MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES

3 4456 0369062 8

ORNL/TM-12249

ornl

OAK RIDGE NATIONAL LABORATORY

MARTIN MARIETTA

The Advanced Neutron Source Research and Development Plan

November 1992

Advanced Neutron Source

OAK RIDGE NATIONAL LABORATORY
CENTRAL RESEARCH LIBRARY
CIRCULATION SECTION
4500N ROOM 175
LIBRARY LOAN COPY
DO NOT TRANSFER TO ANOTHER PERSON
If you wish someone else to see this report, send in name with report and the library will arrange a loan.

MANAGED BY
MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

This report has been reproduced directly from the best available copy.

Available to DOE and DOE contractors from the Office of Scientific and Technical Information, P.O. Box 62, Oak Ridge, TN 37831; prices available from (615) 576-8401, FTS 626-8401.

Available to the public from the National Technical Information Service, U.S. Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161.

This report [illegible] an agency of the United States [illegible] nor any agency thereof [illegible] or implied, or [illegible] completeness [illegible] disclosed, [illegible] rights. Reference [illegible] by trade name, [illegible] constitute or imply its [illegible] Government [illegible] authors expressed [illegible] United States Government [illegible]

ORNL/TM-12249

ADVANCED NEUTRON SOURCE RESEARCH AND DEVELOPMENT PLAN

Principal Author

D. L. Selby

Contributing Authors

J. L. Anderson
G. L. Copeland
D. K. Felde
M. L. Gildner
R. M. Harrington
J. B. Hayter
A. T. Lucas
B. H. Montgomery
R. E. Pawel
C. C. Queen
B. A. Worley
G. T. Yahr
G. L. Yoder

November 30, 1992

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831
managed by
MARTIN MARIETTA ENERGY SYSTEMS, INC.
for the
U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-84OR21400

MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES

3 4456 0369062 8

TABLE OF CONTENTS

1. INTRODUCTION 1

2. RESEARCH AND DEVELOPMENT PROGRAM SUMMARY 4
2.1 RESEARCH AND DEVELOPMENT PROGRAM WBS LEVEL THREE TASKS 4
2.2 RESEARCH AND DEVELOPMENT PROGRAM LEVEL THREE COST AND SCHEDULE 4
2.3 RESEARCH AND DEVELOPMENT TEST PROGRAM 8
2.4 MAJOR PROJECT INTERFACES WITH THE RESEARCH AND DEVELOPMENT PROGRAM 8
2.5 RESEARCH AND DEVELOPMENT PROGRAM PROCESS 11
2.5.1 Identification of Major Research and Development Activity 11
2.5.2 Identification of Tasks Within an Activity 11
2.5.3 Development of Task Plan 17
2.5.4 Submittal of Initiating Document 17
2.5.5 Development of the Software Verification and Validation Plan . . . 17
2.5.6 Review of Test Facilities and Testing 17
2.5.7 Performance of Task 18
2.5.8 Documentation of Task Completion 18
2.6 RESEARCH AND DEVELOPMENT QUALITY ASSURANCE ACTIVITIES 19
2.6.1 Applicability of Quality Assurance to the Research and Development Program 19
2.6.2 Management of Quality Assurance Within the Research and Development Program 19
2.6.3 Performance of Research and Development Quality Assurance . . . 20
2.6.4 Research and Development Assessments 20

3. DESCRIPTIONS OF THE RESEARCH AND DEVELOPMENT ACTIVITIES AND THEIR ASSOCIATED TASKS 22
3.1 REACTOR CORE DEVELOPMENT—WBS 1.1.1 22
3.1.1 Methods Development 26
3.1.1.1 Justification for the Methods Development Task 26
3.1.1.2 Description of the Methods Development Task 26
3.1.2 Preconceptual Core Development 27
3.1.2.1 Justification of Preconceptual Core Development 27
3.1.2.2 Description of the Preconceptual Core Development Task 27
3.1.3 General Design Criteria Development Support 27
3.1.3.1 Justification for General Design Criteria Development Support 27
3.1.3.2 Description of the General Design Criteria Development Support Task. 28
3.1.4 Reactor Physics Support to Design 28
3.1.4.1 Justification for the Reactor Physics Support to Design Task 28

