwaste building are shown in Figure K.3-1.

Estimated occupational radiation doses to cleanup workers during accident cleanup in the radwaste building are shown in Table K.3-3. The radiation doses shown in the table are external doses from gamma radiation. Workers are assumed to use respiratory devices as necessary to protect against the inhalation of radioactive particulates. To maintain individual worker doses within occupational dose limits (assumed to be 5 rem/year per worker based on 10 CFR 20.101), the staff labor requirements shown in Figure K.3-1 must be increased. Cleanup worker requirements adjusted to comply with occupational radiation dose limits are shown in Table K.3-4. The adjusted manpower

TABLE K.3-3. Estimated Occupational Radiation Doses to Workers During Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident

Cleanup Task	Average Dose Rate (rem/hr)	Crew Leader		Utility Operator		Laborer		Craftsman		Health Physics Technician		Task Totals	
		Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)
Initial Decontamination of Radwaste Building	0.020	1 440	28.8	2 880	57.6	2 880	57.6	--	--	1 440	28.8	8 640	172.8
Install Liquid Waste Processing System	0.010	480	4.8	960	9.6	960	9.6	1 920	19.2	480	4.8	4 800	48.0
Flush Radwaste Systems and Process Contaminated Liquids	0.010	1 080	10.8	2 160	21.6	2 160	21.6	--	--	1 080	10.8	6 480	64.8
Additional Decontamination of Radwaste Building	0.010	2 240	22.4	4 480	44.8	4 480	44.8	--	--	2 240	22.4	13 440	134.4
Initial Decontamination of Reactor Building	0.035	480	16.8	960	33.6	960	33.6	960	33.6	480	16.8	3 840	134.4
Process and Package Waste from Cleanup Operations	0.005	4 320	21.6	8 640	43.2	8 640	43.2	--	--	4 320	21.6	25 920	129.6
Construction and Maintenance Support	0.010	2 880	28.8	--	--	--	--	17 280	172.8	2 880	28.8	23 040	230.4
Survey Radwaste Building	0.005	240	1.2	--	--	--	--	--	--	960	4.8	1 680	8.4
Totals		13 160	135.2	20 080	210.4	20 560	212.8	20 160	225.6	13 880	138.8	87 840	922.8

TABLE K.3-4. Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident

Worker Category	Estimated Worker Requirements[a] (man-yr)	Estimated Occupational Dose: Total[b] (man-rem)	Estimated Occupational Dose: Individual Average (man-rem/man-yr)	Adjustment Factor[c]	Adjusted Worker Requirements (man-yr)
Crew Leader	11.7	135.2	11.6	2.4	28.1
Utility Operator	17.0	210.4	12.4	2.5	42.5
Laborer	17.4	212.8	12.3	2.5	43.5
Craftsman	21.0	225.6	10.8	2.2	46.2
Health Physics Technician	12.2	138.8	11.4	2.3	28.1
Totals	79.3	922.8			188.4

(a) Based on Figure K.3-1.
(b) Based on Table K.3-3.
(c) Increase in worker requirements necessary to reduce average individual dose to ≤ 5 man-rem/man-year.

requirements shown in Table K.3-4 are used to compute staff labor costs for accident cleanup in the radwaste building (see Section K.4.2).

K.3.3 Details of Accident Cleanup in the Reactor Building and the Containment Vessel

Procedures and work schedules for accident cleanup in the reference BWR reactor building and containment vessel are presented in this section. Estimated occupational doses and estimated staff labor requirements based on these procedures and work schedules are also shown. The time and manpower requirements for accident cleanup given in this section are based on the three accident scenarios described in Section K.2.

K.3.3.1 Procedures for Accident Cleanup in the Reactor Building and the Containment Vessel

Accident cleanup in the reference BWR reactor building and containment vessel is postulated to include the following tasks:

- 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 E.4.1 of Appendix E.

The postulated filter/demineralizer system for the treatment of contaminated water is described in Section E.4.1.1 and shown schematically in Figure E.4-1. The system is assumed to be installed in the spent fuel pool in the reactor building. 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. For the scenario 2 and scenario 3 accidents, the contribution of suppression pool contamination to the average background level inside the containment vessel is so high that it is deemed advisable to process the suppression pool water through the filter/demineralizer system before the entry of personnel into the containment to begin surface decontamination operations. Some of the processed water is returned to the suppression pool to provide shielding from the contamination that exists on pool surfaces.

Water that has been processed in the filter/demineralizer system is stored in 1000-m^2-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 processed water not needed for reuse is discharged to the river under controlled conditions. The study recognizes that following an accident there could be restrictions against discharge of the processed water. However, it is assumed that processing would have reduced contamination levels in the water to values below the limits discussed in Section 5.3.1 of Chapter 5, and that, therefore, the processed water can eventually be discharged to the river. Other alternatives for water disposition discussed in Section 5.3.1 are not treated in detail in this study. However, a discussion of their relative costs is included in the discussion on the sensitivity of costs to various factors in Section 11.6 of Chapter 11.

Chemical decontamination solutions from accident cleanup operations are processed by evaporation. An evaporation/solidification system is rented from a commercial supplier and installed in the radwaste building during preparations for cleanup. 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.

Procedures for decontamination in the BWR reactor building and containment vessel are similar to those employed for decontamination in the PWR containment building and described in Section E.4.1.2 of Appendix E. The following sequence of operations is postulated:

1. Remove and package debris and small items of contaminated equipment that are easily disposed of.
2. Decontaminate reactor building surfaces and equipment using high-pressure hose wash and hands-on decontamination techniques.
3. Utilize the containment vessel spray system for remote wash of containment vessel surfaces.
4. Employ high-pressure hose wash techniques for semi-remote decontamination of containment vessel surfaces and equipment. The initial high-pressure hose wash of suppression pool surfaces can be accomplished by utilizing some of the existing penetrations through the drywell floor.
5. Perform hands-on decontamination of selected areas in the containment vessel where significant reductions in radiation exposure can be achieved with modest effort.
6. Decontaminate and refurbish or replace essential systems and services.
7. Provide local shielding of "hot spots."

Reactor defueling operations following the postulated BWR accidents are expected to be similar to PWR defueling operations described in Section E.4.3 of Appendix E. 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 related to work in radiation areas will be less for BWR defueling than for PWR defueling.

After reactor defueling is completed, the reactor pressure vessel head is reinstalled. The water is drained from the refueling cavity and the refueling cavity is decontaminated by flushing. The reactor water recirculation (RRC) system and portions of the reactor water cleanup (RWCU) system located in the reactor building are decontaminated to remove fission product plateout and fuel debris. Procedures for the decontamination of these systems are similar to those described in Section E.4.1.4 for cleanup of the PWR primary coolant system. An oxalic-peroxide-gluconic (OPG) solution is used to dissolve fuel debris. An EDTA/oxalic/citric acid solution is employed to remove fission product plateout. Piping jumpers are installed as needed to complete loops and facilitate the circulation of decontamination solutions. The RRC system pumps are assumed to be operable and are used for the circulation of decontamination and flush solutions following the scenario 1 and scenario 2 accidents. Extensive repairs to pump motors are assumed to be necessary prior to the use of these pumps following the scenario 3 accident.

K.3.3.2 Schedules and Cleanup Worker Requirements for Accident Cleanup in the Reactor Building and the Containment Vessel

Task schedules and sequences and cleanup worker requirements for accident cleanup in the reactor building and containment vessel following the postulated BWR accidents are shown in Figures K.3-2 through K.3-4. Accident cleanup is postulated 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. These time requirements are in addition to the requirements for preparations for cleanup given in Section K.1.

The utility staff organization postulated for BWR accident cleanup is the same as that postulated for PWR accident cleanup and shown in Table E.4-8 of Appendix E. This staff organization includes a plant operations branch and several support branches (e.g., engineering, health and safety, security, contracts and accounting, and quality assurance) as well as the cleanup staff. Cleanup staff functions are described in Section E.4.2. The cleanup worker requirements shown in Figures K.3-2 through K.3-4 are the requirements for the actual performance of cleanup tasks and do not include additional personnel needed to maintain compliance with occupational dose limitations (see Section K.3.3.4).

With the exception of assumption number 8 that relates to time requirements for defueling operations, the bases and assumptions used to estimate time and manpower requirements for BWR accident cleanup are the same as those given in Section E.4.2 for PWR accident cleanup. Because BWR defueling operations are performed outside the containment in a relatively low-radiation environment, it is postulated that BWR post-accident defueling

CLEANUP TASK (SHIFTS PER DAY[a]/DURATION IN MONTHS)		CREW LEADER	UTILITY OPERATOR	LABORER	CRAFTSMAN	HEALTH PHYS. TECHNICIAN
		MAN-DAYS PER SHIFT[b]				
INSTALLATION OF DEMINERALIZER SYSTEM	(2/3[c])	1	2	1	3	1
PROCESSING OF CONTAMINATED LIQUIDS						
CONTAINMENT SPRAY SYSTEM WATER	(3/0.5)		2	1		
DRYWELL HOSE WASH WATER	(3/1)		2	1		
SUPPRESSION POOL WATER	(3/5)		2	1		
SUPPRESSION CHAMBER HOSE WASH WATER	(3/1)		2	1		
REACTOR COOLANT SYSTEM WATER	(3/1)		2	1		
REACTOR WELL POOL WATER	(3/1.5)		2	1		
EDTA DECONTAMINATION SOLUTION	(3/1)		2	1		
COOLANT SYSTEM FLUSH WATER	(3/1)		2	1		
DECONTAMINATION OF REACTOR BUILDING & CONTAINMENT VESSEL						
REMOTE WASHDOWN OF CONTAINMENT VESSEL	(1/0.5)	1	2	2	2	1
HIGH-PRESSURE HOSE WASH OF DRYWELL AREA	(2/1)	1	5	3	2	1
HIGH-PRESSURE HOSE WASH OF SUPPRESSION CHAMBER	(2/1)	1	5	3	2	1
DECONTAMINATE & REPAIR SUPPORT SYSTEMS	(2/2)	1	4	2	8	1
HANDS-ON DECONTAMINATION OF DRYWELL AREA	(2[d]/6)	2	4	4		2
DEFUELING OF THE REACTOR						
MOBILIZATION FOR DEFUELING	(2/1)	1	3	2	2	1
RPV HEAD REMOVAL	(2/0.5)	1	3	2	2	1
REMOVE DRYER & SEPARATOR & INSPECT CORE	(2/1)	1	3	2	2	1
REMOVE FUEL	([e]/4)	1	6	2	2	2
DECONTAMINATE REACTOR WELL CAVITY	(2/1)	1	3	2	2	1
CLEANUP OF RRC AND RWCU SYSTEMS						
PREPARATIONS FOR COOLANT SYSTEM CLEANUP	(2/1)	1	3	3	4	1
MIX, INJECT & CIRCULATE EDTA SOLUTION	(3/1)	1	2	1	1	1
DRAIN EDTA SOLUTION	(3/1)	1	2	1	1	1
CIRCULATE & DRAIN RINSE SOLUTION	(3/1)	1	2	1	1	1
SUPPORT OPERATIONS						
WASTE PROCESSING & PACKAGING	(2/20)	1	2	2		1
CONSTRUCTION & MAINTENANCE	(2/16)	1			6	1
FINAL RADIATION SURVEY	(2/1)	1		2		4

TIME (MONTHS) AFTER START OF REACTOR BUILDING CLEANUP OPERATIONS: 1–30 ([f])

LABOR CATEGORY	TOTAL MAN-MONTHS	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30
		MAN-MONTHS PER WORKING MONTH[g]																													
CREW LEADER	174	8	9	10	8	9	9	8	7	12	12	12	12	12	12	6	6	6	6	6	4										
UTILITY OPERATOR	532	22	25	24	20	24	26	25	22	24	24	52	52	52	52	18	14	12	20	20	4										
LABORER	309	14	17	18	16	18	18	17	15	16	14	20	20	20	20	12	12	8	12	12	8										
CRAFTSMAN	333	21	20	18	12	16	20	18	16	16	16	28	28	28	28	16	20	4	4	4											
HEALTH PHYSICS TECHNICIAN	204	8	9	10	8	9	9	8	7	8	8	20	20	20	20	6	6	6	6	6	10										

(a) TWO SHIFTS PER DAY OPERATE ON 5-DAY WEEK. THREE SHIFTS PER DAY OPERATE ON 7-DAY WEEK.
(b) ASSUME 20 MAN-DAYS PER MAN-MONTH.
(c) INSTALLATION OF THE DEMINERALIZER SYSTEM IS POSTULATED TO TAKE PLACE DURING THE FINAL 3 MONTHS OF PREPARATIONS FOR ACCIDENT CLEANUP FOLLOWING THE SCENARIO 1 ACCIDENT.
(d) ONE SHIFT PER DAY FOR LAST 2 MONTHS.
(e) ONCE STARTED, REMOVAL OF FUEL CONTINUES UNINTERRUPTED ON AN AROUND-THE-CLOCK BASIS UNTIL COMPLETED. THIS REQUIRES 6 SHIFTS PER DAY, 7-DAYS PER WEEK.
(f) CREW TRAINING FOR FUEL REMOVAL.
(g) MANPOWER REQUIREMENTS ARE SHOWN ROUNDED TO THE NEAREST MAN-MONTH. REQUIREMENTS ARE BASED ON NECESSARY LABOR TO COMPLETE TASKS AND DO NOT INCLUDE EXTRA MANPOWER NEEDED TO COMPLY WITH OCCUPATIONAL DOSE LIMITS.

FIGURE K.3-2. Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 1 Accident

CLEANUP TASK (SHIFTS PER DAY[a]/DURATION IN MONT	MAN-DAYS PER SHIFT[a] CREW LEADER	UTILITY OPERATOR	LABORER	CRAFTSMAN	HEALTH PHYS. TECHNICIAN
PREPARATIONS FOR PROCESSING CONTAMINATED LIQUID					
INITIAL DECONTAMINATION OF REACTOR BLDG.		4	4		2
INSTALLATION OF DEMINERALIZER SYSTEM	1	2	1	3	1
PROCESSING OF CONTAMINATED LIQUIDS					
SUPPRESSION POOL WATER		2	1		
CONTAINMENT SPRAY SYSTEM WATER		2	1		
DRYWELL HOSE WASH WATER		2	1		
SUPPRESSION CHAMBER HOSE WASH WATER		2	1		
REACTOR COOLANT SYSTEM WATER		2	1		
REACTOR WELL POOL WATER		2	1		
OPG DECONTAMINATION SOLUTION		2	1		
EDTA DECONTAMINATION SOLUTION		2	1		
COOLANT SYSTEM FLUSH WATER		2	1		
DECONTAMINATION OF REACTOR BUILDING & CONTAINM					
DECONTAMINATE REACTOR BUILDING	2	4	4		2
REMOTE WASHDOWN OF CONTAINMENT VESSEL	1	2	2	2	1
HIGH-PRESSURE HOSE WASH OF DRYWELL AREA	1	5	3	2	1
HIGH-PRESSURE HOSE WASH OF SUPPRESSION CHAMB	1	5	3	2	1
DECONTAMINATE & REPAIR SUPPORT SYSTEMS	1	4	2	4	1
HANDS-ON DECONTAMINATION OF CONTAINMENT	2	4	4		2
DEFUELING OF THE REACTOR					
MOBILIZATION FOR DEFUELING	2	4	2	4	2
RPV HEAD REMOVAL	1	3	2	2	1
REMOVE DRYER & SEPARATOR & INSPECT CORE	1	3	2	2	1
REMOVE FUEL	1	6	2	2	2
REMOVE FUEL DEBRIS	1	6	2	2	2
DECONTAMINATE REACTOR WELL CAVITY	1	3	2	2	1
CLEANUP OF RRC AND RWCU SYSTEMS					
PREPARATIONS FOR COOLANT SYSTEM CLEANUP	1	4	3	4	1
MIX, INJECT & CIRCULATE OPG SOLUTION	1	2	1	1	1
DRAIN OPG SOLUTION	1	2	1	1	1
CIRCULATE & DRAIN RINSE SOLUTION	1	2	1	1	1
MIX, INJECT & CIRCULATE EDTA SOLUTION	1	2	1	1	1
DRAIN EDTA SOLUTION	1	2	1	1	1
CIRCULATE & DRAIN RINSE SOLUTION	1	2	1	1	1
SUPPORT OPERATIONS					
WASTE PROCESSING & PACKAGING	1	2	2		1
CONSTRUCTION & MAINTENANCE	1			6	1
FINAL RADIATION SURVEY	1		2		4

LABOR CATEGORY
CREW LEADER
UTILITY OPERATOR
LABORER
CRAFTSMAN
HEALTH PHYSICS TECHNICIAN

(a) TWO SHIFTS PER DAY OPERATE ON 5-DAY WEEK.
(b) ASSUME 20 MAN-DAYS PER MAN-MONTH.
(c) ONE SHIFT PER DAY FOR FIRST FOUR MONTHS.
(d) ONCE STARTED, REMOVAL OF FUEL CONTINUES UNI UNTIL COMPLETED. THIS REQUIRES SIX SHIFTS PEI
(e) CREW TRAINING FOR FUEL REMOVAL.
(f) MANPOWER REQUIREMENTS ARE SHOWN ROUNDED T(BASED ON NECESSARY LABOR TO COMPLETE TASKS TO COMPLY WITH OCCUPATIONAL DOSE LIMITS.

FIGURE K.3-3. Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 2 Accident

CLEANUP TASK (SHIFTS PER DAY[a]/DURAT	63 64	CREW LEADER	UTILITY OPERATOR	LABORER	CRAFTSMAN	HEALTH PHYS. TECHNICIAN
		MAN-DAYS PER SHIFT[b]				
INSTALLATION OF DEMINERALIZER SYSTEM		1	2	1	3	1
PROCESSING OF CONTAMINATED LIQUIDS						
REACTOR BUILDING SUMP WATER			2	1		
REACTOR BUILDING HOSE WASH WATER			2	1		
SUPPRESSION POOL WATER			2	1		
CONTAINMENT SPRAY SYSTEM WATER			2	1		
DRYWELL HOSE WASH WATER			2	1		
SUPPRESSION CHAMBER HOSE WASH			2	1		
WATER			2	1		
REACTOR COOLANT SYSTEM WATER			2	1		
REACTOR WELL POOL WATER			2	1		
OPG DECONTAMINATION SOLUTION			2	1		
EDTA DECONTAMINATION SOLUTION			2	1		
COOLANT SYSTEM FLUSH WATER						
DECONTAMINATION OF REACTOR BLDG. AN VESSEL						
DECONTAMINATE R.B. ABOVE MEZZANINE LEVEL		2	4	4		2
DECONTAMINATE R.B. BELOW MEZZANINE LEVEL		2	4	4		2
REMOTE WASHDOWN OF CONTAINMENT VESSEL		1	2	2	2	1
HIGH-PRESSURE HOSE WASH OF DRYWELL AREA		1	5	3	2	1
HIGH-PRESSURE HOSE WASH OF SUPPRESSION CHAMBER		1	5	3	2	1
DECONTAMINATE AND REPAIR SUPPORT SYSTEMS		1	4	2	4	1
HANDS-ON DECONTAMINATION OF CONTAINMENT		2	4	4		2
DEFUELING OF THE REACTOR						
MOBILIZATION FOR DEFUELING		1	3	2	2	1
RPV HEAD REMOVAL		1	3	2	2	1
REMOVE DRYER AND SEPARATOR AND INSPECT CORE		1	3	2	3	1
REMOVE FUEL		1	6	2	3	2
REMOVE FUEL DEBRIS		1	6	2	3	2
DECONTAMINATE REACTOR WELL CAVITY		1	3	2	[illegible]	1
CLEANUP OF RRC AND RWCU SYSTEMS						
PREPARATIONS FOR COOLANT SYSTEM CLEANUP		1	4	3	1	1
MIX, INJECT AND CIRCULATE OPG SOLUTION		1	2	1	1	1
DRAIN OPG SOLUTION		1	2	1	1	1
CIRCULATE AND DRAIN RINSE SOLUTION		1	2	1	1	1
MIX, INJECT AND CIRCULATE EDTA SOLUTION		1	2	1	1	1
DRAIN EDTA SOLUTION		1	2	1	1	1
CIRCULATE AND DRAIN RINSE SOLUTION		1	2	1	1	1
SUPPORT OPERATIONS						
WASTE PROCESSING AND PACKAGING		1	2	2		1
CONSTRUCTION AND MAINTENANCE		1			6	1
FINAL RADIATION SURVEY		1		2		4

LABOR CATEGORY	MAN 63 64
CREW LEADER	4
UTILITY OPERATOR	4
LABORER	8
CRAFTSMAN	
HEALTH PHYSICS TECHNICIAN	10

(a) TWO SHIFTS PER DAY OPERATE ON 5-D
(b) ASSUME 20 MAN-DAYS PER MAN-MONTH.
(c) ONCE STARTED, FUEL REMOVAL CONTI COMPLETED. THIS REQUIRES SIX SHIFT
(d) CREW TRAINING FOR FUEL REMOVAL.
(e) MANPOWER REQUIREMENTS ARE SHOWN ON NECESSARY LABOR TO COMPLETE T WITH OCCUPATIONAL DOSE LIMITS.

FIGURE K.3-4. Task Schedule and Sequence and Cleanup Worker Requirements for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 3 Accident

TABLE K.3-5. Estimated Occupational Radiation Doses for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 1 Accident

Cleanup Task	Average Dose Rate (rem/hr)	Crew Leader Exposure (man-hr)	Crew Leader Dose (man-rem)	Utility Operator Exposure (man-hr)	Utility Operator Dose (man-rem)	Laborer Exposure (man-hr)	Laborer Dose (man-rem)	Craftsman Exposure (man-hr)	Craftsman Dose (man-rem)	Health Physics Technician Exposure (man-hr)	Health Physics Technician Dose (man-rem)	Task Totals Exposure (man-hr)	Task Totals Dose (man-rem)
Installation of Demineralizer System	0.010	480	4.8	960	9.6	480	4.8	1 440	14.4	480	4.8	3 840	38.4
Processing of Contaminated Liquids													
Containment Spray System Water	0.008			540	4.3	270	2.2					810	6.5
Drywell Hose Wash Water	0.008			1 080	8.6	540	4.3					1 620	12.9
Suppression Pool Water	0.008			5 400	43.2	2 700	21.6					8 100	64.8
Suppression Chamber Hose Wash Water	0.008			1 080	8.6	540	4.3					1 620	12.9
Reactor Coolant System Water	0.008			1 080	8.6	540	4.3					1 620	12.9
Reactor Well Pool Water	0.008			1 620	13.0	810	6.5					2 430	19.5
EDTA Decontamination Solution	0.008			1 080	8.6	540	4.3					1 620	12.9
Coolant System Flush Water	0.008			1 080	8.6	540	4.3					1 620	12.9
Subtotals				12 960	103.5	6 480	51.8					19 440	155.3
Decontamination of Reactor Building and Containment Vessel													
Remote Washdown of Containment Vessel	0.100	8	0.8	16	1.6	16	1.6	16	1.6	8	0.8	64	6.4
High-Pressure Hose Wash of Drywell Area	0.040	160	6.4	800	32.0	480	19.2	320	12.8	160	6.4	1 920	76.8
High-Pressure Hose Wash of Suppression Chamber	0.030	160	4.8	800	24.0	480	14.4	320	9.6	160	4.8	1 920	57.6
Decontaminate and Repair Support Systems	0.010	320	3.2	1 280	12.8	640	6.4	1 280	12.8	320	3.2	3 840	38.4
Hands-on Decontamination of Drywell Area	0.020	1 600	32.0	3 200	64.0	3 200	64.0			1 600	32.0	9 600	192.0
Subtotals		2 248	47.2	6 096	134.4	4 816	105.6	1 936	36.8	2 248	47.2	17 344	371.2
Defueling of the Reactor													
Mobilization for Defueling	0.010	160	1.6	480	4.8	320	3.2	320	3.2	160	1.6	1 440	14.4
RPV Head Removal	0.010	80	0.8	240	2.4	160	1.6	160	1.6	80	0.8	720	7.2
Remove Dryer and Separator and Inspect Core	0.010	160	1.6	480	4.8	320	3.2	320	3.2	160	1.6	1 440	14.4
Remove Fuel	0.012	2 880	34.6	17 280	207.4	5 760	69.1	5 760	69.1	5 760	69.1	37 440	449.3
Decontaminate Reactor Well Cavity	0.010	160	1.6	480	4.8	320	3.2	320	3.2	160	1.6	1 440	14.4
Subtotals		3 440	40.2	18 960	224.2	6 880	80.3	6 880	80.3	6 320	74.7	42 480	499.7
Cleanup of RRC and RWCU Systems													
Preparations for Coolant System Cleanup	0.010	160	1.6	480	4.8	480	4.8	640	6.4	160	1.6	1 920	19.2
Mix, Inject and Circulate EDTA Solution	0.005	360	1.8	720	3.6	360	1.8	360	1.8	360	1.8	2 160	10.8
Drain EDTA Solution	0.005	360	1.8	720	3.6	360	1.8	360	1.8	360	1.8	2 160	10.8
Circulate and Drain Rinse Solution	0.005	360	1.8	720	3.6	360	1.8	360	1.8	360	1.8	2 160	10.8
Subtotals		1 240	7.0	2 640	15.6	1 560	10.2	1 720	11.8	1 240	7.0	8 400	51.6
Support Operations													
Waste Processing and Packaging	0.005	4 800	24.0	9 600	48.0	9 600	48.0			4 800	24.0	28 800	144.0
Construction and Maintenance	0.010	2 560	25.6					15 360	153.6	2 560	25.6	20 480	204.8
Final Radiation Survey	0.005	240	1.2			480	2.4			960	4.8	1 680	8.4
Subtotals		7 600	50.8	9 600	48.0	10 080	50.4	15 360	153.6	8 320	54.4	50 960	357.2
Totals		15 008	150.0	51 216	535.3	30 296	303.1	27 336	296.9	18 608	188.1	142 464	1 473.4

can proceed with greater facility than PWR post-accident defueling. Accordingly, the following bases are used to estimate time requirements for BWR defueling operations:

a) Removal and storage of an undamaged assembly requires 2 hours, including time for inspection with periscopes and underwater TV cameras prior to removal.

b) Removal and storage of a fuel assembly that has experienced cladding failure requires 6 hours, including time to inspect the assembly prior to removal and to overpack the assembly in a stainless steel canister prior to storage.

c) Removal, overpacking, and storage of a fuel assembly that is damaged as a result of fuel melting requires 15 to 30 hours, depending on the extent of the damage.

d) Time estimates for fuel removal based on the above assumptions are multiplied by 1.25 to allow for inefficiencies associated with work in a radiation environment.

A comparison of estimated time requirements for BWR defueling, given in Figures K.3-2 through K.3-4, with estimated time requirements for PWR defueling, given in Figures E.4-5 through E.4-7, shows that approximately twice as much time is required for BWR defueling as is required for PWR defueling. This is because the number of fuel assemblies in the BWR core (764) is substantially greater than the number in the PWR core (193).

K.3.3.3 Occupational Doses for Accident Cleanup in the Reactor Building and the Containment Vessel

Estimated occupational radiation doses to cleanup workers during accident cleanup in the BWR reactor building and containment vessel following the three postulated accidents are shown in Tables K.3-5 through K.3-7. The occupational doses shown in the tables are external doses from gamma radiation. Workers are assumed to use respiratory devices as necessary to protect against the inhalation of radioactive particulates.

TABLE K.3-6. Estimated Occupational Radiation Doses for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 2 Accident

Cleanup Task	Average Dose Rate (rem/hr)	Crew Leader		Utility Operator		Laborer		Craftsman		Health Physics Technician		Task Totals	
		Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)	Exposure (man-hr)	Dose (man-rem)
Installation of Demineralizer System	0.010	480	4.8	960	9.6	480	4.8	1 440	14.4	480	4.8	3 840	38.4
Processing of Contaminated Liquids													
Suppression Pool Water	0.008			8 100	64.8	4 050	32.4					12 150	97.2
Containment Spray System Water	0.008			540	4.3	270	2.2					810	6.5
Drywell Hose Wash Water	0.008			1 620	13.0	810	6.5					2 430	19.5
Suppression Chamber Hose Wash Water	0.008			1 620	13.0	810	6.5					2 430	19.5
Reactor Coolant System Water	0.008			1 080	8.6	540	4.3					1 620	12.9
Reactor Well Pool Water	0.008			1 620	13.0	810	6.5					2 430	19.5
OPG Decontamination Solution	0.008			1 620	13.0	810	6.5					2 430	19.5
EDTA Decontamination Solution	0.008			1 620	13.0	810	6.5					2 430	19.5
Coolant System Flush Water	0.008			2 160	17.3	1 080	8.6					3 240	25.9
Subtotals				19 980	160.0	9 990	80.0					29 970	240.0
Decontamination of Reactor Building and Containment Vessel													
Decontaminate Reactor Building	0.015	1 280	19.2	2 560	38.4	2 560	38.4			1 280	19.2	7 680	115.2
Remote Washdown of Containment Vessel	0.200	16	3.2	32	6.4	32	6.4	32	6.4	16	3.2	128	25.6
High-Pressure Hose Wash of Drywell Area	0.060	240	14.4	1 200	72.0	720	43.2	480	28.8	240	14.4	2 880	172.8
High-Pressure Hose Wash of Suppression Chamber	0.045	320	14.4	1 600	72.0	960	43.2	640	28.8	320	14.4	3 840	172.8
Decontaminate and Repair Support Systems	0.020	800	16.0	3 200	64.0	1 600	32.0	3 200	64.0	800	16.0	9 600	192.0
Hands-on Decontamination of Containment	0.030	2 240	67.2	4 480	134.4	4 480	134.4			2 240	67.2	13 440	403.2
Subtotals		4 896	134.4	13 072	387.2	10 352	297.6	4 352	128.0	4 896	134.4	37 568	1 081.6
Defueling of the Reactor													
Mobilization for Defueling	0.010	320	3.2	640	6.4	320	3.2	640	6.4	320	3.2	2 240	22.4
RPV Head Removal	0.010	80	0.8	240	2.4	160	1.6	160	1.6	80	0.8	720	7.2
Remove Dryer and Separator and Inspect Core	0.015	320	4.8	960	14.4	640	9.6	640	9.6	320	4.8	2 880	43.2
Remove Fuel	0.015	5 760	86.4	34 560	518.4	11 520	172.8	11 520	172.8	11 520	172.8	74 880	1 123.2
Remove Fuel Debris	0.015	160	2.4	960	14.4	320	4.8	320	4.8	320	4.8	2 080	31.2
Decontaminate Reactor Well Cavity	0.010	160	1.6	480	4.8	320	3.2	320	3.2	160	1.6	1 440	14.4
Subtotals		6 800	99.2	37 840	560.8	13 280	195.2	13 600	198.4	12 720	188.0	84 240	1 241.6
Cleanup of RRC and RWCU Systems													
Preparations for Coolant System Cleanup	0.015	240	3.6	960	14.4	720	10.8	960	14.4	240	3.6	3 120	46.8
Mix, Inject and Circulate OPG Solution	0.010	360	3.6	720	7.2	360	3.6	360	3.6	360	3.6	2 160	21.6
Drain OPG Solution	0.010	360	3.6	720	7.2	360	3.6	360	3.6	360	3.6	2 160	21.6
Circulate and Drain Rinse Solution	0.010	360	3.6	720	7.2	360	3.6	360	3.6	360	3.6	2 160	21.6
Mix, Inject and Circulate EDTA Solution	0.010	360	3.6	720	7.2	360	3.6	360	3.6	360	3.6	2 160	21.6
Drain EDTA Solution	0.010	360	3.6	720	7.2	360	3.6	360	3.6	360	3.6	2 160	21.6
Circulate and Drain Rinse Solution	0.010	360	3.6	720	7.2	360	3.6	360	3.6	360	3.6	2 160	21.6
Subtotals		2 400	25.2	5 280	57.6	2 880	32.4	3 120	36.0	2 400	25.2	16 080	176.4
Support Operations													
Waste Processing and Packaging	0.008	9 360	74.9	18 720	149.8	18 720	149.8			9 360	74.9	56 160	449.4
Construction and Maintenance	0.020	5 120	102.4					30 720	614.4	5 120	102.4	40 960	819.2
Final Radiation Survey	0.010	240	2.4			480	4.8			960	9.6	1 680	16.8
Subtotals		14 720	179.7	18 720	149.8	19 200	154.6	30 720	614.4	15 440	186.9	98 800	1 285.4
Totals		29 296	443.3	95 852	1 325.0	56 182	764.6	53 232	991.2	35 936	539.3	270 498	4 063.4

TABLE K.3-7. Estimated Occupational Radiation Doses for Accident Cleanup in the Reactor Building Following the Postulated BWR Scenario 3 Accident

Cleanup Task	Average Dose Rate (rem/hr)	Crew Leader Exposure (man-hr)	Crew Leader Dose (man-rem)	Utility Operator Exposure (man-hr)	Utility Operator Dose (man-rem)	Laborer Exposure (man-hr)	Laborer Dose (man-rem)	Craftsman Exposure (man-hr)	Craftsman Dose (man-rem)	Health Physics Technician Exposure (man-hr)	Health Physics Technician Dose (man-rem)	Task Totals Exposure (man-hr)	Task Totals Dose (man-rem)
Installation of Demineralizer System	0.015	480	7.2	960	14.4	480	7.2	1 440	21.6	480	7.2	3 840	57.6
Processing of Contaminated Liquids													
Reactor Building Sump Water	0.008			1 080	8.6	540	4.3					1 620	12.9
Reactor Building Hose Wash Water	0.008			540	4.3	270	2.2					810	6.5
Suppression Pool Water	0.008			8 100	64.8	4 050	32.4					12 150	97.2
Containment Spray System Water	0.008			1 080	8.6	540	4.3					1 620	12.9
Drywell Hose Wash Water	0.008			2 160	17.3	1 080	8.6					3 240	25.9
Suppression Chamber Hose Wash Water	0.008			2 160	17.3	1 080	8.6					3 240	25.9
Reactor Coolant System Water	0.008			1 080	8.6	540	4.3					1 620	12.9
Reactor Well Pool Water	0.008			1 620	13.0	810	6.5					2 430	19.5
OPG Decontamination Solution	0.008			2 160	17.3	1 080	8.6					3 240	25.9
EDTA Decontamination Solution	0.008			2 160	17.3	1 080	8.6					3 240	25.9
Coolant System Flush Water	0.008			2 160	17.3	1 080	8.6					3 240	25.9
Subtotals				24 300	194.4	12 150	97.0					36 450	291.4
Decontamination of Reactor Building and Containment Vessel													
Decontaminate Reactor Building Above Mezzanine Level	0.020	1 280	25.6	2 560	51.2	2 560	51.2			1 280	25.6	7 680	153.6
Decontaminate Reactor Building Below Mezzanine Level	0.030	1 760	52.8	3 520	105.6	3 520	105.6			1 760	52.8	10 560	316.8
Remote Washdown of Containment Vessel	0.400	24	9.6	48	19.2	48	19.2	48	19.2	24	9.6	192	76.8
High-Pressure Hose Wash of Drywell Area	0.100	320	32.0	1 600	160.0	960	96.0	640	64.0	320	32.0	3 840	384.0
High-Pressure Hose Wash of Suppression Chamber	0.075	320	24.0	1 600	120.0	960	72.0	640	48.0	320	24.0	3 840	288.0
Decontaminate and Repair Support Systems	0.035	960	33.6	3 840	134.4	1 920	67.2	3 840	134.4	960	33.6	11 520	403.2
Hands-on Decontamination of Containment	0.050	3 200	160.0	6 400	320.0	6 400	320.0			3 200	160.0	19 200	960.0
Subtotals		7 864	337.6	19 568	910.4	16 368	731.2	5 168	265.6	7 864	337.6	56 832	2 582.4
Defueling of the Reactor													
Mobilization for Defueling	0.010	320	3.2	960	9.6	640	6.4	640	6.4	320	3.2	2 880	28.8
RPV Head Removal	0.010	160	1.6	480	4.8	320	3.2	320	3.2	160	1.6	1 440	14.4
Remove Dryer and Separator and Inspect Core	0.020	560	11.2	1 680	33.6	1 120	22.4	1 680	33.6	560	11.2	5 600	112.0
Remove Fuel	0.020	14 400	288.0	86 400	1 728.0	28 800	576.0	43 200	864.0	28 800	576.0	201 600	4 032.0
Remove Fuel Debris	0.020	320	6.4	1 920	38.4	640	12.8	960	19.2	640	12.8	4 480	89.6
Decontaminate Reactor Well Cavity	0.010	160	1.6	480	4.8	320	3.2	320	3.2	160	1.6	1 440	14.4
Subtotals		15 920	312.0	91 920	1 819.2	31 840	624.0	47 120	929.6	30 640	606.4	217 440	4 291.2
Cleanup of RRC and RWCU Systems													
Preparations for Coolant System Cleanup	0.025	480	12.0	1 920	48.0	1 440	36.0	1 920	48.0	480	12.0	6 240	156.0
Mix, Inject and Circulate OPG Solution	0.015	360	5.4	720	10.8	360	5.4	360	5.4	360	5.4	2 160	32.4
Drain OPG Solution	0.015	540	8.1	1 080	16.2	540	8.1	540	8.1	540	8.1	3 240	48.6
Circulate and Drain Rinse Solution	0.015	360	5.4	720	10.8	360	5.4	360	5.4	360	5.4	2 160	32.4
Mix, Inject and Circulate EDTA Solution	0.015	360	5.4	720	10.8	360	5.4	360	5.4	360	5.4	2 160	32.4
Drain EDTA Solution	0.015	540	8.1	1 080	16.2	540	8.1	540	8.1	540	8.1	3 240	48.6
Circulate and Drain Rinse Solution	0.015	360	5.4	720	10.8	360	5.4	360	5.4	360	5.4	2 160	32.4
Subtotals		3 000	49.8	6 960	123.6	3 960	73.8	4 440	85.8	3 000	49.8	21 360	382.8
Support Operations													
Waste Processing and Packaging	0.010	15 120	151.2	30 240	302.4	30 240	302.4			15 120	151.2	90 720	907.2
Construction and Maintenance	0.030	8 800	264.0					52 800	1 584.0	8 800	264.0	70 400	2 112.0
Final Radiation Survey	0.015	240	3.6			480	7.2			960	14.4	1 680	25.2
Subtotals		24 160	418.8	30 240	302.4	30 720	309.6	52 800	1 584.0	24 880	429.6	162 800	3 044.4
Totals		51 424	1 125.4	173 948	3 364.4	95 518	1 842.8	110 968	2 886.6	66 864	1 430.6	498 722	10 649.8

Dose calculations are based on time and manpower requirements shown in the task schedules for accident cleanup in the reactor building and the containment (Figures K.3-2, K.3-3, and K.3-4). Exposure hours are estimated on the basis that workers engaged in decontamination operations and in the installation and repair of systems needed for accident cleanup spend an average of 4 hours inside the containment building during an 8-hour shift. Workers who operate the demineralizer and evaporator systems, monitor the reactor coolant system cleanup operations, or are engaged in waste packaging activities spend an average of 6 hours in a radiation area during an 8-hour shift.

Total estimated occupational radiation doses to cleanup workers for accident cleanup in the reactor building and the containment are 1473 man-rem following the scenario 1 accident, 4063 man-rem following the scenario 2 accident, and 10,650 man-rem following the scenario 3 accident.

K.3.3.4 Staff Requirements for Accident Cleanup in the Reactor Building and the Containment Vessel

Cleanup worker requirements for accident cleanup in the BWR reactor building and the containment following the three postulated accidents can be obtained from the cleanup task schedules shown in Figures K.3-2, K.3-3, and K.3-4. Cleanup worker requirements are estimated to be 129 man-years following the scenario 1 accident, 246 man-years following the scenario 2 accident, and 455 man-years following the scenario 3 accident. These requirements include only the labor to actually complete the designated cleanup tasks and do not include either: 1) the additional labor needed to maintain compliance with occupational radiation dose limits, or 2) the management and support staff required during cleanup operations.

Adjustments in cleanup worker requirements to comply with occupational radiation dose limits are shown in Tables K.3-8, K.3-9, and K.3-10 for the three reference accidents. Adjusted cleanup worker requirements are 304 man-years following the scenario 1 accident, 821 man-years following the scenario 2 accident, and 2158 man-years following the scenario 3 accident.

Total utility staff labor requirements for accident cleanup in the reactor building and the containment vessel following the postulated BWR

TABLE K.3-8. Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Scenario 1 Accident

Worker Category	Estimated Worker Requirements[a] (man-yr)	Estimated Occupational Dose: Total[b] (man-rem)	Estimated Occupational Dose: Individual Average (man-rem/man-yr)	Adjustment Factor[c]	Adjusted Worker Requirements (man-yr)
Crew Leader	14.5	150.0	10.4	2.1	30.5
Utility Operator	44.3	535.3	12.1	2.5	110.8
Laborer	25.8	303.1	11.8	2.4	62.0
Craftsman	27.8	296.9	10.7	2.2	61.2
Health Physics Technician	17.0	188.1	11.1	2.3	39.1
Totals	129.4	1473.4			303.6

(a) Based on Figure K.3-2.
(b) Based on Table K.3-5.
(c) Increase in worker requirements necessary to reduce average individual dose to $\leq$ 5 man-rem/man-year.

TABLE K.3-9. Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Scenario 2 Accident

Worker Category	Estimated Worker Requirements[(a)] (man-yr)	Estimated Occupational Dose: Total[(b)] (man-rem)	Estimated Occupational Dose: Individual Average (man-rem/man-yr)	Adjustment Factor[(c)]	Adjusted Worker Requirements (man-yr)
Crew Leader	27.6	443.3	16.1	3.3	91.1
Utility Operator	83.4	1325.0	15.9	3.2	266.9
Laborer	47.9	764.6	16.0	3.2	153.3
Craftsman	53.9	991.2	18.4	3.7	199.5
Health Physics Technician	33.3	539.3	16.2	3.3	109.9
Totals	246.1	4063.4			820.7

(a) Based on Figure K.3-3.
(b) Based on Table K.3-6.
(c) Increase in worker requirements necessary to reduce average individual dose to ≤ 5 man-rem/man-year.

TABLE K.3-10. Adjustments to Cleanup Worker Requirements to Comply with Occupational Radiation Dose Limitations for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Scenario 3 Accident

Worker Category	Estimated Worker Requirements[a] (man-yr)	Estimated Occupational Dose: Total[b] (man-rem)	Estimated Occupational Dose: Individual Average (man-rem/man-yr)	Adjustment Factor[c]	Adjusted Worker Requirements (man-yr)
Crew Leader	48.3	1 125.4	23.3	4.7	227.0
Utility Operator	154.3	3 364.4	21.8	4.4	679.0
Laborer	81.5	1 842.8	22.6	4.6	374.9
Craftsman	110.4	2 886.6	26.2	5.3	585.2
Health Physics Technician	60.8	1 430.6	23.6	4.8	291.9
Totals	455.3	10 649.8			2 158.0

(a) Based on Figure K.3-4.
(b) Based on Table K.3-7.
(c) Increase in worker requirements necessary to reduce average individual dose to $\leq$ 5 man-rem/man-year.

accidents are shown in Table K.3-11. Staff labor requirements include management and support staff and adjusted cleanup worker requirements but do not include contractor personnel. The staff labor man-years shown in Table K.3-11 are used to compute labor costs for accident cleanup as described in Section K.4.

K.4 DETAILS OF COSTS OF ACCIDENT CLEANUP

Details of the costs of accident cleanup in the reference BWR following the postulated accidents are presented in this section. Costs are based on the technical requirements, manpower needs, and cleanup schedules described in Section K.3, and are given in early-1981 dollars. Unit cost information used as bases for these cost estimates is given in Appendix I.

Total estimated costs of accident cleanup following the reference accidents are summarized in Table K.4-1. Accident cleanup costs are estimated to be about $128 million following the scenario 1 accident, $228 million following the scenario 2 accident, and $421 million following the scenario 3 accident. The costs include planning and preparation costs as well as the actual costs of cleanup. 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.

K.4.1 Details of Costs of Preparations for Accident Cleanup

The estimated costs of preparations for accident cleanup following the reference BWR accidents are summarized in Table K.4-2. Preparations for accident cleanup following the scenario 1 accident are estimated to require 1.5 years and to cost approximately $30.1 million. Preparations for cleanup following the scenario 2 accident are estimated to require 2 years and to cost approximately $49.7 million. Preparations for cleanup following the scenario 3 accident are estimated to require 3 years and to cost approximately $90.3 million.

Costs of preparations for cleanup include the costs of maintaining the reactor in a safe shutdown condition as well as the costs of completing the preparations activities described in Section K.3.1. About half of these costs

TABLE K.3-11. Estimated Utility Staff Labor Requirement for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Accidents

	Staff Labor Requirements					
	Scenario 1 Accident		Scenario 2 Accident		Scenario 3 Accident	
Position	man-years/year	Total man-years[(a)]	man-years/year	Total man-years[(b)]	man-years/year	Total man-years[(c)]
Plant Superintendent	1.0	1.7	1.0	3.3	1.0	5.3
Assistant Plant Superintendent	1.0	1.7	1.0	3.3	1.0	5.3
Consultants	3.0	5.1	6.0	19.8	10.0	53.0
Secretaries and Word Processors	8.0	13.6	10.0	33.0	20.0	106.0
Site Support Staff						
Health and Safety Supervisor	1.0	1.7	1.0	3.3	1.0	5.3
Health Physicist	1.0	1.7	1.0	3.3	1.0	5.3
Senior Health Physics Technician	8.0	13.6	8.0	26.4	12.0	63.6
Health Physics Technician[(d)]	8.0	13.6	8.0	26.4	12.0	63.6
Protective Equipment Attendant	4.0	6.8	8.0	26.4	12.0	63.6
Industrial Safety Specialist	1.0	1.7	1.0	3.3	1.0	5.3
Industrial Safety Technician	2.0	3.4	2.0	6.6	2.0	10.6
Security Supervisor	1.0	1.7	1.0	3.3	1.0	5.3
Security Shift Supervisor	4.0	6.8	4.0	13.2	4.0	21.2
Security Patrolman	48.0	81.6	48.0	158.4	48.0	254.4
Contracts and Accounting Supervisor	1.0	1.7	1.0	3.3	1.0	5.3
Accountant	1.0	1.7	1.0	3.3	2.0	10.6
Contracts Specialist	1.0	1.7	1.0	3.3	1.0	5.3
Insurance Specialist	1.0	1.7	1.0	3.3	2.0	10.6
Procurement Specialist	1.0	1.7	1.0	3.3	1.0	5.3
Clerk	2.0	3.4	4.0	13.2	6.0	31.8
Quality Assurance Supervisor	1.0	1.7	1.0	3.3	1.0	5.3
Quality Assurance Engineer	2.0	3.4	2.0	6.6	2.0	10.6
Quality Assurance Technician	2.0	3.4	2.0	6.6	2.0	10.6
Construction Engineering Supervisor	1.0	1.7	1.0	3.3	1.0	5.3
Engineer	6.0	10.2	8.0	26.4	12.0	63.6
Estimator	1.0	1.7	2.0	6.6	4.0	21.2
Draftsman	2.0	3.4	4.0	13.2	6.0	31.8
Subtotals		170.0		366.3		715.5
Plant Operations Staff						
Plant Operations Supervisor	1.0	1.7	1.0	3.3	1.0	5.3
Plant Chemist	1.0	1.7	1.0	3.3	1.0	5.3
Chemist	2.0	3.4	2.0	6.6	2.0	10.6
Reactor Operations Engineer	1.0	1.7	1.0	3.3	1.0	5.3
Engineer	2.0	3.4	2.0	6.6	2.0	10.6
Reactor Operations Shift Supervisor	4.0	6.8	4.0	13.2	4.0	21.2
Senior Reactor Operator	8.0	13.6	8.0	26.4	8.0	42.4
Reactor Operator	16.0	27.2	16.0	52.8	16.0	84.8
Utility Operator	16.0	27.2	16.0	52.8	16.0	84.8
Technician	16.0	27.2	20.0	66.0	24.0	127.2
Craft Supervisor	1.0	1.7	1.0	3.3	1.0	5.3
Crew Foreman	4.0	6.8	4.0	13.2	4.0	21.2
Craftsman[(e)]	8.0	13.6	12.0	39.6	12.0	63.6
Warehouseman	4.0	6.8	8.0	26.4	8.0	42.4
Tool Crib Attendant	4.0	6.8	8.0	26.4	8.0	42.4
Subtotals		149.6		343.2		572.4
Accident Cleanup Staff						
Cleanup Superintendent	1.0	1.7	1.0	3.3	1.0	5.3
Radioactive Shipment Specialist	1.0	1.7	1.0	3.3	1.0	5.3
Clerk	1.0	1.7	1.0	3.3	2.0	10.6
Shift Supervisor	4.0	6.8	4.0	13.2	4.0	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.

TABLE K.4-1. Summary of Estimated Total Costs of Accident Cleanup Following the Reference BWR Accidents

Cleanup Operation	Costs of Cleanup Following Scenario 1 Accident[a] ($ millions)	Cost of Cleanup Following Scenario 2 Accident[a] ($ millions)	Cost of Cleanup Following Sceanrio 3 Accident[a] ($ millions)
Preparations for Accident Cleanup	30.1	49.7	90.3
Accident Cleanup in the Radwaste Building	--[b]	--[b]	13.1[c]
Accident Cleanup in the Reactor Building and the Containment	98.4	178.5	317.5
Total Accident Cleanup Costs	128.5	228.2	420.9

(a) Costs are in early-1981 dollars and include 25% contingency.
(b) Accident cleanup in the radwaste building is not postulated following the scenario 1 and scenario 2 accidents.
(c) Includes the costs of cleanup worker labor, waste management, and 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.

TABLE K.4-2. Summary of Estimated Costs of Preparations for Accident Cleanup Following the Reference BWR Accidents

Cost Category	Preparations for Cleanup Following Scenario 1 Accident			Preparations for Cleanup Following Scenario 2 Accident			Preparations for Cleanup Following Scenario 3 Accident		
	Estimated Costs[a,b] ($ millions)		Percent of Total	Estimated Costs[a,c] ($ millions)		Percent of Total	Estimated Costs[a,d] ($ millions)		Percent of Total
Utility Staff Labor		12.959	53.7		19.505	49.0		35.017	48.6
Waste Management		0.150	0.6		0.355	0.9		0.455	0.6
Energy		3.876	16.1		5.168	13.0		7.752	10.7
Special Equipment and Facilities[e]									
Demineralizer System	1.000			1.000			1.000		
Fuel Racks for Canistered Fuel	--			0.620			0.620		
Processed Water Storage Tanks	0.405			0.405			0.540		
Facilities for Interim Storage of Wastes[f]	0.254			0.434			0.914		
Mockup of Reactor Vessel	--			1.000			3.000		
Total Equipment and Facilities Costs		1.659	6.9		3.459	8.7		6.074	8.4
Miscellaneous Supplies		0.075	0.3		0.100	0.3		0.150	0.2
Specialty Contractors									
Engineering	3.000			8.000			18.000		
Environmental Surveillance	0.063			0.085			0.127		
Laundry	0.075			0.100			0.150		
Total Specialty Contractor Costs		3.138	13.0		8.185	20.6		18.277	25.3
Nuclear Insurance and License Fees		2.257	9.4		3.001	7.5		4.488	6.2
Subtotals		24.114	100.0		39.773	100.0		72.213	100.0
Contingency (25%)		6.029			9.943			18.053	
Total Costs		30.143			49.716			90.266	

(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 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.

are utility staff labor costs. Total labor costs, including the cost of contractor labor for engineering support as well as utility staff labor, are about 70% of the total cost of preparations for accident cleanup. The total cost of planning and preparation is expected to vary approximately linearly with the time required to complete the planning phase following a particular accident. (For example, if preparations for cleanup following the scenario 2 accident were to require 3 years instead of 2, the estimated costs would be approximately $75 million.)

The accumulated spent fuel present in the spent fuel storage pool at the time of an accident is assumed to be transported to an ISFSI for interim storage. The costs of transportation and 10-year storage of this fuel (based on 1-1/3 fuel cores being present in the pool at the time of an accident) are estimated to total about $22 million. This cost is assumed to be an operating cost rather than an accident cleanup cost and is not shown in Table K.4-2.

K.4.1.1 Cost of Labor for Preparations for Accident Cleanup

The costs of utility staff labor for preparations for accident cleanup are shown in detail in Table K.4-3. These costs are based on the utility staff labor requirements shown in Table K.3-11. Utility staff labor costs for preparations for accident cleanup are estimated to be about $13.0 million following the scenario 1 accident, about $19.5 million following the scenario 2 accident, and about $35.0 million following the scenario 3 accident.

Contractor labor costs to provide engineering support during preparations for accident cleanup are not included in Table K.4-3. These contractor labor costs are shown as a separate line item under specialty contractor costs in Table K.4-2. Engineering support staff costs are estimated by postulating a support staff size and using a charge-out rate of $100,000 per man-year. Engineering support staff costs are estimated to be $3 million following the scenario 1 accident, $8 million following the scenario 2 accident, and $18 million following the scenario 3 accident.

K.4.1.2 Cost of Waste Management

The cost of waste management is expected to be small during preparations for accident cleanup. Wastes generated during this period consist mostly of

TABLE K.4-3. Estimated Costs of Utility Staff Labor for Preparations for Accident Cleanup Following the Reference BWR Accidents

Staff Position	Annual Cost per Person[a] ($ thousands)	Scenario 1 Accident: Labor Requirement[b] (man-years)	Scenario 1 Accident: Labor Cost[c] ($ thousands)	Scenario 2 Accident: Labor Requirement[b] (man-years)	Scenario 2 Accident: Labor Cost[c] ($ thousands)	Scenario 3 Accident: Labor Requirement[b] (man-years)	Scenario 3 Accident: Labor Cost[c] ($ thousands)
Plant Superintendent	89.4	1.5	134.1	2.0	178.8	3.0	268.2
Assistant Plant Superintendent	76.2	1.5	114.3	2.0	152.4	3.0	228.6
Consultants	100.0	6.0	600.0	12.0	1 200.0	30.0	3 000.0
Secretaries and Word Processors	24.4	12.0	292.8	20.0	488.0	36.0	878.4
Site Support Staff							
Health and Safety Supervisor	60.5	1.5	90.8	2.0	121.0	3.0	181.5
Health Physicist	47.3	1.5	71.0	2.0	94.6	3.0	141.9
Senior Health Physics Technician	39.5	12.0	474.0	16.0	632.0	36.0	1 422.0
Health Physics Technician	30.1	24.0	722.4	32.0	963.2	72.0	2 167.2
Protective Equipment Attendant	27.8	6.0	166.8	16.0	444.8	24.0	667.2
Industrial Safety Specialist	52.6	1.5	78.9	2.0	105.2	3.0	157.8
Industrial Safety Technician	30.1	3.0	90.3	4.0	120.4	6.0	180.6
Security Supervisor	55.9	1.5	83.8	2.0	111.8	3.0	167.7
Security Shift Supervisor	36.8	6.0	220.8	8.0	294.4	12.0	441.6
Security Patrolman	25.6	72.0	1 843.2	96.0	2 457.6	144.0	3 686.4
Contracts and Accounting Supervisor	47.1	1.5	70.6	2.0	94.2	3.0	141.3
Accountant	39.3	1.5	59.0	2.0	78.6	6.0	235.8
Contracts Specialist	39.3	1.5	59.0	2.0	78.6	3.0	117.9
Insurance Specialist	39.3	1.5	59.0	2.0	78.6	3.0	117.9
Procurement Specialist	39.3	1.5	59.0	2.0	78.6	3.0	117.9
Clerk	24.4	3.0	73.2	8.0	195.2	12.0	292.8
Quality Assurance Supervisor	52.6	1.5	78.9	2.0	105.2	3.0	157.8
Quality Assurance Engineer	47.3	1.5	71.0	4.0	189.2	6.0	283.8
Quality Assurance Technician	27.8	1.5	41.7	4.0	111.2	6.0	166.8
Subtotals		144.0	4 413.4	208.0	6 354.4	351.0	10 845.9
Plant Operations Staff							
Plant Operations Supervisor	61.2	1.5	91.8	2.0	122.4	3.0	183.6
Plant Chemist	52.4	1.5	78.6	2.0	104.8	3.0	157.2
Chemist	46.9	3.0	140.7	4.0	187.6	6.0	281.4
Reactor Operations Engineer	52.4	1.5	78.6	2.0	104.8	3.0	157.2
Engineer	46.9	3.0	140.7	4.0	187.6	6.0	281.4
Reactor Operations Shift Supervisor	52.4	6.0	314.4	8.0	419.2	12.0	628.8
Senior Reactor Operator	46.9	12.0	562.8	16.0	750.4	24.0	1 125.6
Reactor Operator	34.8	24.0	835.2	32.0	1 113.6	48.0	1 670.4
Utility Operator	32.5	24.0	780.0	32.0	1 040.0	48.0	1 560.0
Technician	30.9	18.0	556.2	24.0	741.6	36.0	1 112.4
Craft Supervisor	47.3	1.5	71.0	2.0	94.6	3.0	141.9
Crew Foreman	44.8	6.0	268.8	8.0	358.4	12.0	537.6
Maintenance Mechanic	32.5	24.0	780.0	32.0	1 040.0	48.0	1 560.0
Instrument Technician	32.5	24.0	780.0	32.0	1 040.0	48.0	1 560.0
Warehouseman	27.8	6.0	166.8	8.0	222.4	12.0	333.6
Tool Crib Attendant	27.8	6.0	166.8	8.0	222.4	12.0	333.6
Subtotals		162.0	5 812.4	216.0	7 749.8	324.0	11 624.7
Cleanup Planning Staff							
Cleanup Planning Supervisor	61.2	1.5	91.8	2.0	122.4	3.0	183.6
Engineering Supervisor	52.4	1.5	78.6	2.0	104.8	6.0	314.4
Engineer	46.9	9.0	422.1	16.0	750.4	36.0	1 688.4
Estimator	46.9	1.5	70.4	4.0	187.6	12.0	562.8
Draftsman	30.0	3.0	90.0	8.0	240.0	18.0	540.0
Crew Leader	44.8	1.5	67.2	4.0	179.2	6.0	268.8
Utility Operator	32.5	6.0	195.0	16.0	520.0	48.0	1 560.0
Craftsman	32.5	12.0	390.0	24.0	780.0	48.0	1 560.0
Laborer	31.1	6.0	186.6	16.0	497.6	48.0	1 492.8
Subtotals		42.0	1 591.7	92.0	3 382.0	225.0	8 170.8
Totals		369	12 958.7	552	19 505.4	972	35 016.6

(a) From Table I.1-1 of Appendix I.
(b) From Table K.3-2.
(c) Number of figures shown is for computational accuracy and does not imply precision to the nearest hundred dollars.

compactible and combustible solids (e.g., disposable clothing, rags, plastic covers, laydown pads, and miscellaneous trash) as well as some filters and ion exchange materials. The generation rate for these wastes during preparations for accident cleanup is expected to be similar to the generation rate during normal reactor operations.