3.1.4.2 Description of the Reactor Physics Support to Design Task 28
3.1.5 Thermal-Hydraulic Support to Design 29
3.1.5.1 Justification for the Thermal-Hydraulic Support to Design Task 29
3.1.5.2 Description of the Thermal-Hydraulic Support Task . . . 29
3.2 FUEL DEVELOPMENT—WBS 1.1.2 30
3.2.1 Selection and Verification of Fuel and Cladding 30
3.2.1.1 Justification for the Selection and Verification of Fuel and Cladding Task 30
3.2.1.2 Description of the Selection and Verification of Fuel and Cladding Task 35
3.2.2 Capsule Irradiation Tests 35
3.2.2.1 Justification for Capsule Irradiation Tests Task 35
3.2.2.2 Description of the Capsule Irradiation Tests Task 35
3.2.3 Miniplate Irradiation Tests 36
3.2.3.1 Justification for the Miniplate Irradiation Tests Task . . . 36
3.2.3.2 Description of the Miniplate Irradiation Tests Task 36
3.2.4 Irradiation Damage Simulation Studies and Fuel Performance Modeling 36
3.2.4.1 Justification for Irradiation Damage Simulation Studies and Fuel Performance Modeling Task 36
3.2.4.2 Description of the Irradiation Damage Simulation Studies and Fuel Performance Modeling Task 36
3.2.5 Final Performance Report 37
3.2.5.1 Justification for the Final Performance Report Task . . . 37
3.2.5.2 Description of the Final Performance Report Task 37
3.2.6 Plate and Fuel Element Fabrication 37
3.2.6.1 Justification for the Plate and Fuel Element Fabrication Task 37
3.2.6.2 Description of the Plate and Fuel Element Fabrication Task 37
3.2.7 Dummy Plate and Element Fabrication 38
3.2.7.1 Justification for the Dummy Plate and Element Fabrication Task 38
3.2.7.2 Description of the Dummy Plate and Element Fabrication Task 38
3.2.8 Prototype Fuel Element Fabrication for Critical Experiments 38
3.2.8.1 Justification for Prototype Fuel Element Fabrication for Critical Experiment Task 38
3.2.8.2 Description of the Prototype Fuel Element Fabrication for Critical Experiment Task 38
3.2.9 Full-Sized Plate Irradiation 39
3.2.9.1 Justification for the Full-Sized Plate Irradiation 39
3.2.9.2 Description of the Full-Sized Plate Irradiation Task 39
3.2.10 Burnable Poison Selection and Testing 39
3.2.10.1 Justification for the Burnable Poison Selection and Testing Task 39
3.2.10.2 Description of the Burnable Poison Selection and Testing Task 39

3.2.11 Production Mode Fabrication Process Development 39
3.2.11.1 Justification for the Production Mode Fabrication Process Development Task 39
3.2.11.2 Description of the Production Mode Fabrication Process Development Task 40
3.3 CORROSION TESTS AND ANALYSES—WBS 1.1.3 40
3.3.1 Design/Procurement and Installation of Test Loop 41
3.3.1.1 Justification for the Design/Procurement and Installation of Test Loop Task 41
3.3.1.2 Description of the Design/Procurement and Installation of Test Loop Task 41
3.3.2 Performance of Loop Experiments 41
3.3.2.1 Justification for Performing the Loop Experiments Task 41
3.3.2.2 Description of the Performance of the Loop Experiments Task 42
3.3.3 Analysis of Experimental Data 45
3.3.3.1 Justification for the Analysis of Experimental Data Task 45
3.3.3.2 Description of the Analysis of Experimental Data Task . 45
3.3.4 Analysis of Advanced Test Reactor Oxide Data 45
3.3.4.1 Justification of the Analysis of Advanced Test Reactor Oxide Data Task 45
3.3.4.2 Description of the Analysis of Advanced Test Reactor Oxide Data Task 45
3.3.5 Comparison Tests Under Irradiation Conditions 45
3.3.5.1 Justification for Comparison Tests Under Irradiation Conditions Task 45
3.3.5.2 Description of Comparison Tests Under Irradiation Conditions Task 46
3.4 CORE FLOW TESTS—WBS 1.1.4 46
3.4.1 Thermal-Hydraulic Test Facility 46
3.4.1.1 Justification for the Thermal-Hydraulic Test Facility Task 46
3.4.1.2 Description of the Thermal-Hydraulic Test Facility Task 46
3.4.2 Natural Circulation Tests 47
3.4.2.1 Justification for the Natural Circulation Tests 47
3.4.2.2 Description of the Natural Circulation Tests 50
3.5 CONTROL CONCEPTS—WBS 1.1.5 50
3.5.1 Static Analyses of Control and Safety Poison Systems 50
3.5.1.1 Justification of Static Analyses of Control and Safety Poison Systems 50
3.5.1.2 Description of the Static Analyses of Control and Safety Poison Systems Task 51
3.5.2 Reactor Kinetics Analyses 54
3.5.2.1 Justification of the Reactor Kinetics Analyses 54
3.5.2.2 Description of the Reactor Kinetics Analyses Task 54
3.6 CRITICAL EXPERIMENTS—WBS 1.1.6 54
3.6.1 Development of an Experimental Plan for Critical Experiments .. 56

3.6.1.1 Justification for Development of an Experimental Plan for Critical Experiments 56
3.6.1.2 Description of the Development of an Experimental Plan for Critical Experiments Task 56
3.6.2 Preanalysis of Critical Experiments 56
3.6.2.1 Justification for Preanalysis of Critical Experiments 56
3.6.2.2 Description of the Preanalysis of Critical Experiments Task 69
3.6.3 Modifications of Facilities for Critical Experiments 59
3.6.3.1 Justification for Modifications of Facilities for Critical Experiments 59
3.6.3.2 Description of the Modifications of Facilities for Critical Experiments 59
3.6.4 Perform Critical Experiments 59
3.6.4.1 Justification for Critical Experiments Task 59
3.6.4.2 Description of Critical Experiments Task 59
3.6.5 Analysis of Critical Experiments 59
3.6.5.1 Justification of Analysis of Critical Experiments Task . . . 60
3.6.5.2 Descriptions of Analysis of Critical Experiments Task . . 60
3.7 MATERIAL DATA, STRUCTURAL TESTS, AND ANALYSIS—WBS 1.1.7 61
3.7.1 Structural Design Support 61
3.7.1.1 Justification for Structural Design Support 61
3.7.1.2 Description of the Structural Design Support Task 61
3.7.2 Fuel Plate Stability Tests and Analysis 63
3.7.2.1 Justification for the Fuel Plate Stability Tests and Analysis Task 63
3.7.2.2 Description of the Fuel Plate Stability Tests and Analysis Task 66
3.7.3 Fuel Plate Deflection Due to Thermal Expansion 67
3.7.3.1 Justification for the Fuel Plate Deflection Due to Thermal Expansion 67
3.7.3.2 Description of the Fuel Plate Deflection Due to Thermal Expansion Task 67
3.7.4 Vibration Tests and Analyses 67
3.7.4.1 Justification for Vibration Tests and Analyses Task 67
3.7.4.2 Description of the Vibration Tests and Analyses Task . . 68
3.7.5 Material Properties Evaluations 70
3.7.5.1 Justification for the Material Properties Evaluations Task 70
3.7.5.2 Description of the Material Properties Evaluations Task 70
3.7.6 ASME Code Case for 6061-T6 Aluminum 71
3.7.6.1 Justification for the ASME Code Case for the 6061-T6 Aluminum Task 71
3.7.6.2 Description of the ASME Code Case for 6061-T6 Aluminum Task 71
3.7.7 Materials Properties Data Base 72
3.7.7.1 Justification for the Materials Properties Data Base Task 72

3.7.7.2 Description of the Materials Properties Data Base Task 72
3.7.8 Fracture Assessment 72
3.7.8.1 Justification for the Fracture Assessment Task 72
3.7.8.2 Description of the Fracture Assessment Task 72
3.8 COLD SOURCE DEVELOPMENT—WBS 1.1.8 72
3.8.1 Neutronics Analysis 73
3.8.1.1 Justification for the Neutronics Analysis Task 73
3.8.1.2 Description of the Neutronics Analysis Task 73
3.8.2 Thermal Analysis 73
3.8.2.1 Justification for the Thermal Analysis Task 73
3.8.2.2 Description of the Thermal Analysis Task 77
3.8.3 Stress Analysis and Structural Design 77
3.8.3.1 Justification for the Stress Analysis and Structural Design Task 77
3.8.3.2 Description of the Stress Analysis and Structural Design Task 77
3.8.4 Cryogenic Model Verification Tests 78
3.8.4.1 Justification for the Cryogenic Model Verification Tests Task 78
3.8.4.2 Description of the Cryogenic Model Verification Tests Task 78
3.8.5 Prototype Development and Demonstration 78
3.8.5.1 Justification for the Prototype Development and Demonstration Task 78
3.8.5.2 Description of the Prototype Development and Demonstration Task 79
3.8.6 Cold Source Material Selection 79
3.8.6.1 Justification for Cold Source Material Selection Task . . 79
3.8.6.2 Description of the Cold Source Materials Selection Task 80
3.8.7 Instrumentation and Control 80
3.8.7.1 Justification for the Instrumentation and Control Task . 80
3.8.7.2 Description of the Instrumentation and Control Task . . 80
3.9 BEAM TUBE, GUIDE, AND INSTRUMENT DEVELOPMENT—WBS 1.1.9 80
3.9.1 Beam Transport System Development 81
3.9.1.1 Justification for the Beam Transport System Development Task 81
3.9.1.2 Description of the Beam Transport System Development Task 81
3.9.2 Polarizer Development 85
3.9.2.1 Justification for the Polarizer Development Task 85
3.9.2.2 Description of the Polarizer Development Task 85
3.9.3 Monochromator Development 86
3.9.3.1 Justification for the Monochromator Development Task 86
3.9.3.2 Description of the Monochromator Development Task . 86
3.9.4 Detector Development 86
3.9.4.1 Justification for the Detector Development Task 86