Fuel racks are removed from the spent fuel pool during preparations for accident cleanup to provide space in the pool for the filter/demineralizer system used to process accident water and for new fuel racks to accommodate canistered fuel. Only a few of the racks are postulated to be removed following the scenario 1 accident, but all of the racks are removed following the scenario 2 and scenario 3 accidents. The costs of packaging, transportation, and disposal of the old fuel racks at a shallow-land burial ground are given in Table K.4-4. These costs are included as part of the waste management costs for preparations for accident cleanup.

TABLE K.4-4. Estimated Waste Management Costs for the Disposal of BWR Spent Fuel Racks[(a,b)]

Item	Value
Burial Volume (m^3)	350
Estimated Radioactivity Content (Ci)	3.5
Type of Disposable Container	Plywood Box
Number of Disposable Containers[(c)]	15
Number of Waste Shipments	5
Disposable Container Cost ($)	30 000
Transportation Cost ($)	11 810
Shallow-Land Burial Costs ($)	
Disposal Charge	107 520
State Surcharge	3 710
Handling Surcharge	1 870
Total Waste Management Costs ($)	154 910

(a) Numbers of significant figures shown are for computational accuracy only.
(b) Costs are in early-1981 dollars.
(c) Assumes racks are packaged without sectioning.

K.4.1.3 Cost of Energy

Energy costs include the costs of electricity and fuel oil. On the basis of information from Table I.3-8 of Reference 1, the annual rates of consumption of these commodities during cold shutdown of the reference BWR are estimated to be about 40,000 MWh of electricity and about 6,000 m^3 of fuel oil. These usage rates form the bases for computing energy costs during preparations for accident cleanup.

K.4.1.4 Cost of Special Equipment and Facilities

Special equipment and facility items that are needed for accident cleanup in the reactor building and the containment vessel include:

- a 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
- a reactor vessel mockup (scenario 2 and scenario 3 accidents).

These items are designed, fabricated, and installed during preparations for cleanup, and their costs are shown in Table K.4-2. The bases for these costs are described in Section F.1.4 of Appendix F.

K.4.1.5 Cost 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.

K.4.1.6 Cost of Nuclear Insurance and License Fees

The same bases used to estimate the costs of nuclear insurance and license fees for preparations for PWR accident cleanup are used to estimate these costs for preparations for BWR accident cleanup. These costs are discussed in Section F.1.7 of Appendix F. The costs of nuclear insurance and license fees for preparations for BWR accident cleanup are estimated to be about $2.2 million following the scenario 1 accident, about $3.0 million following the scenario 2 accident, and about $4.5 million following the scenario 3 accident.

K.4.2 Details of Costs of Accident Cleanup in the Radwaste Building

The estimated costs of accident cleanup in the radwaste building following the scenario 3 accident are summarized in Table K.4-5. Accident cleanup in the radwaste building is postulated to take place during preparations for cleanup in the reactor building, and is estimated to require 1.5 years and to cost approximately $13 million.

TABLE K.4-5. Summary of Estimated Costs of Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident[a]

Cost Category		Estimated Costs[b] ($ millions)	Percent of Total
Cleanup Worker Labor		6.432	61.3
Waste Management		0.804	7.7
Special Tools and Equipment[c]		1.200	11.4
Miscellaneous Supplies		0.875	8.3
Specialty Contractors			
Engineering	1.000		
Laundry	0.190		
Total Specialty Contractor Costs		1.190	11.3
Subtotal		10.501	100.0
Contingency (25%)		2.625	
Total Costs		13.126	

(a) Accident cleanup in the radwaste building is assumed to be accomplished during preparations for accident cleanup in the reactor building. Management and support staff costs and incidental costs are included in the costs of preparations for accident cleanup.
(b) Costs are in early-1981 dollars. Number of significant figures is for computational accuracy only.
(c) Includes cost of design and installation of system to process contaminated radwaste system liquids.

Costs shown in Table K.4-5 include cleanup worker labor costs, waste management costs, costs of equipment and supplies, and specialty contractor costs specifically related to 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 such as energy costs, environmental surveillance costs, and insurance costs are included with the costs of preparations for cleanup following the scenario 3 accident shown in Table K.4-2.

K.4.2.1 Cost of Labor for Accident Cleanup in the Radwaste Building

Cleanup worker labor costs for accident cleanup in the radwaste building following the scenario 3 accident are shown in Table K.4-6. These costs are estimated to be about $6.4 million based on cleanup worker requirements described in Section K.3.2.

TABLE K.4-6. Estimated Cleanup Worker Labor Costs for Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident

Worker Category	Annual Cost per Person[a] ($ thousands)	Worker Requirements[b] (man-years)	Labor Cost[c] ($ thousands)
Cleanup Operations Supervisor	61.2	1.5	91.8
Crew Leader	44.8	28.1	1258.9
Utility Operator	32.5	42.5	1381.2
Laborer	31.1	43.5	1352.8
Craftsman	32.5	46.2	1501.5
Health Physics Technician	30.1	28.1	845.8
Totals		189.9	6432.0

(a) From Table I.1-1 of Appendix I.
(b) From Table K.3-4.
(c) Number of figures shown is for computational accuracy and does not imply precision to the nearest hundred dollars.

Contractor labor costs to provide engineering support for accident cleanup activities in the radwaste building are not shown in Table K.4-6. These engineering support staff costs are estimated to be about $1 million and are shown as a line item under specialty contractor costs in Table K.4-5.

K.4.2.2 Cost of Waste Management

Based on the waste management disposal assumptions discussed in Section 10.4.1.5 of Chapter 10, costs of radioactive waste management for accident cleanup in the radwaste building are estimated to be about $0.8 million. These costs are shown in detail in Table K.4-7. As discussed in Section 10.4.1.5, all wastes from accident cleanup except the high-activity wastes (filter cartridges and ion exchange materials) from processing contaminated water are transported by truck to a shallow-land burial ground for disposal. The high-activity wastes are placed in temporary shielded storage at the site and are ultimately transported in shielded containers to a federal repository. Both the shallow-land burial ground and the federal repository are assumed to be located 1600 km from the reactor site.

K.4.2.3 Cost of Special Tools and Equipment

The estimated costs of the special tools and equipment used during accident cleanup in the radwaste building total about $1.2 million. This includes an estimated $1 million for the design and installation of a demineralizer to process contaminated radwaste system liquids.

K.4.2.4 Cost of Miscellaneous Supplies

Expendable supplies include decontamination chemicals, ion exchange resins, glass fiber and HEPA filters, cartridge-type filters, disposable protective clothing, assorted cleanup agents, rags, mops, plastic bags and sheeting, and expendable tools. The estimated cost of these miscellaneous supplies for accident cleanup in the radwaste building is about $0.9 million.

K.4.3 Details of Costs of Accident Cleanup in the Reactor Building and Containment Vessel

The estimated costs of accident cleanup in the reactor building and containment vessel following the postulated BWR accidents are summarized in Table K.4-8. Accident cleanup in the reactor building and containment vessel is estimated to require 1.7 years and cost $98 million following the scenario 1 accident, to require 3.3 years and cost $178 million following the scenario 2 accident, and to require 5.3 years and cost $318 million following the scenario 3 accident.

Waste Category	Burial Volume (m^3)	Esti[...] Radioa[...] Con[...] ([...]	Site Costs[e,f]: [...]nd Costs ($) Liner Surcharge	Curie Surcharge	Federal Repository Costs ($)	Total Waste Management Costs ($)
Sludge	2					9 100
Process Solids						
Filter Cartridges	0.3				2 500	5 905
Zeolite Liners	0.9	20 (			7 500	19 760
Organic Resin Liners	0.3				2 500	5 905
Chemical Decontamination Solutions	200					116 820
Trash						
Compactible, Combustible	52					31 670
Compactible, Noncombustible	171					100 460
Noncompactible	434					245 850
Contaminated Equipment						
LSA Materials	150					84 840
High-Activity Materials	50		7 400			183 690
Subtotals						
Wastes Sent to Shallow-Land Burial	1 059		7 400	0		772 430
Wastes Sent to Federal Repository	2	20			12 500	31 570
Totals	1 061	20	7 400	0	12 500	804 000

(a) Numbers of significant figures shown are for compu[...]
(b) Based on information from Table I.2-1 of Appendix [...]
(c) Based on information from Table I.2-2 of Appendix [...]
(d) Based on information from Table I.3-4 of Appendix [...]
(e) Charges are computed on the assumption that all sh[...]
vary depending on the specific physical and radiol[...]
appropriate for computing total charges.
(f) Based on information from Table I.4-1 of Appendix [...]

TABLE K.4-7. Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Radwaste Building Following the Postulated BWR Scenario 3 Accident[a]

TABLE K.4-8. Summary of Estimated Costs of Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Accidents

Cost Category	Accident Cleanup Following Scenario 1 Accident: Estimated Costs[a] ($ millions)		Percent of Total	Accident Cleanup Following Scenario 2 Accident: Estimated Costs[a] ($ millions)		Percent of Total	Accident Cleanup Following Scenario 3 Accident: Estimated Costs[a] ($ millions)		Percent of Total
Utility Staff Labor									
Management and Support Staff	6.663			15.313			32.673		
Plant Operations Staff	5.471			12.192			20.236		
Accident Cleanup Staff	10.630			28.419			73.604		
Per Diem During Defueling[b]	0.744			2.074			7.128		
Total Staff Labor Costs		23.508	29.9		57.998	40.6		133.641	52.6
Waste Management Costs									
Disposal by Shallow-Land Burial	1.549			2.795			6.779		
Disposal at Federal Repository	0.544			1.843			4.178		
Fuel and Fuel Core Debris	37.906			44.428			44.832		
Total Waste Management Costs		39.999	50.8		49.066	34.4		55.789	22.0
Energy		4.962	6.3		9.614	6.7		15.018	5.9
Special Tools and Equipment		3.025	3.8		6.250	4.4		13.650	5.4
Miscellaneous Supplies		2.057	2.6		7.128	5.0		10.313	4.1
Specialty Contractors									
Engineering	1.700			6.600			15.900		
Environmental Surveillance	0.072			0.140			0.224		
Waste Evaporator System	0.050			0.150			0.200		
Laundry	0.310			0.591			1.093		
Total Specialty Contractor Costs		2.132	2.7		7.481	5.2		17.417	6.9
Nuclear Insurance and License Fees		3.000	3.8		5.232	3.7		8.207	3.2
Subtotals		78.683	99.9[c]		142.769	100.0		254.035	100.1[c]
Contingency (25%)		19.671			35.692			63.509	
Total Costs		98.354			178.461			317.544	

(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.

K.4.3.1 Cost of Labor for Accident Cleanup in the Reactor Building and Containment Vessel

Labor costs are a major cost item for reactor building and containment vessel cleanup. Utility staff labor costs account for about 30 to 50% of the total accident cleanup costs, depending on accident scenario. Contractor costs for engineering support contribute an additional 3 to 7% to the total accident cleanup costs.

Details of estimated utility staff labor costs for accident cleanup in the reactor building and the containment vessel are shown in Table K.4-9. These costs are based on the utility staff labor requirements described in Section K.3.3. An additional labor cost included in Table K.4-8 but not shown in Table K.4-9 is the living allowance paid to crew leaders and utility operators brought from other plants to assist in reactor defueling operations. As explained in Section F.3.1 of Appendix F, personnel on temporary assignment are assumed to be paid a living allowance of $2000 per month in addition to their regular salaries.

Costs of contractor labor to provide engineering support for accident cleanup in the reactor building and containment vessel are not shown in Table K.4-9. These costs are shown as a line item under specialty contractor costs in Table K.4-8. Engineering support staff costs are estimated to be $1.7 million for accident cleanup following the scenario 1 accident, $6.6 million for accident cleanup following the scenario 2 accident, and $15.9 million for accident cleanup following the scenario 3 accident. The bases for estimating these costs are given in Section K.4.1.1.

K.4.3.2 Cost of Waste Management

Based on the waste management disposal assumptions discussed in Section 10.4.1.5 of Chapter 10, estimated costs of radioactive waste management for accident cleanup in the reactor building and containment vessel are shown in detail in Tables K.4-10, K.4-11, and K.4-12 for the three BWR accident scenarios. The costs shown in these tables include container costs, transportation, and disposal costs. Labor costs for packaging the wastes prior to shipment are included in the utility staff labor costs shown in

TABLE K.4-9. Estimated Costs of Utility Staff Labor for Accident Cleanup in the Reactor Building Following the Postulated BWR Accidents

Staff Position	Annual Cost per Person[a] ($ thousands)	Scenario 1 Accident Labor Requirement[b] (man-years)	Scenario 1 Accident Labor Cost[c] ($ thousands)	Scenario 2 Accident Labor Requirement[b] (man-years)	Scenario 2 Accident Labor Cost[c] ($ thousands)	Scenario 3 Accident Labor Requirement[b] (man-years)	Scenario 3 Accident Labor Cost[c] ($ thousands)
Plant Superintendent	89.4	1.7	152.0	3.3	295.0	5.3	473.8
Assistant Plant Superintendent	76.2	1.7	129.5	3.3	251.5	5.3	403.9
Consultants	100.0	5.1	510.0	19.8	1 980.0	53.0	5 300.0
Secretaries and Word Processors	24.4	13.6	331.8	33.0	805.2	106.0	2 586.4
Site Support Staff							
Health and Safety Supervisor	60.5	1.7	102.8	3.3	199.6	5.3	320.6
Health Physicist	47.3	1.7	80.4	3.3	156.1	5.3	250.7
Senior Health Physics Technician	39.5	13.6	537.2	26.4	1 042.8	63.6	2 512.2
Health Physics Technician	30.1	13.6	409.4	26.4	794.6	63.6	1 914.4
Protective Equipment Attendant	27.8	6.8	189.0	26.4	733.9	63.6	1 768.1
Industrial Safety Specialist	52.6	1.7	89.4	3.3	173.6	5.3	278.8
Industrial Safety Technician	30.1	3.4	102.3	6.6	198.7	10.6	319.1
Security Supervisor	55.9	1.7	95.0	3.3	184.5	5.3	296.3
Security Shift Supervisor	36.8	6.8	250.2	13.2	485.8	21.2	780.2
Security Patrolman	25.6	81.6	2 089.0	158.4	4 055.0	254.4	6 512.6
Contracts and Accounting Supervisor	47.1	1.7	80.1	3.3	155.4	5.3	249.6
Accountant	39.3	1.7	66.8	3.3	129.7	10.6	416.6
Contracts Specialist	39.3	1.7	66.8	3.3	129.7	5.3	208.3
Insurance Specialist	39.3	1.7	66.8	3.3	129.7	10.6	416.6
Procurement Specialist	39.3	1.7	66.8	3.3	129.7	5.3	208.3
Clerk	24.4	3.4	83.0	13.2	322.1	31.8	775.9
Quality Assurance Supervisor	52.6	1.7	89.4	3.3	173.6	5.3	278.8
Quality Assurance Engineer	47.3	3.4	160.8	6.6	312.2	10.6	501.4
Quality Assurance Technician	27.8	3.4	94.5	6.6	183.5	10.6	294.7
Construction Engineering Supervisor	61.2	1.7	104.0	3.3	202.0	5.3	324.4
Engineer	52.4	10.2	534.5	26.4	1 383.4	63.6	3 332.6
Estimator	46.9	1.7	79.7	6.6	309.5	21.2	994.3
Draftsman	30.0	3.4	102.0	13.2	396.0	31.8	954.0
Subtotals		170.0	5 539.9	366.3	11 981.1	715.5	23 908.5
Plant Operations Staff							
Plant Operations Supervisor	61.2	1.7	104.0	3.3	202.0	5.3	324.4
Plant Chemist	52.4	1.7	89.1	3.3	172.9	5.3	277.7
Chemist	46.9	3.4	159.5	6.6	309.5	10.6	497.1
Reactor Operations Engineer	52.4	1.7	89.1	3.3	172.9	5.3	277.7
Engineer	46.9	3.4	159.5	6.6	309.5	10.6	497.1
Reactor Operations Shift Supervisor	52.4	6.8	356.3	13.2	691.7	21.2	1 110.9
Senior Reactor Operator	46.9	13.6	637.8	26.4	1 238.2	42.4	1 988.6
Reactor Operator	34.8	27.2	946.6	52.8	1 837.4	84.8	2 951.0
Utility Operator	32.5	27.2	884.0	52.8	1 716.0	84.8	2 756.0
Technician	30.9	27.2	840.5	66.0	2 039.4	127.2	3 930.5
Craft Supervisor	47.3	1.7	80.4	3.3	156.1	5.3	250.7
Crew Foreman	44.8	6.8	304.6	13.2	591.4	21.2	949.8
Craftsman	32.5	13.6	442.0	39.6	1 287.0	63.6	2 067.0
Warehouseman	27.8	6.8	189.0	26.4	733.9	42.4	1 178.7
Tool Crib Attendant	27.8	6.8	189.0	26.4	733.9	42.4	1 178.7
Subtotals		149.6	5 471.4	343.2	12 191.8	572.4	20 235.9
Accident Cleanup Staff							
Cleanup Superintendent	61.2	1.7	104.0	3.3	202.0	5.3	324.4
Radioactive Shipment Specialist	39.3	1.7	66.8	3.3	129.7	5.3	208.3
Clerk	24.4	1.7	41.5	3.3	80.5	10.6	258.6
Shift Supervisor	52.4	6.8	356.3	13.2	691.7	21.2	1 110.9
Crew Leader	44.8	30.5	1 366.4	91.1	4 081.3	227.0	10 169.6
Utility Operator	32.5	110.8	3 601.0	266.9	8 674.2	679.0	22 067.5
Laborer	31.1	62.0	1 928.2	153.3	4 767.6	374.9	11 659.4
Craftsman	32.5	61.2	1 989.0	199.5	6 483.8	585.2	19 019.0
Health Physics Technician	30.1	39.1	1 176.9	109.9	3 308.0	291.9	8 786.2
Subtotals		315.5	10 630.1	843.8	28 418.8	2 200.4	73 603.9
Totals		657.2	22 764.7	1 612.7	55 923.4	3 657.9	126 512.4

(a) From Table I.1-1 of Appendix I.
(b) From Table K.3-11.
(c) Number of figures shown is for computational accuracy and does not imply precision to the nearest hundred dollars.

Waste Category	Burial Volume (m^3)	Est Radio Con (	Site Costs[e,f] nd Costs ($) Liner Surcharge	Curie Surcharge	Federal Repository Costs ($)	Total Waste Management Costs ($)
Sludge	2					7 020
Process Solids						
Filter Cartridges	1				7 500	17 710
Zeolite Liners	2	30			17 500	43 370
Organic Resin Liners	1				7 500	17 710
Process Solids						
Filter Cartridges	1					740
Evaporator Bottoms	74	20			260 000	465 530
Organic Resins	4					16 590
Chemical Decontamination Solutions	79					45 650
Trash						
Compactible, Combustible	90					56 270
Compactible, Noncombustible	287					169 300
Noncompactible	725					410 420
Contaminated Equipment						
LSA Materials	14					8 960
High-Activity Materials	168		24 250			603 390
Irradiated Hardware						
LSA Materials	28					40 810
High-Activity Materials	34	40	60 000	11 520		189 250
Fuel Assemblies						
Intact Assemblies	78				30 272 000	33 480 250
Damaged Assemblies	12				3 344 000	4 426 000
Fuel Core Debris	--					
Subtotals						
Wastes Sent to Shallow-Land Burial	1 432	40	84 250	11 520		1 548 400
Wastes Sent to Federal Repository	78	50			292 500	544 320
Reactor Fuel and Fuel Core Debris	90				33 616 000	37 906 250
Totals	1 600		84 250	11 520	33 908 500	39 998 970

(a) Numbers of significant figures shown are for com
(b) Based on information from Table I.2-1 of Appendi
(c) Based on information from Table I.2-2 of Appendi
(d) Based on information from Table I.3-4 of Appendi
(e) Charges are computed on the assumption that all
vary depending on the specific physical and radi
appropriate for computing total charges.
(f) Based on information from Table I.4-1 of Appendi

TABLE K.4-10. Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Scenario 1 Accident[a]

Waste Category	Burial Volume (m^3)	Est Radio Con (	Site Costs(e,f) nd Costs ($) Liner Surcharge	Curie Surcharge	Federal Repository Costs ($)	Total Waste Management Costs ($)
Sludge	5					19 560
Process Solids						
Filter Cartridges	3				20 000	47 220
Zeolite Liners	6	432			50 000	118 050
Organic Resin Liners	4	2			30 000	70 830
Process Solids						
Filter Cartridges	1			660		3 200
Evaporator Bottoms	148	280				
					520 000	1 607 060
Organic Resins	51					167 990
Chemical Decontamination Solutions	394					230 980
Trash						
Compactible, Combustible	168					103 810
Compactible, Noncombustible	542					317 730
Noncompactible	1 369					774 800
Contaminated Equipment						
LSA Materials	60					34 600
High-Activity Materials	205		42 770			749 510
Irradiated Hardware						
LSA Materials	28					40 810
High-Activity Materials	34	120				
			60 000	18 720		352 450
Fuel Assemblies						
Intact Assemblies						
Damaged Assemblies	115				33 616 000	44 225 250
Fuel Core Debris	1				100 000	202 250
Subtotals						
Wastes Sent to Shallow-Land Burial	2 857	121	102 770	19 380		2 795 440
Wastes Sent to Federal Repository	161	714			620 000	1 843 160
Reactor Fuel and Fuel Core Debris	116				33 716 000	44 427 500
Totals	3 134		102 770	19 380	34 336 000	49 066 100

(a) Numbers of significant figures shown are for com
(b) Based on information from Table I.2-1 of Appendi
(c) Based on information from Table I.2-2 of Appendi
(d) Based on information from Table I.3-4 of Appendi
(e) Charges are computed on the assumption that all
vary depending on the specific physical and radi
appropriate for computing total charges.
(f) Based on information from Table I.4-1 of Appendi

TABLE K.4-11. Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Scenario 2 Accident(a)

Waste Category	Burial Volume (m³)	Esti Radioa Cont (C	Liner Surcharge	Curie Surcharge	Federal Repository Costs ($)	Total Waste Management Costs ($)
			;ite Costs(e,f)			
			ıd Costs ($)			
Sludge	9			2 070		33 780
Process Solids						
Filter Cartridges	11				87 500	208 640
Zeolite Liners	20	3 00			165 000	389 560
Organic Resin Liners	19	1			155 000	365 960
Process Solids						
Filter Cartridges	2			840		12 780
Evaporator Bottoms	296	2 00			1 040 000	3 214 120
Organic Resins	357					1 152 500
Chemical Decontamination Solutions	1 050					613 660
Trash						
Compactible, Combustible	309					188 170
Compactible, Noncombustible	998					583 440
Noncompactible	2 520					1 422 180
Contaminated Equipment						
LSA Materials	144					82 540
High-Activity Materials	312		82 620			1 153 070
Irradiated Hardware						
LSA Materials	28					40 810
High-Activity Materials	91	6 50	160 000	606 120		1 496 190
Fuel Assemblies						
Intact Assemblies						
Damaged Assemblies	115				33 616 000	44 225 250
Fuel Core Debris	4				300 000	606 750
Subtotals						
Wastes Sent to Shallow-Land Burial	5 820	6 5(	242 620	609 030		6 779 120
Wastes Sent to Federal Repository	346	5 01			1 447 500	4 178 280
Reactor Fuel and Fuel Core Debris	119				33 916 000	44 832 000
Totals	6 285		242 620	609 030	35 363 500	55 789 400

(a) Numbers of significant figures shown are for compu
(b) Based on information from Table I.2-1 of Appendix
(c) Based on information from Table I.2-2 of Appendix
(d) Based on information from Table I.3-4 of Appendix
(e) Charges are computed on the assumption that all sł vary depending on the specific physical and radiol appropriate for computing total charges.
(f) Based on information from Table I.4-1 of Appendix

ıBLE K.4-12. Estimated Costs of Radioactive Waste Management for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Scenario 3 Accident[a]

Table K.4-9. Labor costs for transportation and disposal are included in the total charges for these activities shown in the tables. Waste management costs range from 51% of the total cost of accident cleanup following the scenario 1 accident to 22% of the total cost of accident cleanup following the scenario 3 accident. The cost of disposal of the fuel from defueling the reactor accounts for most of the cost of waste management.

As discussed in Section 10.4.1.5, high-activity wastes (filter cartridges, ion exchange resin liners, and evaporator bottoms from processing radioactive liquids) and damaged fuel assemblies are assumed to be transported to a federal repository. Fuel assemblies that are not damaged are transported to an ISFSI. All other radioactive wastes are 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. Although the great majority of the waste (by volume) is shipped to a shallow-land burial ground, most of the costs of waste management is for the packaging, transportation, and disposal of wastes shipped to a federal repository.

K.4.3.3 Cost of Energy

Energy costs represent about 6% of the total cost of reactor building and containment vessel cleanup following the postulated accidents. The following bases and assumptions are used to calculate energy costs:

1) Energy consumption during cold shutdown of the reference BWR is estimated to be about 40,000 MWh of electricity and about 6,000 m^3 of fuel oil annually (see Section K.4.1.3).
2) Use of plant pumps during chemical decontamination adds about 18 MW to the base electrical load while the pumps are running.
3) Operation of the demineralizer system adds about 0.5 MW to the base electrical load while the system is operating.

K.4.3.4 Cost of Special Tools and Equipment

The costs of special tools and equipment for accident cleanup in the reactor building and containment vessel of the reference BWR are estimated to be about the same as those costs for accident cleanup in the containment building of the reference PWR. PWR costs are shown in Table F.3-6 of Appendix F. The estimated costs of special tools and equipment for accident cleanup are about $3 million following the scenario 1 accident, about $6 million following the scenario 2 accident, and about $14 million following the scenario 3 accident. These costs do not include the costs of the special facilities constructed during preparations for accident cleanup and shown as line items in Table K.4-2.

K.4.3.5 Cost of Miscellaneous Supplies

Expendable supplies for accident cleanup in the reactor building and containment vessel of the reference BWR include decontamination chemicals, protective clothing, filters and ion exchange resins, mechanical and electrical supplies, cleaning supplies, and expendable tools. The estimated costs for these items are presented in Table K.4-13. Costs for miscellaneous supplies are estimated to be about $2 million for cleanup following the scenario 1 accident, about $7 million following the scenario 2 accident, and about $10 million following the scenario 3 accident.

K.4.3.6 Cost of Nuclear Insurance and License Fees

The bases used to estimate the costs of nuclear insurance and license fees during accident cleanup are described in Section F.3.7 of Appendix F. For accident cleanup of the reference BWR, these costs are estimated to about $3.0 million following the scenario 1 accident, $5.2 million following the scenario 2 accident, and $8.2 million following the scenario 3 accident.