3.9.4.2 Description of Detector Development Task 87
3.9.5 Instrument Systems Development 87
3.9.5.1 Justification for the Instrument Systems Development Task 87
3.9.5.2 Description of the Instrument Systems Development Task 87
3.10 HOT SOURCE DEVELOPMENT—WBS 1.1.10 88
3.10.1 Neutronics Analyses 88
3.10.1.1 Justification for the Neutronics Analyses Task 88
3.10.1.2 Description of the Neutronics Analyses Task 89
3.10.2 Thermal Analyses 89
3.10.2.1 Justification for the Thermal Analyses Task 89
3.10.2.2 Description of the Thermal Analyses Task 92
3.10.3 Stress Analyses and Structural Design 92
3.10.3.1 Justification for the Stress Analyses and Structural Design Task 92
3.10.3.2 Description of the Stress Analyses and Structural Design Task 92
3.10.4 Prototype Development 92
3.10.4.1 Justification for the Prototype Development Task 92
3.10.4.2 Description of the Prototype Development Task 92
3.10.5 Prototype Demonstration 93
3.10.5.1 Justification for the Prototype Demonstration Task 93
3.10.5.2 Description of the Prototype Demonstration Task 93
3.11 NEUTRON AND GAMMA TRANSPORT AND SHIELDING—WBS 1.1.11 93
3.11.1 Beam Transport Analyses 93
3.11.1.1 Justification for the Beam Transport Analyses Task 93
3.11.1.2 Description of the Beam Transport Analyses Task 94
3.11.2 Impact of Reflector Components 97
3.11.2.1 Justification for the Impact of Reflector Components Task 97
3.11.2.2 Description of the Impact of Reflector Components Task 97
3.11.3 Analyses of Heat Loads and Activation 97
3.11.3.1 Justification for the Analyses of Heat Loads and Activation Task 97
3.11.3.2 Description of the Analyses of Heat Loads and Activation Task 98
3.11.4 Support to the Design of Shielding 98
3.11.4.1 Justification for the Support to the Design of Shielding Task 98
3.11.4.2 Description of the Support to the Design of Shielding Task 98
3.12 INSTRUMENTATION AND CONTROL SYSTEMS DEVELOPMENT . 99
3.12.1 Dynamic Model 99
3.12.1.1 Justification for the Dynamic Model Task 99
3.12.1.2 Description of the Dynamic Model Task 103
3.12.2 Safety Analysis Support 103
3.12.2.1 Justification for the Safety Analysis Support Task 103

3.12.2.2 Description of the Safety Analysis Support Task 103
3.12.3 Reactor Control 103
3.12.3.1 Justification for the Reactor Control Task 103
3.12.3.2 Description of the Reactor Control Task 104
3.12.4 Reactor Protection System 104
3.12.4.1 Justification for the Reactor Protection System Task 104
3.12.4.2 Description of the Reactor Protection System Task 104
3.12.5 Sensor Development 105
3.12.5.1 Justification for the Sensor Development Task 105
3.12.5.2 Description of the Sensor Development Task 105
3.12.6 Instrumentation Development 106
3.12.6.1 Justification for the Instrumentation Development Task . 106
3.12.6.2 Description of the Instrumentation Development Task .. 106
3.12.7 Actuator Development 106
3.12.7.1 Justification for the Actuator Development Task 106
3.12.7.2 Description of the Actuator Development Task 106
3.12.8 Diagnostics and Surveillance 107
3.12.8.1 Justification for the Diagnostics and Surveillance Task ... 107
3.12.8.2 Description of the Diagnostics and Surveillance Task ... 107
3.12.9 Experiment Systems 107
3.12.9.1 Justification for the Experiment Systems Task 107
3.12.9.2 Description of the Experiment Systems Task 107
3.12.10 Control Integration 108
3.12.10.1 Justification for the Control Integration Task 108
3.12.10.2 Description of the Control Integration Task 108
3.12.11 Subcriticality Monitoring 108
3.12.11.1 Justification for the Subcriticality Monitoring Task 108
3.12.11.2 Description of the Subcriticality Monitoring Task 108
3.13 FACILITY CONCEPTS 109
3.13.1 Reactor Components Test Facility 109
3.13.1.1 Justification for the Reactor Components Test Facility Task 109
3.13.1.2 Description of the Reactor Components Test Facility Task 109
3.13.2 Reflector Vessel and Core Flow Test 116
3.13.2.1 Justification for the Reflector Vessel and Core Flow Test Task 116
3.13.2.2 Description of the Reflector Vessel and Core Flow Test Task 116
3.13.3 Refueling Components Test Facility 117
3.13.3.1 Justification for the Refueling Components Test Facility Task 117
3.13.3.2 Description of the Refueling Components Test Facility Task 117
3.13.4 Special Facilities 118
3.13.4.1 Justification for the Special Facilities Task 118
3.13.4.2 Description of the Special Facilities Task 118
3.14 SAFETY RESEARCH AND DEVELOPMENT TESTS 119
3.14.1 Thermal-Hydraulic Experiments in Support of Design and Safety Analysis 120

3.14.1.1 Justification of Thermal-Hydraulic Tests in Support of Safety Analysis 120
3.14.1.2 Description of Thermal-Hydraulic Experiments in Support of Design and Safety Analysis Task 124
3.14.2 Accident-Related Fuel Testing and Severe Accident Tests 125
3.14.2.1 Justification for the Accident-Related Fuel Testing and Severe Accident Testing Task 125
3.14.2.2 Description of the Accident-Related Fuel Testing and Severe Accident Testing Task 128
REFERENCES 132

LIST OF FIGURES

1.1 Advanced Neutron Source level two work breakdown structure 2
2.1 Advanced Neutron Source Research and Development program level three work breakdown structure 5
2.2 Work breakdown structure 1.1 Research and Development program 6
2.3 Advanced Neutron Source planning schedule—summary 9
2.4 Test program schedule 12
2.5 Research and Development task flow chart 16
3.1 Summary of cost estimates for work breakdown structure 1.1.1 24
3.2 Schedule for work breakdown structure 1.1.1, reactor core development 25
3.3 Summary of cost estimates for work breakdown structure 1.1.2, fuel development 33
3.4 Schedule for work breakdown structure 1.1.2, fuel development 34
3.5 Summary of cost estimates for work break structure 1.1.3 43
3.6 Schedule for work breakdown structure 1.1.3, corrosion tests and analysis 44
3.7 Summary of cost estimates for work breakdown structure 1.1.4 48
3.8 Schedule for work breakdown structure 1.1.4, core flow tests 49
3.9 Summary of cost estimates for work breakdown structure 1.1.5 52
3.10 Schedule for work breakdown structure 1.1.5, control concepts 53
3.11 Summary of cost estimates for work breakdown structure 1.1.6 57
3.12 Schedule for work breakdown structure 1.1.6, critical experiments 58
3.13 Summary of cost estimates for work breakdown structure 1.1.7 64
3.14 Schedule for work breakdown structure 1.1.7, material data, structural tests, and analysis 65
3.15 Summary of cost estimates for work breakdown structure 1.1.8 75
3.16 Schedule for work breakdown structure 1.1.8, cold source development 76
3.17 Summary of cost estimates for work breakdown structure 1.1.9, beam tube, guide, and instrument development 83
3.18 Schedule for work breakdown structure 1.1.9, beam tube, guide, and instrument development 84
3.19 Summary of cost estimates for work breakdown structure 1.1.10 hot source development 90
3.20 Schedule for work breakdown structure 1.1.10, hot source development 91
3.21 Summary of cost estimates for work breakdown structure 1.1.11, neutron and gamma transport and shielding 95
3.22 Schedule for work breakdown structure 1.1.11, neutron and gamma transport and shielding 96
3.23 Summary of cost estimates for work breakdown structure 1.1.12, instrumentation and controls system development 101
3.24 Schedule for work breakdown structure 1.1.12, instrumentation and controls system development 102
3.25 Summary of cost estimates for work breakdown structure 1.1.13, reactor concepts 111
3.26 Schedule for work breakdown structure 1.1.13, reactor concepts 112
3.27 Summary of cost estimates for work breakdown structure 1.1.14, safety research and development 121
3.28 Schedule for work breakdown structure 1.1.14, safety research and development 122