K.5 DETAILS OF ACTIVITIES AND MANPOWER REQUIREMENTS FOR DECOMMISSIONING

This section provides details of the technical requirements and manpower needs for post-accident decommissioning at the reference BWR following completion of the accident cleanup campaign. The three decommissioning alternatives that are analyzed include:

- DECON - the immediate removal of all radioactive material to permit license termination and unrestricted release of the property

TABLE K.4-13. Estimated Costs of Miscellaneous Supplies for Accident Cleanup in the Reactor Building and the Containment Following the Postulated BWR Accidents

Item	Accident Cleanup Following Scenario 1 Accident		Accident Cleanup Following Scenario 2 Accident		Accident Cleanup Following Scenario 3 Accident	
	Quantity	Total Cost ($ thousands)[a]	Quantity	Total Cost ($ thousands)[a]	Quantity	Total Cost ($ thousands)[a]
Decontamination Chemicals						
EDTA/Oxalic/Citric Acid	31 250 kg	51	31 250 kg	51	62 500 kg	101
OPG Solution			625 m^3 of solution	35	1 250 m^3 of solution	70
Respirator Facepieces	100 each	10	200 each	20	500 each	50
Anticontamination Clothing	15 515 sets[b]	776	29 540 sets[b]	1 477	54 630 sets[b]	2 732
Cleaning Supplies	See Note[c]	255	See Note[c]	495	See Note[c]	795
Expendable Tools	See Note[d]	170	See Note[d]	330	See Note[d]	530
Ion Exchange Resins	15 m^3	75	30 m^3	150	60 m^3	300
Filters	Unspecified	200	Unspecified	400	Unspecified	600
Mechanical Supplies and Hardware	Unspecified	50	Unspecified	100	Unspecified	300
Electrical Components and Cables	Unspecified	25	Unspecified	50	Unspecified	200
Ion Exchange and Filter Liners	13 each	65	40 each	200	163 each	815
Canisters for Damaged Fuel	76 each	380	764 each	3 820	764 each	3 820
Totals		2 057		7 128		10 313

(a) Costs are in early-1981 dollars and are rounded to the nearest $1 000.
(b) Estimated at two clothing changes per shift per worker. One set of clothing can be laundered and used four times.
(c) Estimated at $150 000 per year.
(d) Estimated at $100 000 per year.

- SAFSTOR - preparation and maintenance of the property so that risk to public safety is acceptable for a period of storage until either the facility is decontaminated or the residual radioactivity decays to an unrestricted release level
- ENTOMB - the encasement and maintenance of the property in a strong and structurally long-lived material to ensure retention and isolation from the environment until the contained radioactivity decays to an unrestricted release level.

The BWR post-accident decommissioning analyses in this section use the results of previous analyses of BWR decommissioning following normal shutdown, presented in Reference 1, with appropriate modifications as necessary to account for post-accident conditions. Comparisons of the applicable activities for normal versus post-accident decommissioning of the reference BWR are presented in Tables K.5-1, K.5-2, and K.5-3 for the DECON, SAFSTOR, and ENTOMB alternatives, respectively.

It is assumed in this study that the accident cleanup activities described in Section K.3 are performed prior to the start of decommissioning. In carrying out accident cleanup, some tasks that would be part of normal decommissioning are completed and other tasks are partially completed. Examples of tasks completed during cleanup include defueling of 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 decontamination of building surfaces in the reactor building and the containment vessel and 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 the temporary onsite waste storage structures specially constructed for the interim storage of wastes from accident cleanup activities.

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

TABLE K.5-1. Comparison of Activities for Normal Versus Post-Accident DECON at the Reference BWR

Task	Applicable to:[a] DECON Following Normal Shutdown	Post-Accident Initial Cleanup	Post-Accident DECON
Reactor Building/Primary Containment			
1. Comprehensive radiation survey	X	X	
2. Install HEPA filters	X[b]	X	
3. Initial decontamination of primary containment and reactor building		X	
4. Remove, segment, and package dryer and separator	X	X	
5. Discharge spent fuel	X[c]	X	
6. Ship spent fuel offsite	X		X
7. Remove spent fuel racks	X	P[d]	X
8. Drain and decontaminate suppression pool	X	X	P
9. Remove, segment, and package reactor vessel internals	X	P	P
10. Ship dryer and separator and reactor vessel internals	X		X
11. Drain and decontaminate reactor well pool	X		X
12. Chemical decon reactor water recirculation and cleanup systems	X	X	P
13. Clean up, stage, and shield hot spots in primary containment	X	P	P
14. Enlarge suppression chamber access	X		X
15. Segment, package, and ship reactor vessel	X		X
16. Remove contaminated piping and equipment from primary containment	X		X
17. Remove contaminated structural materials from primary containment	X		X
18. Drain contaminated systems to radwaste	X	P	P
19. Chemical decon RHR, LPCS, and HPCS systems	X	X	P

(contd on next page)

TABLE K.5-1. contd

Task	Applicable to:(a) DECON Following Normal Shutdown	Post-Accident Initial Cleanup	Post-Accident DECON
20. Remove reactor building piping	X		X
21. Drain and decontaminate dryer and separator pool	X		X
22. Chemical decon drain systems	X	P	X
23. Remove submerged demineralizer system			X
24. Drain and decontaminate spent fuel pool and fuel pool cooling and cleanup system	X		X
25. Remove contaminated reactor building equipment	X		X
26. Remove liners from spent fuel pool, reactor well, and dryer and separator pool	X		X
27. Decontaminate reactor building internal surfaces	X	P	X
28. Remove HVAC and electrical systems	X		X
29. Final radiation survey	X		X
Turbine Generator Building			
All tasks	X		X
Radwaste and Control Building			
All tasks	X		X
Ancillaries			
1. Process accident water		X	
2. Packaging and shipment of wastes	X	X	X
3. Remove onsite waste storage structures			X

(a) Applicability indicated by X, partial applicability by P, nonapplicability by a blank; chronological order of tasks may vary from that shown.
(b) Performed as part of planning and preparations.
(c) Considered to be part of normal shutdown procedures.
(d) Sufficient spent fuel racks removed to provide room for new racks to handle damaged fuel.

TABLE K.5-2. Comparison of Activities for Normal Versus Post-Accident SAFSTOR at the Reference BWR

Task	Applicable to:(a) SAFSTOR Following Normal Shutdown	Post-Accident Initial Cleanup	Post-Accident SAFSTOR
Reactor Building/Primary Containment			
1. Comprehensive radiation survey	X	X	
2. Install HEPA filters	X(b)	X	
3. Initial decontamination of primary containment and reactor building		X	
4. Discharge spent fuel	X(c)	X	
5. Ship spent fuel offsite	X		X
6. Remove spent fuel racks		P(d)	
7. Drain and decontaminate suppression pool	X	X	P
8. Remove, segment, and package reactor vessel internals		P(e)	
9. Drain and decontaminate reactor well pool	X		X
10. Chemical decon reactor water recirculation and cleanup systems	X	X	P
11. Clean up, stage, and shield hot spots in primary containment	X	P	P
12. Drain contaminated systems to radwaste	X	P	P
13. Chemical decon RHR, LPCS, and HPCS systems	X	X	P
14. Drain and decontaminate spent fuel pool and fuel pool cooling and cleanup system	X		X
15. Drain and decontaminate dryer and separator pool	X		X
16. Chemical decon drain systems	X	P	X
17. Cover and seal spent fuel pool and dryer and separator storage pool	X		X
18. Seal equipment and personnel hatches into primary containment	X		X

(contd on next page)

TABLE K.5-2. contd

Task	Applicable to:(a) SAFSTOR Following Normal Shutdown	Post-Accident Initial Cleanup	Post-Accident SAFSTOR
19. Decontaminate HVAC, electrical, internal structures, equipment, and concrete; apply protective paint	X	P	X
20. Isolate and seal equipment, piping, rooms, stack HVAC ducts, rail tunnel, and steam tunnel	X		X
21. Seal drywell top head and unneeded reactor building doors	X		X
22. Install HEPA-filtered vents and deactivate unnecessary utilities	X		X
23. Install intrusion, radiation monitoring, and fire alarm systems	X		X
24. Final radiation survey	X		X
Turbine Generator Building			
All tasks	X		X
Radwaste and Control Building			
All tasks	X		X
Ancillaries			
1. Process accident water		X	
2. Packaging and shipment of wastes	X	X	X

(a) Applicability indicated by X, partial applicability by P, nonapplicability by a blank; chronological order of tasks may vary from that shown.
(b) Performed as part of planning and preparations.
(c) Considered to be part of normal shutdown procedures.
(d) Sufficient spent fuel racks removed to provide room for new racks to handle damaged fuel.
(e) Removal of some internals required to remove damaged fuel from the reactor vessel.

TABLE K.5-3. Comparison of Activities for Normal Versus Post-Accident ENTOMB at the Reference BWR

Task	Applicable to:[a] ENTOMB Following Normal Shutdown	Post-Accident Initial Cleanup	Post-Accident ENTOMB
Reactor Building/Primary Containment			
1. Comprehensive radiation survey	X	X	
2. Install HEPA filters	X[b]	X	
3. Initial decontamination of primary containment and reactor building		X	
4. Remove, segment, and package dryer and separator	X	X	
5. Discharge spent fuel	X[c]	X	
6. Ship spent fuel offsite	X		X
7. Remove spent fuel racks	X	P[d]	X
8. Drain and decontaminate suppression pool	X	X	P
9. Remove, segment, and package reactor vessel internals	X	P	P
10. Ship dryer and separator and reactor vessel internals	X		X
11. Cut suppression pool downcomers and bracing	X		X
12. Drain and decontaminate reactor well pool	X		X
13. Chemical decon reactor water recirculation and cleanup systems	X	X	P
14. Clean up, stage, and shield hot spots in primary containment	X	P	P
15. Chemical decon RHR, LPCS, and HPCS systems	X	X	P
16. Drain contaminated systems to radwaste	X	P	P
17. Drain and decontaminate dryer and separator pool	X		X
18. Cut suppression chamber accesses through drywell floor	X		X
19. Chemical decon drain systems	X	P	X
20. Remove submerged demineralizer system			X
21. Cut and seal primary containment piping penetrations	X		X
22. Cut drywell bellows access openings	X		X

(contd on next page)

TABLE K.5-3. contd

Task	Applicable to:(a) ENTOMB Following Normal Shutdown	Post-Accident Initial Cleanup	Post-Accident ENTOMB
23. Remove reactor building piping	X		X
24. Drain and decontaminate spent fuel pool and fuel pool cooling and cleanup system	X		X
25. Remove liners from spent fuel pool and dryer and separator pool	X		X
26. Remove contaminated reactor building equipment	X		X
27. Seal equipment and personnel hatch openings into primary containment	X		X
28. Decontaminate reactor building internal surfaces	X	P	X
29. Seal rail tunnel, steam tunnel, and biological shield penetrations	X		X
30. Seal drywell top head and reactor building external doors	X		X
31. Remove HVAC and disable crane	X		X
32. Final radiation survey	X		X
33. Install security and surveillance monitoring equipment; disconnect unnecessary utilities	X		X
Turbine Generator Building			
All tasks	X		X
Radwaste and Control Building			
All tasks	X		X
Ancillaries			
1. Process accident water		X	
2. Packaging and shipment of wastes	X	X	X

(a) Applicability indicated by X, partial applicability by P, nonapplicability by a blank; chronological order of tasks may vary from that shown.
(b) Performed as part of planning and preparations.
(c) Considered to be part of normal shutdown procedures.
(d) Sufficient spent fuel racks removed to provide room for new racks to handle damaged fuel.

- selecting specialty contractors - The use of specialty contractors allows certain specialized decommissioning tasks outside the expertise or capability of the decommissioning staff to be performed by experts, thereby increasing the overall efficiency and safety of the decommissioning project.
- installation of HEPA filters - Prior to the start of actual decommissioning, HEPA filters are installed outboard of the blowers in the HVAC exhaust systems of the reactor building and the turbine-generator building. (The radwaste building HVAC system is already equipped with HEPA filters; these filters are changed as required.) These filters are installed to prevent the atmospheric release of airborne radioactivity generated during the decommissioning tasks, since many tasks are expected to generate airborne contamination inside the facility exceeding that produced during normal plant operation.

In this study, planning and preparation for post-accident decommissioning is assumed to take place during the final 1.5 years of accident cleanup. Key supervisory personnel (a decommissioning superintendent, decommissioning engineer, and assistant decommissioning engineer) are designated and supervise the planning and preparation activities. Additional personnel to assist in preparations for decommissioning are assigned as required from the accident cleanup staff. (Personnel are available from the accident cleanup staff because of the extra manpower required to maintain compliance with occupational dose limitations.)

K.5.2 Details of DECON at the Reference BWR

The decontamination and dismantlement activities during post-accident DECON at the reference BWR are similar to those during DECON following normal shutdown, which are described in detail in Appendix H and Appendix I of Reference 1. Decontamination and dismantlement activities, schedules, and manpower requirements are summarized here, with emphasis on those activities and requirements that differ significantly from the ones during normal-shutdown DECON.

K.5.2.1 DECON in the Reactor Building and the Containment Vessel

All of the neutron-activated materials and the majority of the radioactive contamination (both accident-generated and from normal operations) in the reference BWR at the time of decommissioning are located in the containment vessel. Additional radioactive contamination (both accident-generated and from normal operations) is present in the reactor building. The neutron-activated components (the reactor vessel internals, reactor vessel, and portions of the sacrificial shield) are segmented and packaged in steel cask liners for shipment to a shallow-land burial ground. Radioactively contaminated materials (equipment items, conduit and piping, structural members, liners, and concrete) are removed and segmented as required for packaging in steel drums and plywood boxes. Methods postulated for removal of these materials during DECON following normal shutdown are described in Appendix G and Appendix I of Reference 1, and these same methods are generally applied during post-accident DECON.

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 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 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 that are postulated for accident cleanup. The level of effort required for this task is a function of the amount of contamination present, which increases with increasing accident severity.

As discussed in Section K.3.3.1, portions of the reactor vessel internals (e.g., the steam dryer, steam separator, and top fuel guide) are postulated to be removed and packaged for disposal during accident cleanup to facilitate defueling of the reactor, thus reducing the level of effort required during post-accident DECON. On the other hand, additional difficulties may be encountered during DECON because of accident-caused damage to the remaining internals and higher radiation exposure rates in the work area.

substantial changes in time and manpower requirements for post-accident decommissioning. Manpower requirements for carrying out specific tasks are related to a number of factors (e.g., the physical condition of the equipment and structures, local radiation dose rates, and the methods used to complete tasks) that may be affected by the accident and the subsequent accident cleanup campaign. Radiation doses to workers during post-accident decommissioning are likely to be higher than those following normal shutdown because of increased contamination on equipment and structural surfaces. Dose rates to decommissioning workers will increase with accident severity even though a substantial amount of decontamination is performed during accident cleanup. In addition, physical damage to the plant from the more severe accidents may compromise certain systems, equipment items, and structural features that are required for the performance of decommissioning tasks, thus necessitating repairs or substitutions and increasing the time and cost of decommissioning.

The post-accident decommissioning analyses in this study are based on the assumption that the reactor has experienced a scenario 2 accident. (BWR accident scenarios are described in Section K.2.) Variations in decommissioning activities and requirements that would result from the other two accident scenarios are discussed where applicable.

Accident-induced radioactive contamination and physical damage resulting from the BWR scenario 2 accident are assumed to be largely confined within the reactor building and the containment vessel. Accident cleanup activities following the scenario 2 accident are centered in these structures. As a result of the accident and the subsequent cleanup activities, the requirements for post-accident decommissioning in the reactor building and the containment vessel will differ in some respects from those postulated in Reference 1 for decommissioning following normal shutdown. These differences in requirements are described in the following subsections. For other plant structures (e.g., the radwaste building and the turbine-generator building) the differences between the requirements for post-accident and normal-shutdown decommissioning are anticipated to be small. Within the limits of accuracy of the manpower and cost estimates for post-accident decommissioning presented in this appendix, the requirements for post-accident decommissioning of these other buildings following a scenario 2 accident are assumed to be the same as the requirements for decommissioning following normal shutdown.

K.5.1 Details of Planning and Preparation Activities

Planning and preparation activities for decommissioning at the reference BWR following normal shutdown are discussed in Section H.2 of Reference 1. Planning and preparation activities applicable to post-accident decommissioning include:

- satisfying regulatory requirements - The major requirements are 1) providing the necessary documentation to amend the facility operating license to "possession only" status and 2) if required, obtaining an NRC dismantling order.
- gathering and analyzing data - These data support the regulatory requirements for decommissioning and provide the bases for planning decommissioning tasks and selecting appropriate methods and equipment.
- developing detailed work plans and procedures - These plans and procedures contain all the information required to actually carry out the decommissioning tasks. They cover all aspects of the project, including quality assurance, security, and environmental constraints.
- designing, procuring, and testing special equipment - Designs and specifications are prepared for each special equipment item required to complete the decommissioning project. When the item is procured, it is 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.
- selecting and training staff - Staffing requirements are identified, key engineering and operating personnel are selected, and personnel are trained as required to fulfill their roles in the organization.

Chemical decontamination of the reactor water recirculation system and of portions of the reactor water cleanup sysem is accomplished during accident cleanup. However, chemical decontamination of other systems such as the residual heat removal system, the core spray systems, the fuel pool cooling and cleanup system, and contaminated drain piping systems is accomplished during post-accident DECON. Methods for chemical decontamination of these systems are described in Appendix H of Reference 1.

Decontamination of building surfaces in the reactor building and the containment vessel is initiated during accident cleanup to reduce radiation doses to 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 and cutting of metal plates) are the same as those employed during DECON following normal reactor shutdown. However, accident-generated contamination results in a greater level of effort and larger volume of radioactive waste material produced during post-accident DECON than is the case for DECON following normal shutdown.

Estimated direct worker requirements for DECON in the reactor building and the containment vessel following a scenario 2 accident are shown in Table K.5-4. The DECON tasks shown in the table are generally the same as those shown in Table I.2-1 of Reference 1 with the deletion of some tasks completed during accident cleanup and the addition of some new tasks that are required because of the accident. Adjustments have been made to the time requirements for completion of certain tasks to account for the effects of the accident and of post-accident cleanup activities. The labor requirements shown in the table include only the labor estimated to be needed to actually complete each of the designated tasks and do not include support staff or the additional personnel required for compliance with occupational dose limitations.

Estimated occupational radiation doses to workers engaged in DECON activities in the reactor building and the containment vessel following the scenario 2 accident are shown in Table K.5-5. The occupational doses shown in the table are external doses from gamma radiation. Workers are assumed to use

TABLE K.5-4. Estimated Direct Worker Requirements for DECON in the Reactor Building and Containment Vessel Following a Scenario 2 Accident

DECON Activity	Time Required (months)	Shift Basis: 2 Shifts 5-day week	Shift Basis: 3 Shifts 7-day week	Crew Requirement per Shift: Crew Leader	Utility Operator	Laborer	Craftsman	H.P. Technician	Labor Requirement (man-months)[a]: Crew Leader	Utility Operator	Laborer	Craftsman	H.P. Technician	Totals
Cleanup and shield access routes and hot spots	6	x		1	2	2	2	1	12	24	24	24	12	96
Remove reactor vessel internals	6-1/2	x			2		2			26		26		52
Ship activated reactor vessel internals	13		x	1	2			1	52	104			52	208
Drain reactor well pool - water jet clean	1		x	1	4			1	4	16			4	24
Clean up and shield hot spots in containment vessel	3	x				2		1			12		6	18
Enlarge suppression chamber access	1/2	x				1	1				1	1		2
Remove reactor vessel	4	x			2		2			16		16		32
Ship activated reactor vessel segments	5		x	1	2			1	20	40			20	80
Remove containment vessel piping and equipment	8	x		1	2	2	2	1	16	32	32	32	16	128
Remove sacrificial shield and radial beams	5	x		1		2	2	1	10		20	20	10	60
Remove contaminated concrete from containment vessel	6	x				2					24			24
Remove HVAC and electrical systems from containment vessel	1	x		1		2	2	1	2		4	4	2	12
Drain contaminated systems to radwaste	1-1/2		x	1	4			1	6	24			6	36
Chemical decon of RHR and core spray systems	1/2		x	1	4			1	2	8			2	12
Remove reactor building piping	10-1/2	x		1	2	2	2	1	21	42	42	42	21	168
Drain dryer and separator pool - water jet clean	1		x	1	4			1	4	16			4	24
Chemical decon of drain systems	1		x	1	4			1	4	16			4	24
Drain spent fuel pool - water jet clean	2		x	1	4			1	8	32			8	48
Chemical decon of fuel pool cooling and cleanup system	1/2		x	1	4			1	2	8			2	12
Remove reactor building equipment	3	x		1	1	3	1	1	6	6	18	6	6	42
Remove liners from various pools	1-1/2	x		1		2	2	1	3		6	6	3	18
Remove reactor building contaminated concrete	3	x				2					12			12
Remove HVAC and electrical systems from reactor building	1	x		1		2	2	1	2		4	4	2	12
Final radiation survey of reactor building and containment vessel	1	x						2					4	4
Totals									174	410	199	181	184	1148

(a) Based on 20 working days per month.

TABLE K.5-5. Estimated Occupational Radiation Doses for DECON in the Reactor Building and Containment Vessel Following a Scenario 2 Accident

DECON Activity	Estimated Average Dose Rate (rem/hr)	Crew Leader		Utility Operator		Laborer		Craftsman		H.P. Technician		Task Totals	
		Exposure[a] (man-hr)	Dose (man-rem)	Exposure[a] (man-hr)	Dose (man-rem)	Exposure[a] (man-hr)	Dose (man-rem)	Exposure[a] (man-hr)	Dose (man-rem)	Exposure[a] (man-hr)	Dose (man-rem)	Exposure[a] (man-hr)	Dose (man-rem)
Cleanup and shield access routes and hot spots	0.015	1 440	21.6	2 880	43.2	2 880	43.2	2 880	43.2	1 440	21.6	11 520	172.8
Remove reactor vessel internals	0.015			3 120	46.8			3 120	46.8			6 240	93.6
Ship activated reactor vessel internals	0.015	6 240	93.6	12 480	187.2					6 240	93.6	24 960	374.4
Drain reactor well pool - water jet clean	0.008	480	3.8	1 920	15.4					480.	3.8	2 880	23.0
Clean up and shield hot spots in containment vessel	0.030					1 440	43.2			720	21.6	2 160	64.8
Enlarge suppression chamber access	0.010					120	1.2	120	1.2			240	2.4
Remove reactor vessel	0.010			1 920	19.2			1 920	19.2			3 840	38.4
Ship activated reactor vessel segments	0.010	2 400	24.0	4 800	48.0					2 400	24.0	9 600	96.0
Remove containment vessel piping and equipment	0.030	1 920	57.6	3 840	115.2	3 840	115.2	3 840	115.2	1 920	57.6	15 360	460.8
Remove sacrificial shield and radial beams	0.025	1 200	30.0			2 400	60.0	2 400	60.0	1 200	30.0	7 200	180.0
Remove contaminated concrete from containment vessel	0.015					2 880	43.2					2 880	43.2
Remove HVAC and electrical systems from containment vessel	0.015	240	3.6			480	7.2	480	7.2	240	3.6	1 440	21.6
Drain contaminated systems to radwaste	0.020	720	14.4	2 880	57.6					720	14.4	4 320	86.4
Chemical decon of RHR and core spray systems	0.020	240	4.8	960	19.2					240	4.8	1 440	28.8
Remove reactor building piping	0.020	2 520	50.4	5 040	100.8	5 040	100.8	5 040	100.8	2 520	50.4	20 160	403.2
Drain dryer and separator pool - water jet clean	0.008	480	3.8	1 920	15.4					480	3.8	2 880	23.0
Chemical decon of drain systems	0.005	480	2.4	1 920	9.6					480	2.4	2 880	14.4
Drain spent fuel pool - water jet clean	0.005	960	4.8	3 840	19.2					960	4.8	5 760	28.8
Chemical decon of fuel pool cooling and cleanup system	0.005	240	1.2	960	4.8					240	1.2	1 440	7.2
Remove reactor building equipment	0.003	720	2.2	720	2.2	2 160	6.5	720	2.2	720	2.2	5 040	15.3
Remove liners from various pools	0.005	360	1.8			720	3.6	720	3.6	360	1.8	2 160	10.8
Remove reactor building contaminated concrete	0.005					1 440	7.2					1 440	7.2
Remove HVAC and electrical systems from reactor building	0.005	240	1.2			480	2.4	480	2.4	240	1.2	1 440	7.2
Final radiation survey of reactor building and containment vessel	0.001									480	0.5	480	0.5
Totals		20 880	321.2	49 200	703.8	23 880	433.7	21 720	401.8	22 080	343.3	137 760	2 203.8

(a) Based on estimated direct worker requirements from Table K.5-4. Assume 20 working days per month; 6 hours of exposure per shift.

respiratory equipment as necessary to protect against the inhalation of airborne radioactive particulates. Dose calculations are based on time and manpower requirements shown in Table K.5-4. Exposure hours are estimated on the basis that workers engaged in DECON operations spend an average of 6 hours in a radiation area during an 8-hour shift.

Because whole-body radiation doses to the decommissioning workers are limited in accordance with the requirements of 10 CFR 20.101, the estimated worker requirements shown in Table K.5-4 must be adjusted upward so that average individual radiation doses do not exceed 5 man-rem/man-year. Adjusted worker requirements for DECON in the reactor building and the containment vessel are shown in Table K.5-6. For DECON in these structures following the scenario 2 accident, estimated manpower requirements must be increased by factors in the range of 4 to 5, resulting in a total worker requirement of about 440 man-years for DECON in the reactor building and the containment vessel following a scenario 2 accident.

For DECON in the reactor building and the containment vessel following a scenario 1 accident, the occupational radiation dose and the adjusted worker requirement are estimated to be about one-half of their values for DECON

TABLE K.5-6. Adjustments to Decommissioning Worker Requirements to Comply with Occupational Radiation Dose Limitations for DECON in the Reactor Building and Containment Vessel Following a Scenario 2 Accident

Worker Category	Estimated Worker Requirements[(a)] (man-yr)	Estimated Occupational Dose: Total[(b)] (man-rem)	Estimated Occupational Dose: Individual Average (man-rem/man-yr)	Adjustment Factor[(c)]	Adjusted Worker Requirements (man-yr)
Crew leader	14.5	321.1	22.2	4.5	65.3
Utility operator	34.2	703.8	20.6	4.1	140.3
Laborer	16.6	433.7	26.2	5.3	88.0
Craftsman	15.1	401.8	26.6	5.4	81.6
Health Physics Technician	15.3	343.3	22.5	4.5	68.9
Totals	95.7	2203.8			444.1

(a) Based on Table K.5-4.
(b) Based on Table K.5-5.
(c) Increase in worker requirements necessary to reduce average individual dose to ≤ 5 man-rem/man-yr.

following a scenario 2 accident. For DECON in the reactor building and the containment vessel following a scenario 3 accident, the occupational radiation dose and the adjusted worker requirement are estimated to be about 2 times their values for DECON following a scenario 2 accident.

K.5.2.2 DECON in the Other Buildings

The scenario 1 and scenario 2 accidents are not postulated to result in significant radioactive contamination of the radwaste and control building or the turbine-generator building. Contamination of the radwaste building is postulated for the scenario 3 accident; however, accident cleanup is assumed to reduce the contamination in this building to levels that allow routine worker access to systems and equipment needed for subsequent accident cleanup and decommissioning operations. The requirements for post-accident DECON in these buildings and in other plant structures are assumed to be about the same as for decommissioning following normal shutdown. Procedures, schedules, manpower requirements, and occupational doses for DECON following normal shutdown are given in Appendix I (Sections I.1 and I.2) of Reference 1.

Some radioactive contamination of onsite storage structures used for interim storage of the radioactive wastes from accident cleanup is expected due to package failures, smearable contamination on package surfaces, etc. Therefore, these structures require decontamination before DECON is completed. The methods and the time and manpower requirements for DECON in these structures at the reference BWR are assumed to be similar to those for DECON at the reference PWR, described in Appendix G.

K.5.2.3 Schedule and Decommissioning Worker Requirements for DECON

The overall schedule and sequence for DECON at the reference BWR following a scenario 2 accident and the subsequent accident cleanup campaign is shown in Figure K.5-1. DECON begins in the reactor building and the containment vessel which comprise the major effort by the decommissioning staff. The work proceeds through the turbine-generator building, the radwaste and control building, and other support structures as staff are available and as the various systems in these buildings complete their required service functions. As shown in Figure K.5-1, DECON following a scenario 2 accident

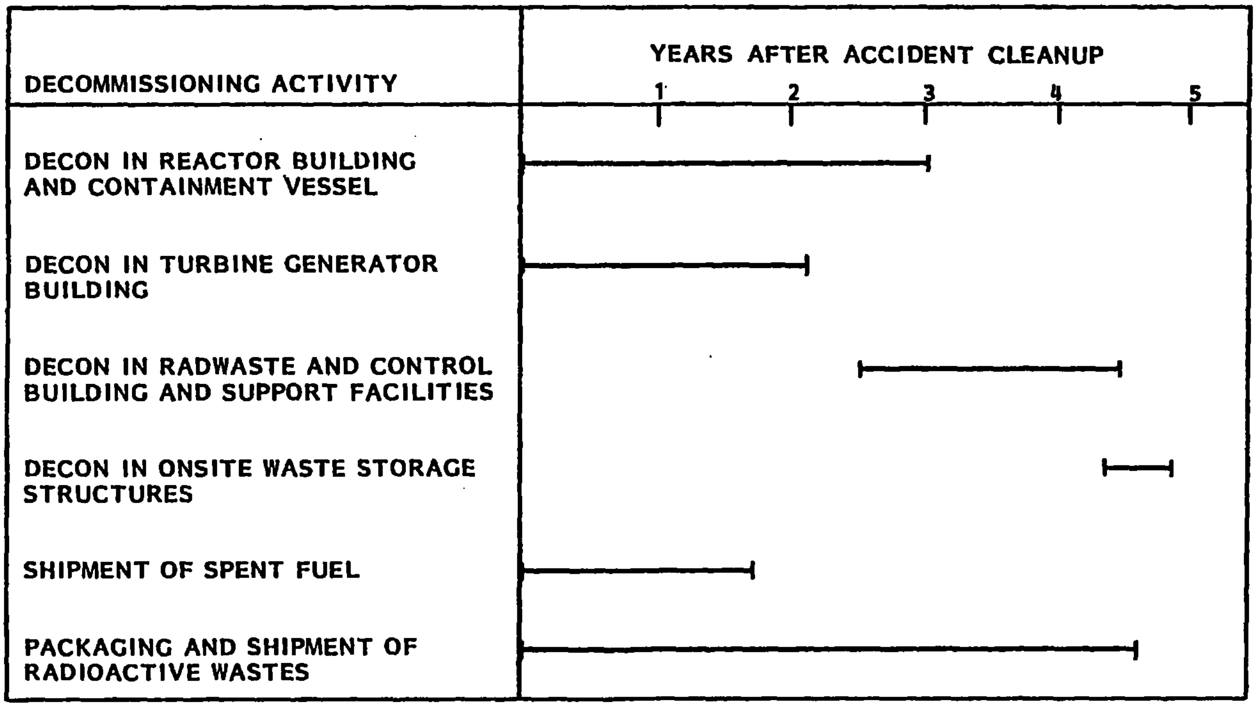


FIGURE K.5-1. Overall Schedule and Sequence for DECON at the Reference BWR Following a Scenario 2 Accident

and the subsequent accident cleanup is estimated to require about 4.8 years for completion. (DECON at the reference BWR following normal shutdown is estimated to require about 3.5 years.[1]) Variations in accident severity, within the range of accident scenarios considered in this study, are estimated to change the duration of post-accident DECON by about ±0.2 years.

The adjusted decommissioning worker requirements for DECON at the reference BWR following a scenario 2 accident are shown in Table K.5-7. The total estimated decommissioning worker requirement for post-accident DECON is about 720 man-years and includes the extra manpower needed to maintain compliance with occupational radiation dose limits but does not include management and support staff.

The packaging and shipping of radioactive wastes generated during DECON is handled by standing crews that are available over the entire duration of DECON activities until tasks that generate the wastes are completed. These crews also package and ship the fuel removed from the reactor during accident

TABLE K.5-7. Adjusted Decommissioning Worker Requirements for DECON at the Reference BWR Following a Scenario 2 Accident[a]

DECON Activity Area	Estimated Decommissioning Worker Requirements (man-years)				
	Crew Leaders	Utility Operators	Laborers	Craftsmen	Health Physics Technicians
Reactor building and containment vessel[b]	65.3	140.3	88.0	81.6	68.9
Turbine generator building[c]	7.9	19.8	10.2	35.2	6.9
Radwaste and control building and support facilities[c]	6.9	17.7	9.1	28.2	18.6
Onsite waste storage structures[d]	0.9	2.5	2.5	1.7	1.3
Packaging and shipment of radioactive wastes[e]	22.1	44.2	44.2	____	____
Totals	103.1	224.5	154.0	146.7	95.7

(a) Includes extra labor needed to maintain compliance with occupational radiation dose limits.
(b) From Table K.5-6.
(c) Estimated on the basis of manpower requirements and exposure hours shown in Table I.2-2 and Table I.4-1 of Reference 1.
(d) From Tables G.2-1 and G.2-2.
(e) Includes shipping activities required to complete spent fuel shipments, remove accident cleanup wastes from onsite waste storage structures, and remove all decommissioning wastes.

cleanup and stored in the spent fuel pool. Because the spent fuel removed after a scenario 2 or scenario 3 accident is placed in canisters, fewer fuel assemblies can be shipped in a cask, and the time requirement for completing the fuel shipment task is increased by about 50% over that required to complete the shipment of uncanistered assemblies (assuming the availability of the same number of casks).

Adjusted manpower requirements for packaging and shipping crews are shown as a line item in Table K.5-7. 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 for post-accident than for 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 post-accident DECON is the limitation on radioactive doses to individual workers. Decommissioning worker manpower requirements for post-accident DECON 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.

K.5.2.4 Total Staff Labor Requirements for DECON

Total utility staff labor requirements for post-accident DECON at the reference BWR following a scenario 2 accident are shown in Table K.5-8. Staff labor requirements include management and support staff and adjusted decommissioning worker requirements but do not include contractor personnel. The total staff labor requirement for DECON following the scenario 2 accident is estimated to be about 1070 man-years. The estimated staff labor requirement for DECON following the scenario 1 accident is approximately 670 man-years and for DECON following the scenario 3 accident is approximately 1850 man-years.

K.5.3 Details of SAFSTOR at the Reference BWR

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. SAFSTOR has the advantage of satisfying the requirements for protection of the public while reducing, to various degrees, initial commitments of time, money, occupational radiation dose, and waste disposal requirements compared to DECON. These advantages are offset somewhat by the need to maintain the nuclear license and by the associated commitment to continuing care of the facility. The decay of radioactive contamination

TABLE K.5-8. 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 and accounting supervisor	0	5.1[b]	5.1
Health and safety supervisor	0	5.1[b]	5.1
Health physics	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.

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. Deferral of decontamination until after long periods of safe storage also has the disadvantage that personnel familiar with the plant and with post-accident cleanup activities are no longer available to staff the decontamination effort.

The activities and requirements for post-accident SAFSTOR, including preparations for safe storage, continuing care, and deferred decontamination are similar to those for SAFSTOR following normal shutdown, which are described in Appendix J of Reference 1. Post-accident SAFSTOR is summarized here, with emphasis on those activities and requirements that differ significantly from the ones for normal shutdown SAFSTOR.

K.5.3.1 Preparations for Safe Storage in the Reactor Building and the Containment Vessel

In general, 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
- installation of barriers and monitoring systems needed during continuing care.

A 13-point procedure for preparing contaminated areas throughout major plant structures for safe storage is described in Appendix J of Reference 1. The following tasks are included:

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.

Methods for accomplishing these tasks during preparations for safe storage following normal shutdown are described in Appendix G and Appendix J of Reference 1, and these same methods are generally applied during post-accident preparations for safe storage.

Preparations for safe storage following normal shutdown include the draining of pools and tanks and the chemical decontamination of reactor water recirculating systems and of contaminated drains and pipes. As described in Section K.5.2.1, some chemical decontamination of these systems is accomplished during accident cleanup. Remaining systems and piping are decontaminated during preparations for safe storage. If reactor fuel is to remain in storage in the spent fuel pool during the continuing care period, this pool is not drained and the fuel pool cooling and cleanup system must be maintained in operable condition. The base case analysis of this study assumes that the fuel is removed, the pool is drained and cleaned, and the fuel pool cooling and cleanup system is decontaminated.

Initial decontamination of building surfaces in the reactor building and the containment vessel is performed during accident cleanup to reduce radiation doses to cleanup workers. Some additional decontamination and shielding of "hot spots" 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.

Estimated direct worker requirements for preparations for safe storage in the reactor building and the containment vessel following a scenario 2 accident are shown in Table K.5-9. The tasks shown in the table are generally the same as those shown in Figure J.4-1 of Reference 1 for preparations for safe storage in the reactor building following normal shutdown, with the addition or deletion of some tasks because of differences in the condition of the building following the accident and accident cleanup. Adjustments to the time requirements for completion of certain tasks have been made to take account of post-accident conditions. The labor requirements shown in the table include only the labor estimated to be needed to actually complete each task and do not include support staff or the additional personnel required for compliance with occupational dose limitations.

Estimated occupational radiation doses to workers engaged in preparations for safe storage in the reactor building and the containment vessel following the scenario 2 accident are shown in Table K.5-10. The occupational doses shown in the table are external doses from gamma radiation. Dose calculations are based on time and manpower requirements shown in Table K.5-9. Exposure hours are estimated assuming that workers engaged in preparations for safe storage spend an average of 6 hours in a radiation area during an 8-hour shift.

As explained in Section K.5.2.1, the estimated worker requirements for decommissioning workers are adjusted upward to provide sufficient manpower to ensure that no individual worker receives a whole-body radiation dose in excess of 5 rem/year. Adjusted worker requirements for preparations for safe storage in the reactor building and the containment vessel following the scenario 2 accident are shown in Table K.5-11. For preparations for safe storage following the scenario 2 accident, this decommissioning worker requirement is estimated to be about 120 man-years. For preparations for safe storage in the reactor building and the containment vessel following the

TABLE K.5-9. Estimated Direct Worker Requirements for Preparations for Safe Storage in the Reactor Building and Containment Vessel Following a Scenario 2 Accident

Preparations for Safe Storage Activity	Time Required (months)	Shift Basis: 2 Shifts 5-day week	Shift Basis: 3 Shifts 7-day week	Crew Requirement per Shift: Crew Leader	Crew Requirement per Shift: Utility Operator	Crew Requirement per Shift: Laborer	Crew Requirement per Shift: Craftsman	Crew Requirement per Shift: H.P. Technician	Labor Requirement (man-months)(a): Crew Leader	Labor Requirement (man-months)(a): Utility Operator	Labor Requirement (man-months)(a): Laborer	Labor Requirement (man-months)(a): Craftsman	Labor Requirement (man-months)(a): H.P. Technician	Labor Requirement (man-months)(a): Totals
Clean up and shield access routes and hot spots	6	x		1	2	2	2	1	12	24	24	24	12	96
Drain contaminated systems to radwaste	1-1/2		x	1	4			1	6	24			6	36
Chemical decon of RHR and core spray systems	1/2		x	1	4			1	2	8			2	12
Drain dryer and separator pool - water jet clean	1		x	1	4			1	4	16			4	24
Chemical decon of drain systems	1		x	1	4			1	4	16			4	24
Drain spent fuel pool - water jet clean	2		x	1	4			1	8	32			8	48
Chemical decon of fuel pool cooling and cleanup system	1/2		x	1	4			1	2	8			2	12
Cover and seal spent fuel pool and dryer and separator pool	1/2	x		1	2	2	3		1	2	2	3		8
Seal equipment and personnel hatches into primary containment	1	x		1	2	2	4	1	2	4	4	8	2	20
Decontaminate HVAC, electrical, and miscellaneous structures and equipment and apply protective paint	3	x		1	3	3		1	6	18	18		6	48
Isolate and seal equipment, piping, rooms, ducts, and tunnels	3	x		1	2	2	4	1	6	12	12	24	6	60
Seal drywell top head and unneeded doors	1/2	x		1			6	1	1			6	1	8
Install HEPA-filtered vents	1/2	x		1	1	1	4	1	1	1	1	4	1	8
Deactivate unnecessary utilities	1/2	x		1	1	2	3		1	1	2	3		7
Install intrusion, radiation monitoring, and fire alarm systems	1	x		1		1	2	1	2		2	4	2	10
Final radiation survey of reactor building	1	x						2					4	4
Totals									58	166	65	76	60	425

(a) Based on 20 working days per month.

TABLE K.5-10. Estimated Occupational Radiation Doses for Preparations for Safe Storage in the Reactor Building and Containment Vessel Following a Scenario 2 Accident

Preparations for Safe Storage Activity	Estimated Average Dose Rate (rem/hr)	Crew Leader Exposure(a) (man-hr)	Crew Leader Dose (man-rem)	Utility Operator Exposure(a) (man-hr)	Utility Operator Dose (man-rem)	Laborer Exposure(a) (man-hr)	Laborer Dose (man-rem)	Craftsman Exposure(a) (man-hr)	Craftsman Dose (man-rem)	H.P. Technician Exposure(a) (man-hr)	H.P. Technician Dose (man-rem)	Task Totals Exposure(a) (man-hr)	Task Totals Dose (man-rem)
Clean up and shield access routes and hot spots	0.015	1 440	21.6	2 880	43.2	2 880	43.2	2 880	43.2	1 440	21.6	11 520	172.8
Drain contaminated systems to radwaste	0.020	720	14.4	2 880	57.6					720	14.4	4 320	86.4
Chemical decon of RHR and core spray systems	0.020	240	4.8	960	19.2					240	4.8	1 440	28.8
Drain dryer and separator pool - water jet clean	0.008	480	3.8	1 920	15.4					480	3.8	2 880	23.0
Chemical decon of drain systems	0.008	480	3.8	1 920	15.4					480	3.8	2 880	23.0
Drain spent fuel pool - water jet clean	0.008	960	7.6	3 840	30.8					960	7.6	5 760	46.0
Chemical decon of fuel pool cooling and cleanup system	0.008	240	1.9	960	7.7					240	1.9	1 440	11.5
Cover and seal spent fuel pool and dryer and separator pool	0.008	120	1.0	240	1.9	240	1.9	360	2.9			960	7.7
Seal equipment and personnel hatches into primary containment	0.015	240	3.6	480	7.2	480	7.2	960	14.4	240	3.6	2 400	36.0
Decontaminate HVAC, electrical, and miscellaneous structures and equipment and apply protective paint	0.010	720	7.2	2 160	21.6	2 160	21.6			720	7.2	5 760	57.6
Isolate and seal equipment, piping, rooms, ducts, and tunnels	0.010	720	7.2	1 440	14.4	1 440	14.4	2 880	28.8	720	7.2	7 200	72.0
Seal drywell top head and unneeded doors	0.005	120	0.6					720	3.6	120	0.6	960	4.8
Install HEPA-filtered vents	0.005	120	0.6	120	0.6	120	0.6	480	2.4	120	0.6	960	4.8
Deactivate unnecessary utilities	0.005	120	0.6	120	0.6	240	1.2	360	1.8			840	4.2
Install intrusion, radiation monitoring and fire alarm systems	0.005	240	1.2			240	1.2	480	2.4	240	1.2	1 200	6.0
Final radiation survey of reactor building	0.005									480	2.4	480	2.4
Totals		6 960	79.9	19 920	235.6	7 800	91.3	9 120	99.5	7 200	80.7	51 000	587.0

(a) Based on estimated direct worker requirements from Table K.5-9. Assume 20 working days per month; 6 hours of exposure per shift.

TABLE K.5-11. Adjustments to Decommissioning Worker Requirements to Comply with Occupational Radiation Dose Limitations for Preparations for Safe Storage in the Reactor Building and Containment Vessel Following a Scenario 2 Accident

Worker Category	Estimated Worker Requirements[a] (man-yr)	Estimated Occupational Dose: Total[b] (man-rem)	Estimated Occupational Dose: Individual Average (man-rem/man-yr)	Adjustment Factor[c]	Adjusted Worker Requirements (man-yr)
Crew leader	4.9	79.9	16.3	3.3	16.2
Utility operator	13.9	235.6	17.0	3.4	47.3
Laborer	5.4	91.3	16.9	3.4	18.4
Craftsman	6.4	99.5	15.6	3.2	20.5
Health Physics Technician	5.0	80.7	16.2	3.3	16.5
Totals	35.6	587.0			118.9

(a) Based on Table K.5-9.
(b) Based on Table K.5-10.
(c) Increase in worker requirements necessary to reduce average individual dose to $\leq$5 man-rem/man-yr.

scenario 1 accident, this adjusted worker requirement is estimated to be about 80 man-years, and following the scenario 3 accident this requirement is estimated to be about 180 man-years.

K.5.3.2 Preparations for Safe Storage in the Other Buildings

The time requirements, schedules, and manpower requirements for 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 decommissioning as they are for decommissioning following normal shutdown. Procedures, schedules, and manpower requirements for preparations for safe storage at the reference BWR following normal shutdown are given in Sections J.3 and J.4 (Appendix J) of Reference 1.

The procedures and 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 at the reference PWR, described in Appendix G.

K.5.3.3 Schedule and Decommissioning Worker Requirements for Preparations for Safe Storage

The overall schedule and sequence for preparations for safe storage at the reference BWR following a scenario 2 accident and the subsequent accident cleanup campaign is shown in Figure K.5-2. As with DECON, the preparations for safe storage phase of SAFSTOR begins in the reactor building and the containment vessel, which represent the major effort for the decommissioning staff. The work proceeds through the other buildings as staff are available and as the various systems involved complete their required service functions. As shown in Figure K.5-2, preparations for safe storage following a scenario 2 accident and the subsequent accident cleanup campaign is estimated to require about 2.8 years for completion. (Preparations for safe

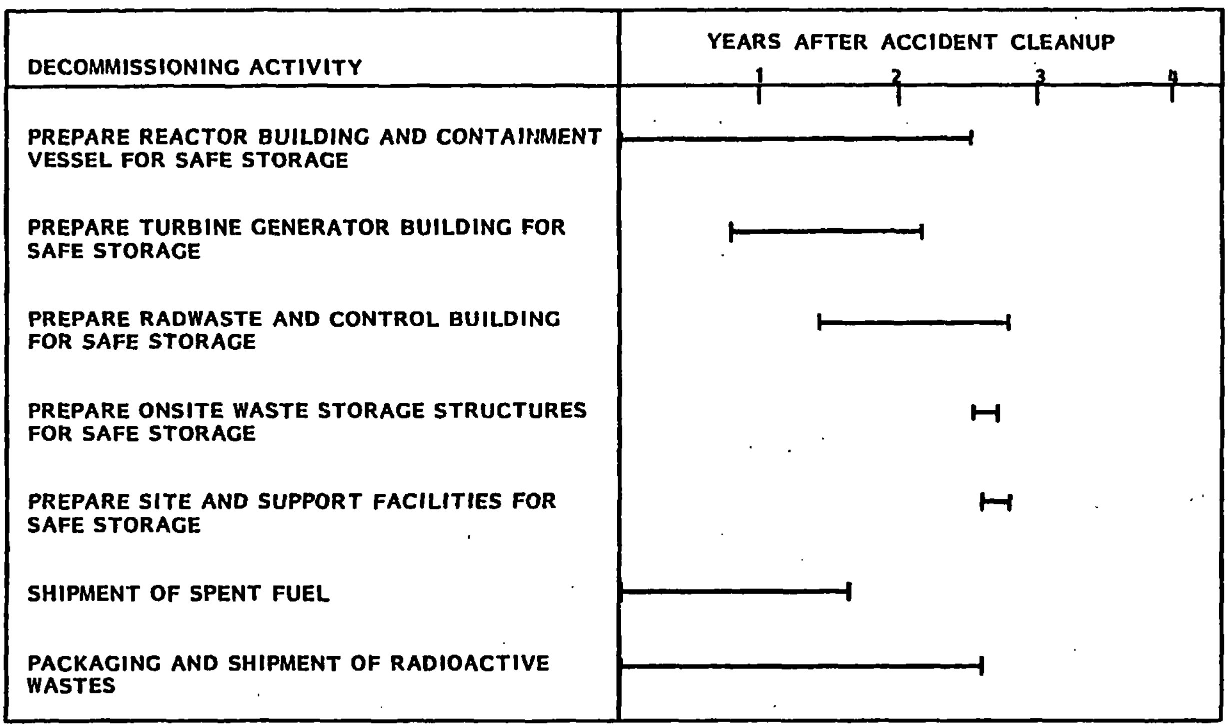


FIGURE K.5-2. Overall Schedule and Sequence for Preparations for Safe Storage at the Reference BWR Following a Scenario 2 Accident storage at the reference BWR following normal shutdown is estimated to require about 2.5 years.)[1] Variations in accident severity, within the range of accident scenarios considered in this study, are estimated to change the duration of preparations for safe storage by about ±0.1 years.

The adjusted decommissioning worker requirements for preparations for safe storage at the reference BWR following a scenario 2 accident are shown in Table K.5-12. The total estimated decommissioning worker requirement for post-accident preparations for safe storage is about 220 man-years and includes the requirements of packaging and shipping crews as described in Section K.5.2.3 and the extra manpower needed to maintain compliance with occupational radiation dose limits, but does not include management and support staff.

Because the duration of the preparations for safe storage effort is only slightly altered by accident severity, the major factor affecting manpower requirements for post-accident preparations for safe storage is the limitation on radioactive doses to individual workers. Decommissioning worker requirements for post-accident preparations for safe storage 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 3 accident as following a scenario 2 accident.

K.5.3.4 Total Staff Labor Requirements for Preparations for Safe Storage

Total utility staff labor requirements for post-accident preparations for safe storage at the reference BWR following a scenario 2 accident are shown in Table K.5-13. Staff labor requirements include management and support staff and adjusted decommissioning worker requirements but do not include contractor personnel. The total staff labor requirement for preparations for safe storage following the scenario 2 accident is estimated to be about 425 man-years. The estimated staff labor requirement for preparations for safe storage following the scenario 1 accident is approximately 340 man-years, and for preparations for safe storage following the scenario 3 accident is approximately 560 man-years.

TABLE K.5-12. Adjusted Decommissioning Worker Requirements for Preparations for Safe Storage at the Reference BWR Following a Scenario 2 Accident[(a)]

Preparations for Safe Storage Activity Area	Estimated Decommissioning Worker Requirements (man-years)				
	Crew Leaders	Utility Operators	Laborers	Craftsmen	Health Physics Technicians
Reactor building and containment vessel[(b)]	16.2	47.3	18.4	20.5	16.5
Turbine generator building[(c)]	7.2	7.7	2.2	9.2	2.4
Radwaste and control building and support facilities[(d)]	4.8	13.1	3.7	6.4	2.4
Onsite waste storage structures[(d)]	0.4	1.0	1.0	0.7	0.5
Packaging and shipment of Radioactive Wastes[(e)]	7.8	15.6	15.6	___	___
Totals	36.4	84.7	40.9	36.8	21.8

(a) Includes extra labor needed to maintain compliance with occupational radiation dose limits.
(b) From Table K.5-11.
(c) Estimated on the basis of manpower requirements and exposure hours shown in Figure J.4-1 and Table J.6-1 of Reference 1.
(d) Estimated by multiplying the decommissioning worker requirements shown in Table K.5-7 by the ratios of exposure hours for preparations for safe storage to exposure hours for DECON in onsite waste storage structures obtained from Tables G.2-2 and G.3-1 of Appendix G.
(e) Includes shipping activities required to complete spent fuel shipment, remove accident cleanup wastes from onsite waste storage structures and remove all decommissioning wastes.

TABLE K.5-13. 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	Staff Labor Requirement (man-years) in Decommissioning Phase[a]: Preparations for Safe Storage	Total Staff Labor Required (man-years)
Management and Support Staff			
Decommissioning superintendent	0.5	3.0[b]	4.6
Secretary	3.0	8.7[b]	00.7
Clerk	0.0	5.6	6.6
Decommissioning engineer	0.5	3.0[b]	4.6
Assistant decommissioning engineer	0.5	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	00.2	00.2
Security patrolman	0	33.6	33.6
Contracts and accounting supervisor	0	3.0[b]	3.0
Health and safety supervisor	0	3.0[b]	3.0
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	3.0[b]	3.0
Quality assurance engineer	0	2.8	2.8
Quality assurance technician	0	00.2	00.2
Consultant (safety review)	0	0.4	0.4
Instrument technician[d]	0	00.2	00.2
Maintenance mechanic[d]	0	00.2	00.2
Warehouseman	0	5.6	5.6
Subtotals	8.5	068.4	076.9
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	00.2	00.2
Craftsman[e]	0	36.8	36.8
Senior health physics technician	0	00.2	00.2
Health physics technician[e]	0	20.8	20.8
Subtotals	0	248.6	248.6
Totals	8.5	407.0	425.5

(a) Rounded to the nearest 0.0 man-years.
(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, 3 shifts per day, 7 days per week.
(d) Based on one per shift, three shifts per day, 7 days per week to maintain essential sources.
(e) From Table K.5-02.

K.5.3.5 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 BWR following post-accident preparations for safe storage is assumed to be approximately the same as it is for the normal shutdown case. From Table J.4-2 of Reference 1, the annual labor requirement is estimated to be less than 1.5 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 45 man-years or 150 man-years, respectively.

The level of effort required to efficiently perform the work of deferred decontamination is assumed to be about the same as that required for DECON, described in Section K.5.2. A number of dismantlement tasks, such as the draining and decontamination of contaminated liquid systems and the removal of radioactive wastes such as filters, resins, and evaporator bottoms, 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 that were unneeded during the continuing care period. Therefore, it is assumed that the basic work force (i.e., the decommissioning worker requirement for efficient performance of the decontamination tasks) and the time required for deferred decontamination are the same as for DECON.

As described in Section K.5.2.1, the actual decommissioning worker requirements for DECON and for deferred decontamination 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 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.

K.5.4 Details of ENTOMB at the Reference BWR

Post-accident 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) the residual radioactivity levels in the facility following an accident, even after substantial reactor 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 large quantities of ^{137}Cs (with a 30-year half-life) released by the accident. Post-accident ENTOMB requires continuation of the facility's nuclear license during a period of safe storage until the entombment structure is reopened and the materials stored inside are surveyed and either released for unrestricted use or packaged and shipped to a disposal site.

The entombment activities during post-accident ENTOMB at the reference BWR are similar to those during ENTOMB following normal shutdown, which are described in detail in Appendix K of Reference 1. Entombment of radioactive materials in the reference BWR is assumed to take place within the confines of the steel primary containment vessel and the surrounding concrete biological shield. All plant areas outside of the entombment barrier are decontaminated to allow unrestricted release if desired. Post-accident entombment activities, schedules, and manpower requirements are summarized here, with emphasis on those activities and requirements that differ significantly from the ones during normal-shutdown ENTOMB.

K.5.4.1 ENTOMB in the Reactor Building and the Containment Vessel

The entombment barrier for the reference BWR consists of the primary containment vessel enclosed within a monolithic concrete envelope (the sealed biological shield) that rests on the reactor building foundation mat. Sealing of all barrier penetrations (e.g., for personnel, equipment, material, and services) is necessary. Equipment and personnel access openings into the containment vessel, as well as the stub ends of cut-off piping, are sealed by welded plate closures. All openings through the biological shield are then filled with cast-in-place, reinforced concrete. The removable concrete shield plugs are grouted in place to complete the encasement of the radioactive materials within an integral, monolithic concrete envelope. The reactor building is sealed and left in place to provide a secondary barrier that provides all-weather protection and enhanced security for the entombment structure.

Prior to sealing of the primary containment, those reactor vessel internals containing long-lived activation products (e.g., ^{59}Ni, ^{94}Nb) are removed from the facility and shipped offsite to a nuclear waste repository. Dismantlement of the facility outside the entombment structure is carried out as for the DECON alternative, the major difference being that as much as possible of the contaminated equipment and material in the plant is consolidated within the entombment structure rather than being packaged and shipped to offsite disposal. This reduces the radioactive material disposal costs associated with initial decommissioning. Methods postulated for entombment of the reactor facility following normal shutdown are described in Appendix K of Reference 1, and these same methods are generally applied to post-accident ENTOMB.

To reduce radiation doses to workers engaged in post-accident entombment activities inside the reactor building and the containment vessel, the major access routes used by these 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 ENTOMB to obtain the maximum dose-reduction benefits. Cleanup methods postulated for this task are the same as those used for accident cleanup described in Appendix E.

Estimated direct worker requirements for ENTOMB in the reactor building and the containment vessel following a scenario 2 accident are shown in Table K.5-14. The tasks shown in the table are generally the same as those shown in Table K.2-1 of Reference 1, with the deletion of some tasks completed during accident cleanup and the addition of some new tasks that are required because of the accident. Time requirements for completing some tasks are adjusted to account for the effects of the accident and of post-accident cleanup activities. Labor requirements shown in the table include only the staff labor needed to actually complete the designated tasks and do not include management and support staff or the additional personnel required for compliance with occupational dose limitations.

Estimated occupational radiation doses to workers engaged in entombment operations in the reactor building and the containment vessel following the scenario 2 accident are shown in Table K.5-15. The occupational doses shown in the table are external doses from gamma radiation. Dose calculations are based on time and manpower requirements shown in Table K.5-14. Exposure hours are estimated assuming that workers spend an average of 6 hours in a radiation area during an 8-hour shift.

Adjusted worker requirements necessary to reduce average individual radiation doses to ≤5 man-rem/man-year are shown in Table K.5-16. For ENTOMB activities in the reactor building and the containment vessel following the scenario 2 accident, this decommissioning worker requirement is estimated to be about 315 man-years. For ENTOMB activities in the reactor building and the containment vessel following the scenario 1 accident, the adjusted worker requirement is estimated to be about 160 man-years, and following the scenario 3 accident this requirement is estimated to be about 630 man-years.

K.5.4.2 ENTOMB in the Other Buildings

ENTOMB activities outside of the entombment structure area are generally the same as the corresponding DECON activities in those areas. Of the

TABLE K.5-14. Estimated Direct Worker Requirements for ENTOMB in the Reactor Building and Containment Vessel Following a Scenario 2 Accident

ENTOMB Activity	Time Required (months)	Shift Basis: 2 Shifts 5-day week	Shift Basis: 3 Shifts 7-day week	Crew Requirement per Shift: Crew Leader	Crew Requirement per Shift: Utility Operator	Crew Requirement per Shift: Laborer	Crew Requirement per Shift: Craftsman	Crew Requirement per Shift: H.P. Technician	Labor Requirement (man-months)[a]: Crew Leader	Labor Requirement (man-months)[a]: Utility Operator	Labor Requirement (man-months)[a]: Laborer	Labor Requirement (man-months)[a]: Craftsman	Labor Requirement (man-months)[a]: H.P. Technician	Totals
Clean up and shield access routes and hot spots	6	X		1	2	2	2	1	12	24	24	24	12	96
Remove reactor vessel internals	6-1/2	X			2		2			26		26		52
Ship activated reactor vessel internals	13		X	1	2			1	52	104			52	208
Cut suppression pool downcomers and bracing	1	X				4	2	1			8	4	2	14
Chemical decon of RHR and core spray systems	1/2		X	1	4			1	2	8			2	12
Drain contaminated systems to radwaste	1-1/2		X	1	4			1	6	24			6	36
Drain reactor well pool—water jet clean	1		X	1	4			1	4	16			4	24
Clean up and shield hot spots in primary containment	3	X				2		1			12		6	18
Drain dryer and separator pool—water jet clean	1		X	1	4			1	4	16			4	24
Cut suppression chamber access through drywell floor	1/2	X				4	2	1			4	2	1	7
Chemical decon of drain systems	1		X	1	4			1	4	16			4	24
Cut primary containment piping penetrations and seal	2	X		1	2	4	4		4	8	16	16		44
Cut drywell bellows access openings.	1	X				2	1				4	2		6
Remove reactor building piping	10-1/2	X		1	2	2	2	1	21	42	42	42	21	168
Drain spent fuel pool—water jet clean	2		X	1	4			1	8	32			8	48
Chemical decon fuel pool cooling and cleanup system	1/2		X	1	4			1	2	8			2	12
Remove liners from spent fuel pool and dryer and separator pool	1-1/2	X		1		2	2	1	3		6	6	3	18
Remove reactor building equipment	3	X		1	1	3	1	1	6	6	18	6	6	42
Seal equipment and personnel hatches into primary containment	1	X		1		2	2		2		4	4		10
Remove reactor building contaminated concrete	3	X				2					12			12
Seal rail and steam tunnel and biological shield penetrations	1	X		1		2	1	1	2		4	2	2	10
Seal drywell top head and reactor building external doors	1/2	X				2	2				2	2		4
Remove HVAC and disable crane	2	X		1		2	2	1	2		4	4	2	12
Install monitoring equipment; disconnect unnecessary utilities	1			1		1	2	1	2		2	4	2	10
Final radiation survey of reactor building	1	X						2					4	4
Totals									136	330	162	144	143	915

(a) Based on 20 working days per month.

TABLE K.5-15. Estimated Occupational Radiation Doses for ENTOMB in the Reactor Building and Containment Vessel Following a Scenario 2 Accident

ENTOMB Activity	Estimated Average Dose Rate (rem/hr)	Crew Leader		Utility Operator		Laborer		Craftsman		H.P. Technician		Task Totals	
		Exposure[a] (man-hr)	Dose (man-rem)	Exposure[a] (man-hr)	Dose (man-rem)	Exposure[a] (man-hr)	Dose (man-rem)	Exposure[a] (man-hr)	Dose (man-rem)	Exposure[a] (man-hr)	Dose (man-rem)	Exposure[a] (man-hr)	Dose (man-rem)
Clean up and shield access routes and hot spots	0.015	1 440	21.6	2 880	43.2	2 880	43.2	2 880	43.2	1 440	21.6	11 520	172.8
Remove reactor vessel internals	0.015			3 120	46.8			3 120	46.8			6 240	93.6
Ship activated reactor vessel internals	0.015	6 240	93.6	12 480	187.2					6 240	93.6	24 960	374.4
Cut suppression pool downcomers and bracing	0.030					960	28.8	480	14.4	240	7.2	1 680	50.4
Chemical decon of RHR and core spray systems	0.020	240	4.8	960	19.2					240	4.8	1 440	28.8
Drain contaminated systems to radwaste	0.020	720	14.4	2 880	57.6					720	14.4	4 320	86.4
Drain reactor well pool--water jet clean	0.008	480	3.8	1 920	15.4					480	3.8	2 880	23.0
Clean up and shield hot spots in primary containment	0.030					1 440	43.2			720	21.6	2 160	64.8
Drain dryer and separator pool--water jet clean	0.008	480	3.8	1 920	15.4					480	3.8	2 880	23.0
Cut suppression chamber access through drywell floor	0.025					480	12.0	240	6.0	120	3.0	840	21.0
Chemical decon of drain systems	0.005	480	2.4	1 920	9.6					480	2.4	2 880	14.4
Cut primary containment piping penetrations and seal	0.015	480	7.2	960	14.4	1 920	28.8	1 920	28.8			5 280	79.2
Cut drywell bellows access openings	0.015					480	7.2	240	3.6			720	10.8
Remove reactor building piping	0.020	2 520	50.4	5 040	100.8	5 040	100.8	5 040	100.8	2 520	50.4	20 160	403.2
Drain spent fuel pool--water jet clean	0.005	960	4.8	3 840	19.2					960	4.8	5 760	28.8
Chemical decon fuel pool cooling and cleanup system	0.005	240	1.2	960	4.8					240	1.2	1 440	7.2
Remove liners from spent fuel pool and dryer and separator pool	0.005	360	1.8			720	3.6	720	3.6	360	1.8	2 160	10.8
Remove reactor building equipment	0.003	720	2.2	720	2.2	2 160	6.5	720	2.2	720	2.2	5 040	15.3
Seal equipment and personnel hatches into primary containment	0.015	240	3.6			480	7.2	480	7.2			1 200	18.0
Remove reactor building contaminated concrete	0.005					1 440	7.2					1 440	7.2
Seal rail and steam tunnel and biological shield penetrations	0.010	240	2.4			480	4.8	240	2.4	240	2.4	1 200	12.0
Seal drywell top head and reactor building external doors	0.005					240	1.2	240	1.2			480	2.4
Remove HVAC and disable crane	0.005	240	1.2			480	2.4	480	2.4	240	1.2	1 440	7.2
Install monitoring equipment; disconnect unnecessary utilities	0.005	240	1.2			240	1.2	480	2.4	240	1.2	1 200	6.0
Final radiation survey of reactor building	0.003									480	1.5	480	1.5
Totals		16 320	220.4	39 600	535.8	19 440	298.1	17 280	265.0	17 160	242.9	109 800	1 562.2

(a) Based on estimated direct worker requirements from Table K.5-14. Assume 20 working days per month; 6 hours of exposure per shift.

TABLE K.5-16. Adjustments to Decommissioning Worker Requirements to Comply with Occupational Radiation Dose Limitations for ENTOMB in the Reactor Building and Containment Vessel Following a Scenario 2 Accident

Worker Category	Estimated Worker Requirements[a] (man-yr)	Estimated Occupational Dose: Total[b] (man-rem)	Estimated Occupational Dose: Individual Average (man-rem/man-yr)	Adjustment Factor	Adjusted Worker Requirements (man-yr)
Crew leader	11.4	220.4	19.4	3.9	44.5
Utility operator	27.5	535.8	19.5	3.9	107.3
Laborer	13.5	298.1	22.1	4.5	60.8
Craftsman	12.0	265.0	22.1	4.5	54.0
Health Physics Technician	11.9	242.9	20.4	4.1	48.8
Totals	76.3	1562.2			315.4

(a) Based on Table K.5-14.
(b) Based on Table K.5-15.
(c) Increase in worker requirements necessary to reduce average individual dose to ≤5 man-rem/man-yr.

approximately 11,000 m^3 of contaminated material from these other areas that requires disposal, approximately 7000 m^3 can be segmented and entombed. The remaining material is packaged for offsite 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. Entombing of this structure involves the sealing of the cover blocks in place and the decontamination of the upper parts of the structure, and is estimated to require about the same length of time and the same manpower as for DECON of the structure. The waste storage warehouse is decontaminated as during DECON.

K.5.4.3 Schedule and Decommissioning Worker Requirements for ENTOMB

The overall schedule and sequence for ENTOMB at the reference BWR following a scenario 2 accident and the subsequent accident cleanup campaign is shown in Figure K.5-3. 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. As shown

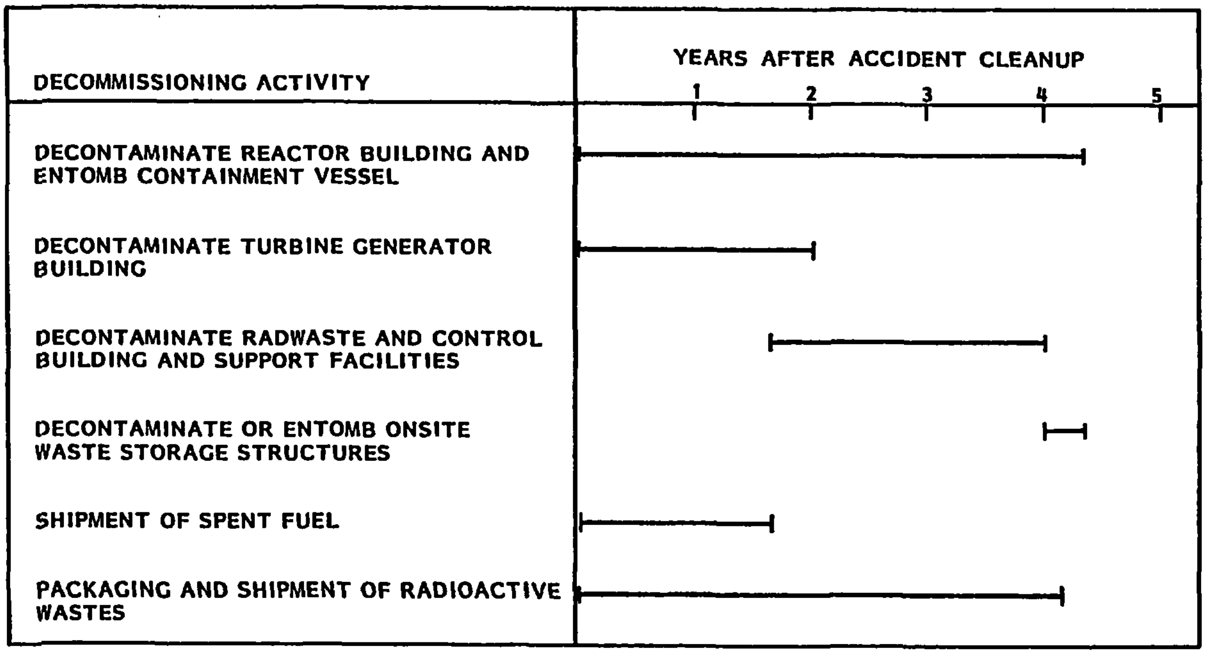


FIGURE K.5-3. Overall Schedule and Sequence for ENTOMB at the Reference BWR Following a Scenario 2 Accident

in Figure K.5-3, ENTOMB following a scenario 2 accident and the subsequent accident cleanup is estimated to require about 4.4 years for completion. (ENTOMB at the reference BWR following normal shutdown is estimated to require about 4.0 years.)[1] Variations in accident severity, within the range of accident scenarios considered in this study, are estimated to change the duration of post-accident ENTOMB by about ±0.2 years.

The adjusted decommissioning worker requirements for ENTOMB at the reference BWR following a scenario 2 accident are shown in Table K.5-17. The total estimated decommissioning worker requirement for post-accident ENTOMB is about 560 man-years and includes the requirements for packaging and shipping crews (who assist in the placement of wastes within the entombment structure as well as the shipment of wastes offsite) and the extra manpower needed to maintain compliance with occupational radiation dose limits, but does not include management and support staff.

As is the case for DECON manpower requirements discussed in Section K.5.2.3, decommissioning worker manpower requirements for ENTOMB are estimated

TABLE K.5-17. Adjusted Decommissioning Worker Requirements for ENTOMB at the Reference BWR Following a Scenario 2 Accident[(a)]

ENTOMB Activity Area	Crew Leaders	Utility Operators	Laborers	Craftsmen	Health Physics Technicians
	Estimated Decommissioning Worker Requirements (man-years)				
Reactor building and containment vessel[(b)]	44.5	107.3	60.8	54.0	48.8
Turbine generator building[(c)]	6.9	21.5	11.3	41.2	6.7
Radwaste and control building and support facilities[(c)]	6.1	19.3	10.1	32.8	17.9
Onsite waste storage structures[(d)]	0.9	2.5	2.5	1.7	1.3
Packaging and shipment of radioactive wastes[(e)]	12.3	24.6	24.6		
Totals	70.7	175.2	109.3	129.7	74.7

(a) Includes extra labor needed to maintain compliance with occupational radiation dose limits.
(b) From Table K.5-16.
(c) Estimated on the basis of manpower requirements and exposure hours shown in Figure K.2-1 and Table K.4-1 of Reference 1.
(d) Estimated to be approximately the same as for DECON (see Table K.5-7).
(e) Includes shipping activities required to complete spent fuel shipment, remove accident cleanup wastes from onsite waste storage structures, and remove all decommissioning wastes.

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.

K.5.4.4 Total Staff Labor Requirements for ENTOMB

Total utility staff labor requirements for ENTOMB at the reference BWR following a scenario 2 accident are shown in Table K.5-18. Staff labor requirements include management and support staff and adjusted decommissioning worker requirements but do not include contractor personnel. The total staff labor requirement for ENTOMB following the 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.

K.5.4.5 Continuing Care and Possible Deferred Decontamination of the Entombed Plant

The initial decommissioning activities for ENTOMB are followed by a period of continuing care that includes security, surveillance, and maintenance. Continuing care activities for ENTOMB are judged to require a lower level of effort than the comparable activities for SAFSTOR, because of the more elaborate preparation of the facility and the resulting reduced risk of a release of radioactivity. In Reference 1, the costs of continuing care for ENTOMB are estimated to be about half of those of continuing care for SAFSTOR. Assuming that labor accounts for about the same percentage of the total costs in either case and that the makeup of the labor force is approximately the same, the annual labor requirement following ENTOMB is about half the labor requirement for continuing care for SAFSTOR, or about 0.8 man-year/year. Thus, the total cummulative labor requirement for 100 years of continuing care following ENTOMB is estimated to be about 80 man-years.

Deferred decontamination following ENTOMB is anticipated to require about the same level of effort as deferred decontamination following SAFSTOR, discussed in Section K.5.3.5. Although there is less radioactive material to remove from 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 entombment structure. The methods used for deferred decontamination following ENTOMB are similar to those for DECON, described in Section K.5.2.

K.6 DETAILS OF COSTS OF DECOMMISSIONING

Details of the costs of decommissioning at the reference BWR following the postulated accidents and subsequent accident cleanup are given in this section. Cost estimates are made for each of the three decommissioning alternatives: DECON, SAFSTOR, and ENTOMB. Costs are based on the technical requirements, manpower needs, and work schedules described in Section K.5, and are in early-1981 dollars. Unit cost information used as bases for these cost estimates is given in Appendix I.

Detailed cost estimates are prepared only for decommissioning following a scenario 2 accident. Estimates of decommissioning costs following the other two accidents are obtained by adjustment of the scenario 2 decommissioning costs to account for costs that vary significantly with accident severity.

TABLE K.5-18. 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	Staff Labor Requirement (man-years) in Decommissioning Phase[a]: 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	6.2
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 Attendent	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 and Accounting Supervisor	0	4.7[b]	4.7
Health and 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	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.

K.6.1 Details of DECON Costs

The estimated cost of DECON at the reference BWR following a scenario 2 accident and the subsequent accident cleanup is summarized in Table K.6-1. The total estimated cost of DECON following a scenario 2 accident, including a 25% contingency, is about $86 million. Corresponding costs following a scenario 1 or scenario 3 accident are estimated to be about $67 million and $119 million, respectively. Information pertaining to individual cost categories is given in the following subsections.

K.6.1.1 Cost of Staff Labor

The costs of utility staff labor for DECON following a scenario 2 accident are shown in Table K.6-2. These costs are based on the utility staff labor requirements described in Section K.5.2. A total staff labor cost of about $37 million (without contingency) is estimated for DECON following a scenario 2 accident. This cost represents about 54% of the total DECON cost. Staff labor costs following a scenario 1 or scenario 3 accident are estimated to be about $23 million and $64 million, respectively, with the differences between labor costs for the various accident scenarios attributable mainly to the number of decommissioning workers needed to comply with individual radiation dose limitations to those workers (see Section K.5.2.1 and K.5.2.3). Specialty contractor labor is not included in the labor costs given here, but rather in the costs of specialty contractors presented in Section K.6.1.6.

K.6.1.2 Cost of Waste Management

Estimated costs of radioactive waste management for DECON following a scenario 2 accident are shown in Table K.6-3. A total waste management cost of about $14 million is estimated for DECON following a scenario 2 accident, representing about 21% of DECON costs. The waste management cost includes the container, transportation, and burial site costs, but does not include the direct labor costs for removing and packaging these materials, because these costs are included with the total costs of staff labor.

Three types of radioactive waste materials that require packaging, shipping, and disposal during DECON are: 1) neutron-activated materials, 2) contaminated materials, and 3) radioactive wastes.

TABLE K.6-1. Summary of Estimated Costs of DECON at the Reference BWR Following a Scenario 2 Accident

Cost Category	Estimated Costs ($ millions)[a]		Percent of Total
Staff Labor			
Management & Support Staff	10.625		
Decommissioning Workers	26.522		
Total Staff Labor Costs		37.147	54.2
Waste Management			
Neutron-Activated Materials	1.720		
Contaminated Materials	10.335		
Radioactive Wastes	2.308		
Total Waste Management Costs		14.363	20.9
Energy		9.302	13.6
Special Tools & Equipment		2.621	3.8
Miscellaneous Supplies		2.678	3.9
Specialty Contractors		0.417	0.6
Nuclear Insurance & License Fees		2.051	3.0
Subtotal		68.579	100.0
Contingency (25%)		17.145	
Total DECON Costs		85.724	

(a) Costs are adjusted to early 1981; the number of significant figures is for computational accuracy only.

TABLE K.6-2. Estimated Costs of Staff Labor During DECON at the Reference BWR Following a Scenario 2 Accident

Position	Unit Cost for Labor(a) ($ thousands/ man-year)	Total Staff Labor Required(b) (man-years)	Total Staff Labor Costs(c) ($ thousands)
Management and Support Staff			
Decommissioning Superintendent	89.4	6.6	590.0
Secretary	24.4	17.7	431.9
Clerk	24.4	10.6	258.6
Decommissioning Engineer	76.2	6.6	502.9
Assistant Decommissioning Engineer	52.6	6.3	331.4
Radioactive Shipment Specialist	39.5	4.8	189.6
Procurement Specialist	39.3	4.8	188.6
Tool Crib Attendant	27.8	9.6	266.9
Reactor Operator	34.8	38.4	1 336.3
Security Supervisor	55.9	4.8	268.3
Security Shift Supervisor	36.8	19.2	706.6
Security Patrolman	25.6	57.6	1 474.6
Contracts & Accounting Supervisor	47.1	5.1	240.2
Health & Safety Supervisor	60.5	5.1	308.6
Health Physicist	47.3	4.8	227.0
Protective Equipment Attendant	27.8	9.6	266.9
Industrial Safety Specialist	52.6	4.8	252.5
Quality Assurance Supervisor	52.6	5.1	268.3
Quality Assurance Engineer	47.3	4.8	227.0
Quality Assurance Technician	27.8	19.2	533.8
Consultant (Safety Review)	100.0	2.4	240.0
Instrument Technician	32.5	19.2	624.0
Maintenance Mechanic	32.5	19.2	624.0
Warehouseman	27.8	9.6	266.9
Subtotals		295.9	10 624.9
Decommissioning Workers			
Shift Engineer	52.4	9.6	503.0
Crew Leader	44.8	103.1	4 618.9
Utility Operator	32.5	224.5	7 296.2
Laborer	31.1	154.0	4 789.4
Craft Supervisor	47.3	19.2	908.2
Craftsman	32.5	146.7	4 767.8
Senior Health Physics Technician	39.5	19.2	758.4
Health Physics Technician	30.1	95.7	2 880.6
Subtotals		772.0	26 522.5
Totals		1 067.9	37 147.4

(a) Data from Table I.1-1 of Appendix I.
(b) Data from Table K.5-8
(c) Number of figures shown is for computational accuracy only.

Waste Category	Estimated Mass (kg)	Est Radio Conte	Burial Site Costs(d,e) State Surcharge ($)	Liner Surcharge ($)	Curie Surcharge ($)	Total Waste Management Costs ($)
Neutron-Activated Steels	242 930	6 0	1 110	74 000	613 920	1 558 600
Neutron-Activated Concrete	270 000	0	950			161 550
Contaminated Equipment	10 351 400	0	162 850			8 748 740
Contaminated Concrete	3 375 000	0	23 850			1 586 430
Compactible, Combustible Trash(f)	65 500	0	1 460			83 780
Compactible, Noncombustible Trash(f)	212 000	0	4 730			255 170
Noncompactible Trash	642 000	0	11 910			635 340
Solidifed Evaporator Bottoms	112 000	0	3 390	84 900		1 144 630
Neutralized Decontamination Solutions	150 000	0	1 590			90 710
Filter Cartridges & Spent Resins	10 000	0	300	4 110		98 550
Totals	15 430 830	~6 0	212 140	163 010	613 920	14 363 500

(a) Based on information from Table I.2-1 of Appendix I.
(b) Based on information from Table I.2-2 of Appendix I;
(c) Based on information from Table I.3-4 of Appendix I;
(d) Charges for individual shipments may vary depending o
(e) Based on information from Table I.4-1 of Appendix I.
(f) Values shown are for waste after treatment.

TABLE K.6-3. Estimated Costs of Radioactive Waste Management During BWR DECON Following a Scenario 2 Accident of changes in the number of workers needed to maintain compliance with individual radiation dose limits. Total staff labor costs for preparations for safe storage following a scenario 1 or scenario 3 accident are estimated to be about $12 million or $20 million, respectively.

K.6.2.2 Cost of Waste Management

Estimated costs of radioactive waste management for preparations for safe storage following a scenario 2 accident are shown in Table K.6-8. A waste management cost of about $1.8 million, representing about 7% of the total costs of preparations for safe storage following a scenario 2 accident, is estimated for this item. The waste management cost includes the container, transportation, and burial site costs, but does not include the cost of labor for removing and packaging the waste. (These labor costs are included in the costs of staff labor shown in Table K.6-7.)

The only wastes requiring offsite shipment during preparations for safe storage are the solidified evaporator bottoms, neutralized decontamination solutions, filter cartridges and spent resins, and the radioactive trash. Volumes of these wastes requiring packaging and disposal and waste management costs are estimated as described in Section K.6.1.2. The requirements and costs of waste management during preparations for safe storage are judged to be only slightly affected by changes in accident scenario.

K.6.2.3 Cost of Energy

The estimated cost of energy for preparations for safe storage following a scenario 2 accident is about $5.4 million, or about 21% of the total cost of preparations for safe storage following this accident. Energy costs are calculated as described in Section K.6.1.3 and vary slightly with accident scenario because of changes in the time requirement for preparations for safe storage.

K.6.2.4 Cost of Special Tools and Equipment

The estimated cost of special tools and equipment for preparations for safe storage following a scenario 2 accident is about $0.5 million, which represents about 2% of the total cost of preparations for safe storage

TABLE K.6-7. Estimated Costs of Staff Labor During Preparations for Safe Storage at the Reference BWR Following a Scenario 2 Accident

Position	Unit Cost for Labor[a] ($ thousands/ man-year)	Total Staff Labor Required[b] (man-years)	Total Staff Labor Costs[c] ($ thousands)
Management and Support Staff			
Decommissioning Superintendent	89.4	4.6	411.2
Secretary	24.4	11.7	285.5
Clerk	24.4	6.6	161.0
Decommissioning Engineer	76.2	4.6	350.5
Assistant Decommissioning Engineer	52.6	4.3	226.2
Radioactive Shipment Specialist	39.5	2.8	110.6
Procurement Specialist	39.3	2.8	110.0
Tool Crib Attendant	27.8	5.6	155.7
Reactor Operator	34.8	22.4	779.5
Security Supervisor	55.9	2.8	156.5
Security Shift Supervisor	36.8	11.2	412.2
Security Patrolman	25.6	33.6	860.2
Contracts & Accounting Supervisor	47.1	3.1	146.0
Health & Safety Supervisor	60.5	3.1	187.6
Health Physicist	47.3	2.8	132.4
Protective Equipment Attendant	27.8	5.6	155.7
Industrial Safety Specialist	52.6	2.8	147.3
Quality Assurance Supervisor	52.6	3.1	163.1
Quality Assurance Engineer	47.3	2.8	132.4
Quality Assurance Technician	27.8	11.2	311.4
Consultant (Safety Review)	100.0	1.4	140.0
Instrument Technician	32.5	11.2	364.0
Maintenance Mechanic	32.5	11.2	364.0
Warehouseman	27.8	5.6	155.7
Subtotals		176.9	6 418.7
Decommissioning Workers			
Shift Engineer	52.4	5.6	293.4
Crew Leader	44.8	36.4	1 630.7
Utility Operator	32.5	84.7	2 752.8
Laborer	31.1	40.9	1 272.0
Craft Supervisor	47.3	11.2	529.8
Craftsman	32.5	36.8	1 196.0
Senior Health Physics Technician	39.5	11.2	442.4
Health Physics Technician	30.1	21.8	656.2
Subtotals		248.6	8 773.3
Totals		425.5	15 192.0

(a) Data from Table I.1-1 of Appendix I.
(b) Data from Table K.5-13
(c) Number of figures shown is for computational accuracy only.

Total estimated SAFSTOR costs following a scenario 2 accident, including a 25% contingency, are about $104 million with a 30-year safe storage period and about $94 million with a 100-year safe storage period. Costs are in constant 1981 dollars, with no escalation for inflationary effects included. Preparations for safe storage are estimated to cost a total of about $32 million, or 31 to 34% of the total SAFSTOR costs. Annual continuing care costs during the safe storage period are estimated to be about $100,000, for cumulative totals of $3 million or $10 million for 30 years or 100 years, respectively, of safe storage. Deferred decontamination, representing the majority of the SAFSTOR costs, is estimated to require expenditures of about $69 million (in 1981 dollars) after a 30-year safe storage period and about $51 million after a 100-year safe storage period.

Although no detailed analyses are made of the costs of SAFSTOR following a scenario 1 or scenario 3 accident, the overall variation of SAFSTOR costs with accident severity can be approximated using information presented in the following subsections. SAFSTOR costs following a scenario 1 accident are estimated to total about $85 million with 30-year safe storage and about $78 million with 100-year safe storage. SAFSTOR costs following a scenario 3 accident are estimated to total about $138 million with 30-year safe storage and about $120 million with 100-year safe storage.

Information pertaining to individual cost categories is given in the following subsections.

K.6.2.1 Cost of Staff Labor

The costs of utility staff labor during preparations for safe storage following a scenario 2 accident are shown in Table K.6-7. These costs are based on the utility staff labor requirements described in Section K.5.3. A staff labor cost of about $15 million (without contingency), representing about 59% of the total cost of preparations for safe storage following the scenario 2 accident, is estimated for this item.

Based on the discussion of staff labor requirements in Section K.5.3, the costs of the management and support staff are not anticipated to be greatly affected by the postulated accident scenario. However, the decommissioning worker requirements are anticipated to change with accident scenario because

TABLE K.6-6. Summary of Estimated Costs of SAFSTOR at the Reference BWR Following a Scenario 2 Accident

Cost Category		Estimated Costs ($ millions)[a]	Percent of Total
Preparations for Safe Storage			
Staff Labor			
Management & Support Staff	6.419		
Decommissioning Workers	8.773		
Total Staff Labor Costs		15.192	58.7
Waste Management		1.766	6.8
Energy		5.426	21.0
Special Tools & Equipment		0.456	1.8
Miscellaneous Supplies		1.576	6.1
Specialty Contractors		0.229	0.9
Nuclear Insurance & License Fees		1.238	4.8
Subtotal		25.883	100.1[b]
Contingency (25%)		6.471	
Total, Preparations for Safe Storage Costs		32.354	
Annual Continuing Care Costs		0.100	
Deferred Decontamination Costs			
After 30-Year Safe Storage		69.1	
After 100-Year Safe Storage		51.2	
Total SAFSTOR Costs			
With 30-Year Safe Storage		104.4	
With 100-Year Safe Storage		93.5	

(a) Costs are adjusted to early 1981; the number of significant figures is for computational accuracy only.
(b) Total does not equal 100 because individual percentages are rounded to the nearest one-tenth.

TABLE K.6-5. Estimated Costs of Nuclear Liability Insurance and License Fees During DECON at the Reference BWR Following a Scenario 2 Accident

Category	Unit Cost ($)	Total Cost ($)
Nuclear Liability Insurance	400 000/yr	2 000 000[(a)]
License Fees:[(b)]		
Facility License Amendment (Class IV)	12 300	24 600[(c)]
Routine Health, Safety, and Environmental Inspections	650/yr	3 250[(d)]
Routine Safeguards Inspections	11 800	23 600[(e)]
Total		2 051 450

(a) Prorated by quarters for the duration of the decommissioning project (i.e., ~5.0 years for DECON).
(b) From 10 CFR 170.
(c) Based on two license amendments: one to allow possession but not operation of the plant, obtained prior to decommissioning, and one to terminate the license following the completion of DECON.
(d) Based on annual inspections for the duration of the decommissioning project (i.e., five annual inspections for DECON).
(e) Based on having spent fuel onsite for approximately 2 years (i.e., two yearly fees charged).

K.6.2 Details of SAFSTOR Costs

The estimated costs of SAFSTOR at the reference BWR following a scenario 2 accident and the subsequent accident cleanup are summarized in Table K.6-6. Costs are included for the three phases of SAFSTOR:

- preparations for safe storage
- continuing care (i.e., the safe storage period)
- deferred decontamination.

TABLE K.6-4. Estimated Costs of Miscellaneous Supplies During DECON at the Reference BWR Following a Scenario 2 Accident

Item	Quantity	Total Cost[a] ($ Thousands)
Decontamination Chemicals (EDTA/Oxalic Acid/Citric Acid)	17 650 kg[b]	29
Ion Exchange Resins	6 000 kg[c]	45
Filters	Unspecified[d]	294
Protective Clothing	22 190 sets[e]	1 110
Cleaning Supplies	Unspecified[f]	720
Expendable Tools & Materials	Unspecified[g]	480
Total		2 678

(a) Rounded to the nearest $1000.
(b) Based on the requirements shown in Table H.5-2 of Reference 1, with the requirements for decontamination of the RRC and RWCU systems deleted.
(c) Estimated at 50% of the requirement shown in Table H.5-7 of Reference 1.
(d) Estimated at 50% of the requirement shown in Table I.3-10 of Reference 1.
(e) Estimated at two clothing changes per shift per decommissioning worker. One set of clothing can be laundered and used four times.
(f) Estimated at $150,000/yr.
(g) Estimated at $100,000/yr.

K.6.1.7 Cost of Nuclear Insurance and License Fees

The costs of nuclear liability insurance, for an assumed policy limit of $160 million carried through the DECON period, and fees charged for licensing services performed by the NRC during DECON are shown in Table K.6-5. Insurance and license fees total about $2 million, or about 3% of the cost of DECON following a scenario 2 accident, and are judged not to be significantly altered by changes in accident severity.

fuel oil annually (see Section K.4.1.3). A load factor of 75% is applied to these basic consumption rates to account for the decrease in energy consumption as decommissioning is completed in various areas of the plant.

K.6.1.4 Cost of Special Tools and Equipment

The estimated costs of special tools and equipment for DECON at the reference BWR following a scenario 2 accident is about $2.6 million, which represents about 4% of the cost of DECON following this accident. Requirements for special tools and equipment are assumed to be about the same for post-accident DECON as they are for DECON following normal shutdown, shown in Table I.3-9 of Reference 1. An adjustment factor of 1.3 is applied to the special tool and equipment costs shown in Reference 1 to adjust these costs to the early-1981 cost base. Costs of special tools and equipment for post-accident DECON are assumed not to vary significantly with accident scenario.

K.6.1.5 Cost of Miscellaneous Supplies

Expendable supplies for post-accident DECON at the reference BWR include decontamination chemicals, protective clothing, filters and ion exchange resins, cleaning supplies, and expendable tools. The estimated costs of these items are presented in Table K.6-4. The total cost of expendable supplies for post-accident DECON at the reference BWR is estimated to be about $2.7 million. This cost is assumed not to vary significantly with accident scenario.

K.6.1.6 Cost of Specialty Contractors

Specialty contractors are required to perform explosive work, temporary radwaste handling, and environmental monitoring. The requirements for specialty contractors for post-accident DECON are assumed to be about the same as they are for DECON following normal shutdown, shown in Table I.3-11 of Reference 1. An adjustment factor of 1.17 is applied to the costs shown in Reference 1 to adjust these costs to the early-1981 cost base resulting in an estimated cost of about $0.42 million. Specialty contractor costs for post-accident DECON are assumed not to vary significantly with accident scenario.

The neutron-activated materials are contained in the reactor pressure vessel, the vessel internal structures, and in the surrounding steel and concrete biological shield. A detailed breakdown of these wastes is given in Table I.3-3 of Reference 1, which provides the basis for the neutron-activated materials cost estimates developed here. During accident cleanup following a scenario 2 accident, the steam separator assembly and the top fuel guide are removed to facilitate defueling and are disposed of as part of the accident cleanup wastes. Thus, these materials are not included in the waste management estimates for DECON. To calculate waste management costs for neutron-activated steels and concrete, the basic information pertinent to post-accident DECON is taken from Table I.3-3 of Reference 1 and the costs are recalculated based on the cost estimating bases in Appendix I of this study.

A detailed breakdown of contaminated materials in the reference BWR is given in Table I.3-4 of Reference 1, which provides the basis for the contaminated materials cost estimates developed here. Adjustments are made to the contaminated materials volumes in Reference 1 to account for materials removed during accident cleanup and for the additional contaminated concrete resulting from the accident. Waste management costs are recalculated based on the cost estimating bases in Appendix I of this study.

Radioactive wastes include solidified evaporator bottoms, neutralized decontamination solutions, filter cartridges and spent resins, and radioactive trash. Volumes of evaporator bottoms, decontamination solutions, and filter cartridges and spent resins requiring disposal are estimated on the basis of information given in Section H.5 of Reference 1. Volumes of radioactive trash are estimated using the bases and assumptions in Section H.1.2.3 of this study. Waste management costs are calculated using the cost estimating bases in Appendix I of this study.

K.6.1.3 Cost of Energy

The estimated cost of energy for DECON at the reference BWR following a scenario 2 accident is about $9 million, which represents about 14% of the total cost of DECON following the scenario 2 accident. Energy costs are estimated on the basis that energy consumption during cold shutdown of the reference BWR includes about 40,000 MWh of electricity and about 6,000 m^3 of

Waste Category	Estimated Mass (kg)	Estim Radioac Content	Burial Site Costs(d,e) State Surcharge ($)	Liner Surcharge ($)	Curie Surcharge ($)	Total Waste Management Costs ($)
Compactible, Combustible Trash(f)	29 000		650			38 000
Compactible, Noncombustible Trash(f)	93 500		2 090			112 810
Noncompactible Trash	284 000		5 270			281 300
Solidified Evaporator Bottoms	112 000	1	3 390	84 900		1 144 630
Neutralized Decontamination Solutions	150 000	.	1 590			90 710
Filter Cartridges & Spent Resins	10 000		300	4 110		98 550
Totals	678 500	1	13 290	89 010		1 766 000

(a) Based on information from Table I.2-1 of Appendix I.
(b) Based on information from Table I.2-2 of Appendix I; as
(c) Based on information from Table I.3-4 of Appendix I; in
(d) Charges for individual shipments may vary depending on
(e) Based on information from Table I.4-1 of Appendix I.
(f) Values shown are for waste after treatment.

ABLE K.6-8. Estimated Costs of Radioactive Waste Management During BWR Preparations for Safe Storage Following a Scenario 2 Accident following this accident. Requirements for special tools and equipment for post-accident preparations for safe storage are assumed to be about the same as they are for preparations for safe storage following normal shutdown, shown in Table J.5-6 of Reference 1. An adjustment factor of 1.3 is applied to the tool and equipment cost shown in Reference 1 to adjust these costs to the early-1981 cost base. Costs of special tools and equipment for post-accident preparations for safe storage are assumed not to vary significantly with accident scenario.

K.6.2.5 Cost of Miscellaneous Supplies

Estimated costs of miscellaneous supplies for preparations for safe storage following a scenario 2 accident are shown in Table K.6-9. Costs for these supplies are about $1.6 million and represent about 6% of the total cost of preparations for safe storage following a scenario 2 accident. The cost of miscellaneous supplies is assumed not to vary significantly with accident scenario.

K.6.2.6 Cost of Specialty Contractors

The requirements for specialty contractors for post-accident preparations for safe storage are assumed to be about the same as they are for preparations for safe storage following normal shutdown, shown in Table J.5-8 of Reference 1. The specialty contractor costs in Reference 1 are multiplied by a factor of 1.17 to adjust them to the early-1981 cost base, resulting in an estimated cost of about $0.23 million. Specialty contractor costs for post-accident preparations for safe storage are assumed not to vary significantly with accident scenario.

K.6.2.7 Cost of Nuclear Insurance and License Fees

The estimated costs of nuclear liability insurance and license fees during preparations for safe storage are shown in Table K.6-10. These costs total about $1.2 million (about 5% of the total cost of preparations for safe storage) and do not vary with changes in accident severity.

TABLE K.6-9. Estimated Costs of Miscellaneous Supplies During Preparations for Safe Storage at the Reference BWR Following a Scenario 2 Accident

Item	Quantity	Total Cost[(a)] ($ Thousands)
Decontamination Chemicals (EDTA/Oxalic Acid/Citric Acid)	17 650 kg[(b)]	29
Ion Exchange Resins	6 000 kg[(c)]	45
Filters	Unspecified[(d)]	294
Protective Clothing	10 160 sets[(e)]	508
Cleaning Supplies	Unspecified[(f)]	420
Expendable Tools & Materials	Unspecified[(g)]	280
Total		1 576

(a) Rounded to the nearest $1000.
(b) Based on the requirements shown in Table H.5-2 of Reference 1, with the requirements for decontamination of the RRC and RWCU systems deleted.
(c) Estimated at 50% of the requirement shown in Table H.5-7 of Reference 1.
(d) Estimated at 50% of the requirement shown in Table I.3-10 of Reference 1.
(e) Estimated at two clothing changes per shift per decommissioning worker. One set of clothing can be laundered and used four times.
(f) Estimated at $150,000/yr.
(g) Estimated at $100,000/yr.

K.6.2.8 Cost of Continuing Care During Safe Storage

The cost of continuing care during safe storage at the reference BWR is judged to be substantially unaffected by whether or not the reactor has experienced an accident. Thus, the estimated annual costs of continuing care are updated from those presented in Section J.5.2 of Reference 1, and are shown in Table K.6-11. The total estimated annual cost during the safe storage period (expressed in constant 1981 dollars) is approximately $100,000.

TABLE K.6-10. Estimated Costs of Nuclear Liability Insurance and License Fees During Preparations for Safe Storage at the Reference BWR Following a Scenario 2 Accident

Category	Unit Cost ($)	Total Cost ($)
Nuclear Liability Insurance	400 000/yr	1 200 000[a]
License Fees[b]		
Facility License Amendment (Class IV)	12 300	12 300[c]
Routine Health, Safety, & Environmental Inspections	650/yr	1 950[d]
Routine Safeguards Inspections	11 800/yr	23 600[e]
		1 237 850

(a) Prorated by quarters for the duration of the decommissioning project (i.e., ~3.0 years for preparations for safe storage).
(b) From 10 CFR 170.
(c) Based on one license amendment obtained prior to decommissioning to allow possession but not operation of the plant.
(d) Based on annual inspections for the duration of the decommissioning project (i.e., three annual inspections for preparations for safe storage).
(e) Based on having spent fuel onsite for approximately 2 years (i.e., two yearly fees charged).

K.6.2.9 Cost of Deferred Decontamination to Terminate SAFSTOR

As discussed in Section H.2.3 of Appendix H, the cost of deferred decontamination at the reference BWR is 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 that precedes SAFSTOR achieves the objectives presented in Section K.3. Based on this assumption, a comparison of the costs of DECON and deferred decontamination, both following normal shutdown and

TABLE K.6-11. Estimated Annual Costs During Safe Storage of the Reference BWR

Cost Category	Estimated Annual Costs ($ Thousands)
Labor	54.98[a]
Equipment & Supplies	1.30[b]
Annual Allowance for Repairs	6.50[b]
Utilities & Services	8.50[c]
License Fee	0.65[d]
Nuclear Liability Insurance	10.00[e]
Subtotal	81.93
Contingency (25%)	20.48
Total	102.41

(a) Updated from Table J.5-11 of Reference 1 by a factor of 1.2 (see Section I.1 of Appendix I).
(b) Updated from Table J.5-11 of Reference 1 by a factor of 1.3.
(c) Updated from Table J.5-11 of Reference 1 by the ratio of electricity costs reported in Section I.6 of this report and in Section M.6 of Reference 1.
(d) From 10 CFR 170.
(e) Assumed cost.

following a scenario 2 accident, is presented in Table K.6-12. The information about decontamination costs following normal shutdown is taken from Table J.7-3 of Reference 1. The estimated cost of deferred decontamination following a scenario 2 accident is about $69 million after 30 years of safe storage and about $51 million after 100 years of safe storage. Applying the same assumption to SAFSTOR following the other accident scenarios considered in this study results in deferred decontamination costs of $54 million and $40 million after 30 and 100 years, respectively, of safe storage following a scenario 1 accident, and $97 million and $72 million after 30 and 100 years, respectively, of safe storage following a scenario 3 accident.

TABLE K.6-12. Comparison of DECON and Deferred Decontamination Costs for the Reference BWR

Decommissioning Activity	Costs Following Normal Shutdown[(a)] Cost[(b)] ($ millions)	Percent of DECON Cost	Estimated Cost Following a Scenario 2 Accident[(b,c)] ($ millions)
DECON	43.6	100	85.3
Deferred Decontamination After 30 years	35.5	81	69.1
Deferred Decontamination After 100 years	26.3	60	51.2

(a) From Table J.7-3 of Reference 1.
(b) Costs include a 25% contingency.
(c) From Table K.6-1 of this study.

K.6.3 Details of ENTOMB Costs

The estimated cost of ENTOMB at the reference BWR following a scenario 2 accident and the subsequent accident cleanup campaign is summarized in Table K.6-13. The total estimated cost of ENTOMB following a scenario 2 accident, including a 25% contingency, is about $67 million. Corresponding costs following a scenario 1 or scenario 3 accident are estimated to be about $52 million or $93 million, respectively. Continuing care of the entombed plant is estimated to cost about $50,000 annually, independent of the accident that is postulated to have occurred. Information about individual cost categories is given in the following subsections.

K.6.3.1 Cost of Staff Labor

The costs of utility staff labor for DECON following a scenario 2 accident are shown in Table K.6-14. These costs are based on the utility staff labor requirements described in Section K.5.4. A total staff labor cost of about $30 million (without contingency) is estimated for ENTOMB following a scenario 2 accident. This cost represents about 57% of the total entombment cost.

TABLE K.6-13. Summary of Estimated Costs of ENTOMB at the Reference BWR Following a Scenario 2 Accident

Cost Category	Estimated Costs ($ millions)[a]		Percent of Total
Staff Labor			
Management & Support Staff	9.784		
Decommissioning Workers	20.713		
Total Staff Labor Costs		30.497	57.1
Waste Management			
Neutron-Activated Materials	1.480		
Contaminated Materials	3.568		
Radioactive Wastes	2.181		
Total Waste Management Costs		7.229	13.5
Energy		8.528	16.0
Special Tools & Equipment		2.621	4.9
Miscellaneous Supplies		2.481	4.6
Specialty Contractors		0.201	0.4
Nuclear Insurance & License Fees		1.839	3.4
Subtotal		53.396	99.9[b]
Contingency (25%)		13.349	
Total		66.745	
Annual Continuing Care Costs		0.050	

(a) Costs are adjusted to early 1981; the number of significant figures is for computational accuracy only.
(b) Total does not equal 100 because individual percentages are rounded to the nearest one-tenth.

TABLE K.6-14. Estimated Costs of Staff Labor During ENTOMB at the Reference BWR Following a Scenario 2 Accident

Position	Unit Cost for Labor[a] ($ thousands/ man-year)	Total Staff Labor Required[b] (man-years)	Total Staff Labor Costs[c] ($ thousands)
Management and Support Staff			
Decommissioning Superintendent	89.4	6.2	554.3
Secretary	24.4	16.5	402.6
Clerk	24.4	9.8	239.1
Decommissioning Engineer	76.2	6.2	472.4
Assistant Decommissioning Engineer	52.6	5.9	310.3
Radioactive Shipment Specialist	39.5	4.4	173.8
Procurement Specialist	39.3	4.4	172.9
Tool Crib Attendant	27.8	8.8	244.6
Reactor Operator	34.8	35.2	1 225.0
Security Supervisor	55.9	4.4	246.0
Security Shift Supervisor	36.8	17.6	647.7
Security Patrolman	25.6	52.8	1 351.7
Contracts & Accounting Supervisor	47.1	4.7	221.4
Health & Safety Supervisor	60.5	4.7	284.4
Health Physicist	47.3	4.4	208.1
Protective Equipment Attendant	27.8	8.8	244.6
Industrial Safety Specialist	52.6	4.4	231.4
Quality Assurance Supervisor	52.6	4.7	247.2
Quality Assurance Engineer	47.3	4.4	208.1
Quality Assurance Technician	27.8	17.6	489.3
Consultant (Safety Review)	100.0	2.2	220.0
Instrument Technician	32.5	17.6	572.0
Maintenance Mechanic	32.5	17.6	572.0
Warehouseman	27.8	8.8	244.6
Subtotals		272.1	9 783.5
Decommissioning Workers			
Shift Engineer	52.4	8.8	461.1
Crew Leader	44.8	70.7	3 167.4
Utility Operator	32.5	175.2	5 694.0
Laborer	31.1	109.3	3 399.2
Craft Supervisor	47.3	17.6	832.5
Craftsman	32.5	129.7	4 215.2
Senior Health Physics Technician	39.5	17.6	695.2
Health Physics Technician	30.1	74.7	2 248.5
Subtotals		603.6	20 713.1
Totals		875.7	30 496.6

(a) Data from Table I.1-1 of Appendix I.
(b) Data from Table K.5-18
(c) Number of figures shown is for computational accuracy only.

Staff labor costs following a scenario 1 or scenario 3 accident are estimated to be about $20 million or $51 million, respectively, with the difference between the labor costs for the three accident scenarios attributable mainly to the number of decommissioning workers necessary to comply with individual radiation dose limitations to the workers (see Sections K.5.4.1 and K.5.4.3). Specialty contractor labor is included in the specialty contractor costs given below and is not included in the labor costs given here.

K.6.3.2 Cost of 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, transportation, and disposal at an offsite shallow-land burial facility. Estimated costs of radioactive waste management for ENTOMB following a scenario 2 accident are shown in Table K.6-15. A total cost of about $7.3 million is estimated for this activity, representing about 13% of ENTOMB costs. Waste management costs following a scenario 1 or scenario 3 accident are estimated to be about $6.8 million or $7.4 million, respectively.

A comparison of waste management requirements for ENTOMB shown in Table K.6-15 with those for DECON shown in Table K.6-3 illustrates the volumes and kinds of radioactive material that are entombed. (Compare also Table I.3-3 and Table K.3-3 of Reference 1.) In general, about 60% of the contaminated equipment and all of the activated and contaminated concrete are assumed to remain in the entombed structure. The reactor vessel is entombed, but the vessel internals, which contain long-lived activation products, are packaged and shipped offsite for disposal.

K.6.3.3 Cost of Energy

The estimated cost of energy for ENTOMB following a scenario 2 accident is about $8.5 million, representing 16% of ENTOMB costs. Energy costs are calculated as described in Section K.6.1.3 and vary by about $\pm$5% with accident scenario.

Waste Category	Estimated Mass (kg)	Estim Radioac Content	Burial Site Costs(d,e) State Surcharge ($)	Liner Surcharge ($)	Curie Surcharge ($)	Total Waste Management Costs ($)
Neutron-Activated Steels	85 230	6 550	1 030	68 000	611 940	1 480 190
Contaminated Equipment	5 077 500	—	66 460			3 568 100
Compactible, Combustible Trash(f)	5 700		1 270			72 270
Compactible, Noncombustible Trash(f)	184 000		4 100			222 480
Noncompactible Trash	558 000		10 360			552 410
Solidifed Evaporator Bottoms	112 000	1	3 390	84 900		1 144 630
Neutralized Decontamination Solutions	150 000		1 590			90 710
Filter Cartridges & Spent Resins	10 000		300	4 110		98 550
Totals	6 233 730	~6 600	88 500	157 010	611 940	7 229 340

(a) Based on information from Table I.2-1 of Appendix I.
(b) Based on information from Table I.2-2 of Appendix I; as
(c) Based on information from Table I.3-4 of Appendix I; ir
(d) Charges for individual shipments may vary depending on
(e) Based on information from Table I.4-1 of Appendix I.
(f) Values shown are for waste after treatment.

TABLE K.6-15. Estimated Costs of Radioactive Waste Management During BWR ENTOMB Following a Scenario 2 Accident

K.6.3.4 Cost of Special Tools and Equipment

The estimated cost of special tools and equipment for ENTOMB are assumed to be the same as those for DECON (see Section K.6.1.4). Costs of special tools and equipment are assumed not to vary significantly with accident scenario.

K.6.3.5 Cost of Miscellaneous Supplies

The costs of miscellaneous supplies for ENTOMB following the scenario 2 accident are given in Table K.6-16. The estimated cost of miscellaneous supplies for ENTOMB is about $2.5 million, which represents about 5% of the

TABLE K.6-16. Estimated Costs of Miscellaneous Supplies During ENTOMB at the Reference BWR Following a Scenario 2 Accident

Item	Quantity	Total Cost[a] ($ Thousands)
Decontamination Chemicals (EDTA/Oxalic Acid/Citric Acid)	17 650 kg[b]	29
Ion Exchange Resins	6 000 kg[c]	45
Filters	Unspecified[d]	294
Protective Clothing	20 260 sets[e]	1 013
Cleaning Supplies	Unspecified[f]	660
Expendable Tools & Materials	Unspecified[g]	440
Total		2 481

(a) Rounded to the nearest $1000.
(b) Based on the requirements shown in Table H.5-2 of Reference 1, with the requirements for decontamination of the RRC and RWCU systems deleted.
(c) Estimated at 50% of the requirement shown in Table H.5-7 of Reference 1.
(d) Estimated at 50% of the requirement shown in Table I.3-10 of Reference 1.
(e) Estimated at two clothing changes per shift per decommissioning worker. One set of clothing can be laundered and used four times.
(f) Estimated at $150,000/yr.
(g) Estimated at $100,000/yr.

total ENTOMB cost following a scenario 2 accident. Cost of miscellaneous supplies are judged not to vary significantly with changes in accident severity, within the range of accident scenarios considered in this study.

K.6.3.6 Cost of Specialty Contractors

The requirements for specialty contractors for post-accident ENTOMB are assumed to be about the same as they are for ENTOMB following normal shutdown, shown in Table K.3-7 of Reference 1. The specialty contractor costs in Reference 1 are multiplied by a factor of 1.17 to adjust them to the early-1981 cost base, resulting in an estimated cost of about $0.2 million. Specialty contractor costs for ENTOMB are assumed not to vary significantly with accident scenario.

K.6.3.7 Cost of Nuclear Insurance and License Fees

The estimated costs of nuclear liability insurance and license fees during ENTOMB are shown in Table K.6-17. These costs total about $1.8 million, or about 4% of the total cost of ENTOMB following the scenario 2 accident. Costs of nuclear liability insurance and license fees are assumed not to vary with changes in accident severity.

K.6.3.8 Cost of Continuing Care and Possible Deferred Decontamination

The costs of continuing care (i.e., maintenance and surveillance) of the entombed plant are assumed to be less than those of the reference BWR in safe storage. (See Section K.6.2.8 for the safe storage costs). As explained in Section H.3.8, annual continuing care costs following ENTOMB are likely to be about one-half of the corresponding costs during the safe storage period of SAFSTOR. Thus, the annual continuing care costs following ENTOMB at the accident-damaged reference BWR are estimated to total about $50,000. These costs are assumed to be unaffected by the severity of the accident, within the range of accident scenarios considered in this study.

Although no firm estimate of the cost of possible deferred decontamination of the entombment structure is made, this operation is anticipated to be an extensive, time-consuming, and costly project. There is less radioactive material to remove from the plant during deferred decontamination following ENTOMB than there is following preparations for safe storage, but the removal

TABLE K.6-17. Estimated Costs of Nuclear Liability Insurance and License Fees During ENTOMB at the Reference BWR Following a Scenario 2 Accident

Category	Unit Cost ($)	Total Cost ($)
Nuclear Liability Insurance	400 000/yr	1 800 000[a]
License Fees:[b]		
Facility License Amendment (Class IV)	12 300	12 300[c]
Routine Health, Safety, and Environmental Inspections	650/yr	3 250[d]
Routine Safeguards Inspections	11 800/yr	23 600[e]
Total		1 839 150

(a) Prorated by quarters for the duration of the decommissioning project (i.e., ~4.5 years for ENTOMB).
(b) From 10 CFR 170.
(c) Based on one license amendment prior to decommissioning to allow possession, but not operation of the plant.
(d) Based on annual inspections for the duration of the decommissioning project (i.e., five annual inspections for ENTOMB).
(e) Based on having spent fuel onsite for approximately 2 years (i.e., two yearly fees charged).

of this material is complicated by the necessity to break into the entombment structure. Therefore, the costs during deferred decontamination following ENTOMB are anticipated to be similar to those during deferred decontamination for SAFSTOR.[a] Following a postulated scenario 2 accident, deferred decontamination could add another $51 million to the estimated cost of ENTOMB (assuming 100 years of continuing care), bringing the total cost of this decommissioning alternative to about $122 million (in constant 1981 dollars).

(a) See Section K.3.9 of Reference 1.

K.7 SAFETY ASSESSMENT DETAILS

This section provides details of the safety impacts of post-accident cleanup and decommissioning at the reference BWR. 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 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.

Basic assumptions used to estimate the safety impacts of accident cleanup and decommissioning at the reference BWR are defined in Section 14.1 of Chapter 14. Those assumptions not specific to the PWR (assumption 2 relates specifically to the PWR) also serve as bases for the BWR safety analysis described in this section. Safety impacts of accident cleanup at the reference BWR are described in Section K.7.1. Safety impacts of decommissioning following accident cleanup are described in Section K.7.2.

K.7.1 Accident Cleanup Safety

This section contains a discussion of the safety impacts resulting from the accident cleanup activities at the reference BWR, described in Section K.3. Radiological safety impacts to the public are discussed in Section K.7.1.1. Occupational safety impacts of accident cleanup are discussed in Sections K.7.1.2 and K.7.1.3. Transportation safety impacts, both public and occupational, are addressed in Section K.7.1.4.

K.7.1.1 Public Safety Impacts of Accident Cleanup

Public safety impacts of accident cleanup include radiation doses to the public from routine cleanup activities and from postulated industrial accidents during cleanup, nonradiological impacts to the public from onsite activities, and safety impacts from offsite transportation of radioactive wastes. 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 discussed in Section K.7.1.3.

During accident cleanup, the routine cleanup tasks and the postulated industrial accidents can generate airborne radioactivity in the plant, primarily in the form of solid particulates and/or suspended liquid droplets. The airborne radionuclide concentration depends on the particular task or accident considered and on the corresponding radionuclide inventory at the location involved. Contamination control measures, where applied, and HEPA filters in plant ventilation systems reduce the levels of radioactivity in the air leaving the plant.

Calculations of radiation doses to the public from estimated releases of radioactivity during PWR accident cleanup are summarized in Tables 14.2-1, 14.2-2, and 14.2-3 of Chapter 14. The consequences of atmospheric releases of radioactivity during routine accident cleanup tasks 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 result in airborne releases of radioactivity are determined by calculating inhalation radiation doses to the maximum-exposed individual.

For the reference PWR, radiation doses to the public from routine post-accident cleanup operations are estimated to be 1 or 2 orders of magnitude below permissible radiation dose levels in unrestricted areas and within the range of annual radiation doses from normal background. 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 accident results in a first-year dose of about 0.4 rem and fifty-year dose of about 0.8 rem to the lung of the maximum-exposed individual for cleanup following a scenario 2 accident. In general, the calculated doses for accident cleanup following a scenario 1 accident are 1 or 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.

No analysis or detailed dose estimate of the radiological consequences to the public from BWR accident cleanup is made in this report. It is believed that public safety impacts of accident cleanup at the reference BWR are similar to those for accident cleanup at the reference PWR. Justifications for this assumption include the following:

1. The accident scenarios postulated for the two reference reactors are similar. The total amount of radioactivity released in a given accident is postulated to be the same for the reference BWR as it is for the reference PWR.
2. The same reference site and same population density distribution is assumed for the reference BWR and the reference PWR.
3. Radioactivity released from the fuel assemblies during the postulated BWR accidents is largely confined to the primary containment vessel and most accident cleanup operations are performed inside the containment vessel. The primary containment is located inside the BWR reactor building (see Figure K.1-2), which provides a second level of confinement for the radioactivity released during normal cleanup operations and postulated cleanup accidents.

K.7.1.2 Occupational Radiation Doses from Accident Cleanup

A summary of the estimated occupational radiation doses for accident cleanup at the reference BWR is given in Table K.7-1. Doses are based on postulated external gamma radiation dose rates in various areas of the plant during accident cleanup and on estimated staff labor requirements for completing the accident cleanup tasks. The total estimated occupational radiation doses during BWR accident cleanup are about 1490 man-rem following a postulated scenario 1 accident, about 4170 man-rem following a scenario 2

TABLE K.7-1. Summary of Estimated Occupational Radiation Doses from Accident Cleanup at the Reference BWR

Cleanup Activity	Estimated Total Occupational Doses[a] (man-rem) Scenario 1 Accident	Scenario 2 Accident	Scenario 3 Accident
Preparations for Cleanup	57	146	426
Cleanup in Radwaste Building	--[b]	--[b]	923
Processing of Contaminated Liquids	155	240	291
Decontamination of Reactor Bldg. and Containment Vessel	371	1 082	2 582
Defueling of the Reactor	500	1 242	4 291
Cleanup of Reactor Water Recirculation System	52	176	383
Support Operations	357	1 285	3 044
Totals	1 492	4 171	11 940

(a) Doses shown are external doses from gamma radiation; workers are assumed to use respiratory equipment as appropriate to protect against inhalation of radioactive materials.
(b) Not postulated to be required for this reactor accident.

accident, and almost 12,000 man-rem following a scenario 3 accident. These results do not include the radiation doses to transportation workers which are given in Section K.7.1.4.

The occupational dose estimates are based on the following assumptions: 1) personnel exposure to radiation is minimized by using temporary shielding, remote handling techniques, respiration equipment where 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. Although the radioactive materials that are the source of the radiation doses decay throughout the accident cleanup period, no credit is taken for this decay because the anticipated effect is minimal due to the 30-year half-life of the dominant radionuclide (^{137}Cs).

The occupational doses shown in Table K.7-1 are summarized from dose information for the performance of specific accident cleanup tasks given in Tables K.3-2, K.3-3, K.3-5, K.3-6, and K.3-7. The detailed analysis of occupational radiation doses from accident cleanup is presented in Section K.3 because the results of the analysis are needed to adjust manpower requirements to ensure compliance with individual radiation dose limitations of 5 rem/year.[5] The results of these analyses are summarized here.

K.7.1.3 Industrial Safety Impacts of Accident Cleanup

Industrial safety impacts of accident cleanup include potential injuries and fatalities resulting from industrial accidents among the cleanup workers. Estimated casualties are calculated by finding the products of the frequencies of injuries and fatalities during various categories of work and the estimated worker time applied to each work category. Estimated worker injuries and fatalities during accident cleanup at the reference BWR following each of the three postulated reactor accidents considered in this study are summarized in Table K.7-2. 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 from industrial accidents appear to be unlikely during accident cleanup.

Frequency estimates for injuries and fatalities during accident cleanup are based on data collected by the U.S. AEC for the period 1943-1970.[6] The applicable staff man-hours used to estimate the potential injuries and fatalities are assumed to be the exposure hours given in Section K.3 for the various accident cleanup tasks and are divided into three categories of accident potential.[7] The category with the highest potential impact, heavy construction, is not applicable to accident cleanup. The next category, light construction, primarily involves reactor defueling, installation of equipment, and other miscellaneous construction and maintenance tasks. The remainder of the accident cleanup activities are categorized as equivalent to operational support.

TABLE K.7-2. Estimated Occupational Lost-Time Injuries and Fatalities During Accident Cleanup at the Reference BWR

Accident-Potential Category	Frequency (Accidents/man-hr)		Accident Scenario 1			Accident Scenario 2			Accident Scenario 3		
	Lost-Time Injuries	Fatalities	man-hr[a]	Lost-Time Injuries	Fatalities	man-hr[b]	Lost-Time Injuries	Fatalities	man-hr[c]	Lost-time Injuries	Fatalities
Heavy Construction	10×10^{-6}	4.2×10^{-8}	NA[d]	--	--	NA[d]	--	--	NA[d]	--	--
Light Construction	5.4×10^{-6}	3.0×10^{-8}	7.2×10^{4}	0.39	2.2×10^{-3}	1.4×10^{5}	0.76	4.2×10^{-3}	3.4×10^{5}	1.8	1.0×10^{-2}
Operational Support	2.1×10^{-6}	2.3×10^{-8}	7.0×10^{4}	0.15	1.6×10^{3}	1.3×10^{5}	0.27	3.0×10^{-3}	2.5×10^{5}	0.5	5.8×10^{-3}
Totals			1.4×10^{5}	0.54	3.8×10^{-3}	2.7×10^{5}	1.0	7.2×10^{-3}	5.9×10^{5}	2.3	1.6×10^{-2}

(a) Summarized from Table K.3-5.
(b) Summarized from Table K.3-6.
(c) Summarized from Tables K.3-3 and K.3-7.
(d) Heavy construction is not applicable to accident cleanup activities.

K.7.1.4 Transportation Safety Impacts of Accident Cleanup

Radioactive waste materials that result from accident cleanup are assumed to be shipped offsite to appropriate repositories as part of the planned accident cleanup activities. The potential safety impacts from the transportation of this material are as follows:

- radiation doses from the radioactive materials to transport workers and to members of the public along the transportation routes
- radiation doses to the maximum-exposed individual from accidental atmospheric releases during the transportation accidents
- injuries and fatalities resulting from transportation accidents.

The safety impacts of radioactive material transportation during accident cleanup at the reference BWR are discussed in this section.

Radioactive waste materials resulting from accident cleanup are assumed to be shipped by truck to either a shallow-land burial ground or a federal repository, either of which is assumed to be located 1600 km from the reference BWR. Spent fuel removed from the reactor during accident cleanup requires shipment to an ISFSI or to a federal repository, both of which are assumed to be located 1600 km from the reference BWR. Spent fuel shipments are assumed to be made by rail. The method used to estimate radiation doses to transportation workers and to members of the public along the transportation route is based on information in Reference 8. Radiation doses received by workers unloading the radioactive materials at the disposal site of the repository are not considered in this study since these doses are assumed to occur at separate licensed facilities.

The following assumptions are made about shipments of radioactive materials:

1. Each truck or rail shipment contains enough radioactive material to result in the maximum radiation exposure rates allowable by regulations. Department of Transportation (DOT) regulations[9] set the following exposure limits:
   - 1000 mR/hr at 1 m from the external surface of any package transported in a closed vehicle

- 200 mR/hr at the external surface of the vehicle
- 10 mR/hr at any point 2 m from the vehicle
- 2 mR/hr at any normally occupied position in the vehicle.

2. For each truck shipment of radioactive waste, two truck drivers spend 24 hours inside the cab (with an exposure rate of 2 mR/hr) and 2 hours outside the cab at a distance of 2 m from the cargo (with an exposure rate of 10 mR/hr).

3. For each truck shipment of radioactive waste, two garagemen each spend 20 minutes at an average distance of 2 m from the truck payload (at an exposure rate of 10 mR/hr).

4. For each truck shipment, 20 onlookers from the general public each spend 3 minutes at an average distance of 2 m from the payload (at an exposure rate of 10 mR/hr).

5. All truck shipments maintain an average speed of 65 km/hr; thus, the cumulative dose to the public is 2.3×10^{-6} man-rem/km.

6. For rail shipments of spent fuel, two train brakemen are assumed to spend 10 minutes during each of 10 stops (one every 160 km) at an average distance of 1 m from the shipping cask (at an assumed exposure rate of 25 mR/hr).

7. The population density along the transportation corridors is 120 persons/km^2.

The estimated radiation doses from transportation activities during accident cleanup at the reference BWR are listed in Table K.7-3. The numbers of truck and rail shipments for each accident scenario are taken from information on waste management requirements and costs presented in Sections K.4.2 and K.4.3. The total estimated doses from transportation activities following a scenario 3 accident are 120 man-rem to transport workers and 11 man-rem to members of the public along the transportation routes. The corresponding doses following a scenario 1 or scenario 2 accident are approximately 25% or 45%, respectively, of those following a scenario 3 accident.

TABLE K.7-3. Estimated Radiation Doses from Routine Transportation Activities During Accident Cleanup at the Reference BWR

Activity/Group	Radiation Dose per Shipment[a] (man-rem)	Accident Scenario 1		Accident Scenario 2		Accident Scenario 3	
		Number of Shipments[b]	Radiation Dose (man-rem)	Number of Shipments[c]	Radiation Dose (man-rem)	Number of Shipments[d]	Radiation Dose (man-rem)
Truck Shipments							
Truck Drivers	1.4×10^{-1}	166	23	294	41	744	104
Garagemen	6.7×10^{-3}	166	1.1	294	2.0	744	5.0
Total Worker Dose			24		43		109
Onlookers	1.0×10^{-2}	166	1.7	294	2.9	744	7.4
General Public	3.7×10^{-3}	166	0.61	294	1.1	744	2.8
Total Public Dose			2.3		4.0		10
Rail Shipments							
Train Brakemen	8.3×10^{-2}	49	4.1	78	6.5	80	6.6
Onlookers	1.0×10^{-2}	49	0.49	78	0.78	80	0.80
General Public	3.7×10^{-3}	49	0.18	78	0.29	80	0.30
Total Public Dose			0.67		1.1		1.1
Totals							
Total Transport Worker Dose			28		50		120
Total Public Dose			3		5		11

(a) Based on one-way trips of 1600 km.
(b) From Table K.4-10.
(c) From Table K.4-11.
(d) From Table K.4-7 and Table K.4-12.

Transportation accidents during offsite shipment of radioactive materials from accident cleanup at the reference BWR have the potential to result in inadvertent releases of radioactivity and corresponding radiation doses to individuals near the accident location. Realistic "worst-case" accidents involving truck transport of accident cleanup wastes from PWR accident cleanup are discussed in Appendix J. An accident involving a Type B container that is broken open and subjected to fire is discussed in Section J.2.3.2. Accidents involving Type A containers subjected to similar accident impacts are discussed in Section J.3.3.2. Truck transportation accidents involving wastes from BWR accident cleanup are postulated to be similar in nature and to have similar consequences to those described in Appendix J for PWR accident cleanup. Therefore, no further analysis of truck accident consequences is given here.

As discussed in Section J.2.3.2 of Appendix J, spent fuel is transported in thick-walled containers designed to withstand all but the most severe, highly unusual types of accidents. For a release of spent fuel to occur during transport, radioactive material must leave both the fuel cladding (or, for damaged fuel, the fuel assembly canister) and the cask containment. Since the transportation of spent fuel is not unique to accident cleanup, and since the probabilities of accidents that lead to atmospheric releases of radionuclides during spent fuel transport are very low, no further analysis or dose calculations are presented in this study. A discussion of the impact of spent fuel transportation accidents on public safety is given in Reference 10.

As with any transportation activity, a certain potential for accidental injury or death exists from transportation accidents that occur during the shipment of accident cleanup wastes. A summary of casualties estimated to result during transportation activities for accident cleanup at the reference BWR is shown in Table K.7-4. Accident frequency data are taken from Reference 8. The number of truck and rail shipments for each accident scenario is taken from Sections K.4.2 and K.4.3. As shown in Table K.7-4, about 1.3 injuries and 0.076 fatalities are estimated to result from accident cleanup transportation activities following a scenario 3 accident. The corresponding values following a scenario 2 or scenario 1 accident are estimated to be lower by factors of about 2 or 4, respectively. In all cases,

TABLE K.7-4. Estimated Casualties from Transportation Accidents During Accident Cleanup at the Reference BWR

Transportation Category	Accident Frequency Data[a]: Accidents per Vehicle-km	Injuries per Accident	Fatalities per Accident	Number of Shipments	Total Round-Trip Distances[b] (km)	Transportation Casualties: Injuries	Fatalities
Accident Scenario 1							
Rail Transport	8.7×10^{-8}	2.7	0.2	49[c]	1.6×10^5	0.038	0.0028
Truck Transport	1.0×10^{-6}	0.51	0.03	166[c]	5.3×10^5	0.27	0.016
Totals					6.9×10^5	0.31	0.019
Accident Scenario 2							
Rail Transport	8.7×10^{-8}	2.7	0.2	78[d]	2.5×10^5	0.059	0.0044
Truck Transport	1.0×10^{-6}	0.51	0.03	294[d]	9.4×10^5	0.48	0.028
Totals					1.2×10^6	0.54	0.032
Accident Scenario 3							
Rail Transport	8.7×10^{-8}	2.7	0.2	80[e]	2.6×10^5	0.061	0.0045
Truck Transport	1.0×10^{-6}	0.51	0.03	744[e]	2.4×10^6	1.2	0.072
Totals					2.7×10^6	1.3	0.076

(a) Based on data presented in Reference 8.
(b) Assuming a 3200-km round-trip distance.
(c) From Table K.4-10.
(d) From Table K.4-11.
(e) From Table K.4-7 and Table K.4-12

casualties from truck transport are estimated to be much greater than those from rail transport because of the greater number of truck shipments and the higher incidence of truck accidents per vehicle-kilometer.

K.7.2 Decommissioning Safety

This section contains a discussion of the safety impacts from decommissioning activities at the reference BWR following completion of accident cleanup at the plant. Analyses are performed for decommissioning following a scenario 2 accident, and the effects of variations in the severity of the postulated reactor accident are discussed. Radiological safety impacts to the public are discussed in Section K.7.2.1. Radiological and nonradiological occupational safety impacts are discussed in Sections K.7.2.2 and K.7.2.3. Transportation safety impacts, both public and occupational, are addressed in Section K.7.2.4.

K.7.2.1 Public Safety Impacts of Post-Accident Decommissioning

Estimates of radiation doses to the maximum-exposed individual and to the population from releases of radioactivity during post-accident decommissioning at the reference PWR are given in Section 14.3.1 of Chapter 14. Similar estimates for normal-shutdown decommissioning at the reference PWR are given in Section 11.2.1 of Reference 11. A comparison of estimated public doses from routine releases of radioactivity during post-accident and normal shutdown decommissioning at the reference PWR shows that the doses are comparable and that they are a small fraction of the doses to be expected from natural background sources.

No analyses or detailed dose estimates of the radiological consequences to the public from post-accident decommissioning activities at the reference BWR are made in this study. Because post-accident decommissioning activities and requirements are generally similar to those for decommissioning following normal shutdown, it is believed that the public safety impacts of post-accident decommissioning would be comparable to the impacts of normal-shutdown decommissioning at the reference BWR, presented in Section 11.3 of Reference 1. Radiation doses from routine decommissioning following normal shutdown are shown in Reference 1 to be a small fraction of the normal background radiation dose and to be below permissible radiation dose levels in unrestricted areas. Radiation doses to the lung of the maximum-exposed individual from postulated decommissioning accidents that are estimated to have a medium (10^{-5}/yr to 10^{-2}/yr) or high (>10^{-2}/yr) frequency of occurrence are also estimated to be very small (first-year lung doses <1 x 10^{-4} rem).

K.7.2.2 Occupational Radiation Doses from Post-Accident Decommissioning

A summary of the estimated occupational radiation doses during DECON, preparations for safe storage, and ENTOMB at the reference BWR, following a scenario 2 accident and the subsequent accident cleanup, is given in Table K.7-5. Doses are based on postulated external gamma radiation dose rates in various areas of the plant during decommissioning and on estimated staff labor requirements for completing the decommissioning tasks. Basic assumptions used to make the dose estimates are the same as those discussed previously in Section K.7.1.2. Occupational doses for decommissioning in the reactor building and the containment vessel are summarized from dose information presented in Tables K.5-5, K.5-10, and K.5-15 of Section K.5. Since the requirements for decommissioning in other plant areas are assumed to

TABLE K.7-5. Summary of Estimated Occupational Radiation Doses from Decommissioning at the Reference BWR following a Scenario 2 Accident

	Estimated Total Occupational Doses[a] (man-rem)		
Area Being Decommissioned	DECON	Preparations for Safe Storage	ENTOMB
Reactor Building/Primary Containment	2204	587	1562
Turbine Generator Building	193	18	195
Radwaste & Control Building	530	99	521
Ancillaries	254	113	253
Totals	3181	817	2531

(a) Doses shown are external doses from gamma radiation; workers are assumed to use respiratory equipment as appropriate to protect against inhalation of radioactive materials.

be approximately the same for post-accident and for normal-shutdown decommissioning, occupational doses for decommissioning in other buildings are summarized from Tables I.5-1, J.6-1, and K.4-1 of Reference 1.

The total estimated occupational doses during BWR decommissioning following a scenario 2 accident are about 3180 man-rem for DECON, about 820 man-rem during preparations for safe storage, and about 2530 man-rem for ENTOMB. Most of the occupational radiation dose from decommissioning (approximately two-thirds) results from activities in the reactor building and the containment vessel. The doses shown in Table K.7-5 are estimated to be increased by a factor of about 2 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 transportation workers, which are given in Section K.7.2.4.

Except for DECON, which on completion results in unrestricted release of the facility, the initial phase of decommissioning is followed by a period of continuing care and possible eventual deferred decontamination. Table K.7-6 contains a summary of the estimated total occupational radiation doses for all phases of SAFSTOR, assuming continuing care periods of 30 and 100 years. Total occupational doses for SAFSTOR at the reference BWR are estimated to be about 2480 man-rem with 30 years of continuing care and about 1260 man-rem

TABLE K.7-6. Summary of Estimated Occupational Radiation Doses During All Phases of SAFSTOR

SAFSTOR Phase	Estimated Occupational Doses[a] (man-rem) With Safe Storage Period of: 30 Years	100 Years
Preparations for Safe Storage	817	817
Continuing Care	65	120
Deferred Decontamination	1600	320
Totals	2482	1257

(a) Doses shown are external doses from gamma radiation; workers are assumed to use respiratory equipment as appropriate to protect against inhalation of radioactive materials.

with 100 years of continuing care. The dominant isotope that is postulated to control the rate of radioactive decay during these time periods is ^{137}Cs.

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. It is further assumed that deferred decontamination following ENTOMB is similar in level of effort and occupational radiation dose to deferred decontamination for SAFSTOR. Based on these assumptions, the total occupational radiation doses resulting from ENTOMB at the reference BWR following a scenario 2 accident are expected to be about 2900 man-rem, based on a retention period of 100 years for the entombment structure.

K.7.2.3 Industrial Safety Impacts of Post-Accident Decommissioning

Table K.7-7 contains a summary of estimated worker injuries and fatalities during DECON, preparations for safe storage, or ENTOMB at the reference BWR following a scenario 2 accident. As shown in the table, nearly 2 lost-time injuries are 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 BWR 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 DECON, and no further estimates for deferred decontamination are made.

Frequency estimates for injuries and fatalities during decommissioning are based on data collected by the U.S. AEC for the period 1943-1970.[6] The applicable staff man-hours used to estimate the potential casualties are obtained as follows. For decommissioning in the reactor building and the

TABLE K.7-7. Estimated Occupational Lost-Time Injuries and Fatalities During Decommissioning at the Reference BWR Following a Scenario 2 Accident

Accident-Potential Category	Frequency (Accidents/man-hr)		DECON			Preparations for Safe Storage			ENTOMB		
	Lost-Time Injuries	Fatalities	man-hrs[a]	Lost-Time Injuries	Fatalities	man-hrs[b]	Lost-Time Injuries	Fatalities	man-hrs[c]	Lost-Time Injuries	Fatalities
Heavy Construction	10×10^{-6}	4.2×10^{-8}	1.1×10^{5}	1.1	4.6×10^{-3}	NA[d]	—	—	1.2×10^{5}	1.2	5.0×10^{3}
Light Construction	5.4×10^{-6}	3.0×10^{-8}	1.2×10^{5}	0.65	3.6×10^{-3}	2.4×10^{4}	0.13	7.2×10^{-4}	7.7×10^{4}	0.42	2.3×10^{-3}
Operational Support	2.1×10^{-6}	2.3×10^{-8}	5.2×10^{4}	0.11	1.2×10^{-3}	6.0×10^{4}	0.13	1.4×10^{-3}	5.4×10^{4}	0.11	1.2×10^{-3}
Totals			2.8×10^{5}	1.9	9.4×10^{-3}	8.4×10^{4}	0.26	2.1×10^{-3}	2.5×10^{5}	1.7	8.5×10^{-3}

(a) Summarized from Table K.5-5 of this study and Table I.4-1 of Reference 1.
(b) Summarized from Table K.5-10 of this study and Table J.6-1 of Reference 1.
(c) Summarized from Table K.5-15 of this study and Table K.4-1 of Reference 1.
(d) Heavy construction is not applicable to preparations for safe storage.

containment vessel, the applicable staff man-hours are assumed to be the exposure hours given in Section K.5 for the various decommissioning tasks in these structures. For decommissioning in the other structures of the reference BWR, the applicable staff man-hours are assumed to be the exposure hours for the various decommissioning tasks given in Tables I.4-1, J.6-1, and K.4-1 of Reference 1. The exposure hours are divided into three categories of accident potential:

- heavy construction--primarily includes large-scale removal of piping, equipment, and concrete
- light construction--includes minor removal tasks and some waste handling activities
- operational support--miscellaneous support activities required to complete the decommissioning.

K.7.2.4 Transportation Safety Impacts of Post-Accident Decommissioning

Radioactive wastes generated during post-accident decommissioning at the reference BWR are assumed to be shipped in exclusive-use trucks to a shallow-land burial site 1600 km from the reactor facility. The safety impacts of these 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.

The estimated radiation doses from transportation activities during DECON, preparations for safe storage, or ENTOMB at the reference BWR following a scenario 2 reactor accident are listed in Table K.7-8. The assumptions made to estimate radiation doses from transportation activities during decommissioning are the same as those for accident cleanup, discussed previously in Section K.7.1.4. 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 numbers of truck

TABLE K.7-8. Estimated Radiation Doses from Routine Truck Shipments of Radioactive Materials During Post-Accident Decommissioning at the Reference BWR

Group	Radiation Dose per Shipment[a] (man-rem)	DECON		Preparations for Safe Storage		ENTOMB	
		Number of Shipments[b]	Radiation Dose (man-rem)	Number of Shipments[c]	Radiation Dose (man-rem)	Number of Shipments[d]	Radiation Dose (man-rem)
Truck Drivers	1.4×10^{-1}	1149	160.0	163	23.0	529	74.0
Garagemen	6.7×10^{-3}	1149	7.7	163	1.1	529	3.5
Total Worker Dose			170		24		78
Onlookers	1.0×10^{-2}	1149	12.0	163	1.60	529	5.3
General Public	3.7×10^{-3}	1149	4.2	163	0.60	529	2.0
Total Public Dose			16		2.2		7.3

(a) Based on one-way trips of 1600 km.
(b) From Table K.6-3.
(c) From Table K.6-8.
(d) From Table K.6-15.

shipments for each decommissioning alternative are taken from information on waste management requirements presented in Sections K.6.1.2, K.6.2.2, and K.6.3.2.

The estimated total doses from transportation activities during DECON are about 170 man-rem to transport workers and about 16 man-rem to members of the public along the transportation route. The corresponding doses during preparations for safe storage and during ENTOMB are about 14% and 46%, 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.

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.

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. Impacts of transportation accidents during BWR post-accident decommissioning are expected to be similar to those for PWR post-accident decommissioning, described in Section J.3.3.2 of Appendix J. Two accidents involving combustible radioactive wastes in Type A packages are analyzed in Section J.3.3.2. Both accidents are expected to have a low frequency of occurrence. Waste packages of 1 curie each are assumed to rupture and burn, releasing 5×10^{-4} of the contained radioactivity. The severe accident is assumed to involve 40 waste packages and the minor accident 1 such package. The severe accident results in estimated first-year doses of 0.024 rem to the bone and 0.032 rem to the lung of the maximum-exposed individual. Corresponding fifty-year committed dose equivalents are estimated to be 0.19 rem to the bone and 0.064 rem to the lung. Doses from the minor accident are a factor of 40 less than those from the severe accident.

As discussed previously for accident cleanup, any transportation task has a certain potential for accidents that could result in accidental injury or death. A summary of casualties estimated to result from transportation activities during DECON, preparations for safe storage, or ENTOMB at the reference BWR following a scenario 2 accident is given in Table K.7-9. Accident frequency data are taken from Reference 8. The numbers of truck shipments for each decommissioning alternative are taken from waste shipment requirements described in Sections K.6.1.2, K.6.2.2, and K.6.3.2. As shown in Table K.7-9, about 1.9 injuries and 0.11 fatalities are estimated for DECON following a scenario 2 reactor accident. The corresponding values for preparations for safe storage and for ENTOMB are estimated to be lower by factors of about 7 and 2, respectively.

TABLE K.7-9. Estimated Casualties from Truck Accidents During Post-Accident Decommissioning at the Reference BWR

Decommissioning Alternative	Accident Frequency Data[a]			Number of Shipments	Total Round-Trip Distances[b] (km)	Transportation Casualties	
	Accidents per Vehicle-km	Injuries per Accident	Fatalities per Accident			Injuries	Fatalities
DECON	1.0×10^{-6}	0.51	0.03	1149[c]	3.7×10^{6}	1.9	0.11
Preparations for Safe Storage	1.0×10^{-6}	0.51	0.03	163[d]	5.2×10^{5}	0.26	0.016
ENTOMB	1.0×10^{-6}	0.51	0.03	529[e]	1.7×10^{6}	0.87	0.051

(a) Based on data presented in Reference 8.
(b) Assuming a 3200-km round-trip distance.
(c) From Table K.6-3.
(d) From Table K.6-8
(e) From Table K.6-15.

REFERENCES

1. 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.

2. WPPSS Nuclear Project No. 2 Final Safety Analysis Report, Washington Public Power Supply System, Richland, Washington, February 1977.

3. WPPSS Nuclear Project No. 2 Environmental Report, Operating License Stage, Washington Public Power Supply System, Richland, Washington, March 1977.

4. T. Rockwell, Reactor Shielding Design Manual, TID-7004, U.S. Atomic Energy Commission, Washington, D.C., March 1956.

5. U.S. Code of Federal Regulations, Title 10, Part 20, Section 101, "Radiation Dose Standards for Individuals in Restricted Areas," (10 CFR 20.101), U.S. Government Printing Office, Washington, D.C., 1981.

6. Operational Accidents and Radiation Exposure Experiences Within the U.S. AEC 1943-1970, WASH-1192, U.S. Atomic Energy Commission, Washington, D.C., 1971.

7. Method of Recording and Measuring Work Injury Experience, ANSI 216.1, American National Standards Institute, 1967.

8. Directorate of Regulatory Standards, Environmental Safety of Transportation of Radioactive Materials to and from Nuclear Power Plants, WASH-1238, U.S. Atomic Energy Commission, Washington, D.C., 1972.

9. U.S. Code of Federal Regulations, Title 49, Part 173, Section 393, "General Packaging and Shipping Requirements," (49 CFR 173.393), U.S. Government Printing Office, Washington, D.C., 1978.

10. Technology for Commercial Radioactive Waste Management, DOE/ET-0028, Vol. 4, Chapter 6.2, Pacific Northwest Laboratory for U.S. Department of Energy, Richland, Washington, May 1979.

11. 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.

NRC FORM 335 (7 77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. NUREG/CR-2601, Vol. 2
PNL-4247

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)
Technology, Safety and Costs of Decommissioning Reference Light Water Reactors Following Postulated Accidents
Appendices

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
E. S. Murphy, G. M. Holter

5. DATE REPORT COMPLETED

MONTH	YEAR
October	1982

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Battelle Pacific Northwest Laboratory
Richland, Washington 99352

DATE REPORT ISSUED

MONTH	YEAR
November	1982

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.
B1171

13. TYPE OF REPORT
Technical data base

PERIOD COVERED (Inclusive dates)

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less)

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.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS (This report)
Unclassified

20. SECURITY CLASS (This page)

21. NO. OF PAGES

22. PRICE
$

NRC FORM 335 (7-77)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FOURTH CLASS MAIL
POSTAGE & FEES PAID
USNRC
WASH. D. C.
PERMIT No G-67