LIST OF TABLES

2.1 Integrated Research and Development Program costs (FY 1993 through FY 2001) (all costs in 1992 dollars) 7
2.2 Test program activities 10
2.3 Research and Development Program critical milestone events 15
3.1 Summary description of reactor core development work breakdown structure level four tasks 23
3.2 Summary description of the fuel development work breakdown structure level four tasks 31
3.3 Summary description of the corrosion tests and analysis work breakdown structure level four tasks 42
3.4 Summary description of the core flow tests work breakdown structure level four tasks 47
3.5 Summary description of the control concepts work breakdown structure level four tasks 51
3.6 Summary description of the critical experiments work breakdown structure level four tasks 55
3.7 Summary description of the material data, structural tests, and analyses work breakdown structure level four tasks 62
3.8 Summary description of the cold source development work breakdown structure level four tasks 74
3.9 Summary description of the beam tube, guide, and instrument development task work breakdown structure level four tasks 82
3.10 Summary description of the hot source development work breakdown structure level four tasks 89
3.11 Summary description of the neutron and gamma transport and shielding level four tasks 94
3.12 Summary description of the instrumentation and controls system development work breakdown structure leevel four tasks 100
3.13 Summary description of the reactor concepts work breakdown structure level four tasks 110
3.14 Summary description of the safety research and development work breakdown structure level four tasks 120

ACRONYMS

1-D	one dimensional
2-D	two dimensional
3-D	three dimensional
ANS	Advanced Neutron Source
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor
CPBT	core pressure boundary tube
DNP	departure from nucleate boiling
ES&H	environmental, safety, and health
FIV	flow-induced vibration
FSAR	Final Safety Analysis Report
FY	fiscal year
HFIR	High Flux Isotope Reactor
ILL	Institut Laue Langevin
INEL	Idaho National Engineering Laboratory
MMES	Martin Marietta Energy Systems, Inc.
NQA	Quality Assurance Standards for Nuclear Facilities
ORNL	Oak Ridge National Laboratory
PSAR	Preliminary Safety Analysis Report
QA	quality assurance
R&D	research and development
V&V	verification and validation
WBS	work breakdown structure

1. INTRODUCTION

The Advanced Neutron Source (ANS) is being designed as a user-oriented neutron research laboratory centered around the most intense continuous beams of thermal and subthermal neutrons in the world (an order of magnitude more intense than beams available from the most advanced existing reactors). The ANS will be built around a new research reactor of ~330 MW fission power, producing an unprecedented peak thermal flux of $> 7 \times 10^{19}\ m^{-2} \cdot s^{-1}$. Primarily a research facility, the ANS will accommodate more than 1000 academic, industrial, and government researchers each year. They will conduct basic research in all branches of science—as well as applied research—leading to better understanding of new materials, including high temperature super conductors, plastics, and thin films. Some 48 neutron beam stations will be set up in the ANS beam rooms and the neutron guide hall for neutron scattering and for fundamental and nuclear physics research. There also will be extensive facilities for materials irradiation, isotope production, and analytical chemistry.

The top level work breakdown structure (WBS) for the project is shown in Fig. 1.1. As noted in this figure, one component of the project is a research and development (R&D) program (WBS 1.1). This program interfaces with all of the other project level two WBS activities. Because one of the project guidelines is to meet minimum performance goals without relying on new inventions, this R&D activity is not intended to produce new concepts to allow the project to meet minimum performance goals. Instead the R&D program will focus on the four objectives described in the following paragraphs:

Address feasibility issues. The tasks performed under this category involve analyses and tests necessary to show that the basic technical assumptions that have a major influence on the ANS design are credible. Some technical assumptions used in the early design process will be based on extrapolations of existing data but will be unproven for the unique ANS conditions. As the design progresses, the R&D program will evaluate the validity of these assumptions and provide feedback to the design process.

Provide analysis support. The unique nature of the ANS requires that special analysis techniques and models be developed and used to perform appropriate evaluations of system performance in certain areas (e.g., nuclear and thermal-hydraulic performance, structural behavior, cold source development). Tasks of this nature that are not normally performed by the engineering design or the safety analysis teams are performed within the R&D program.

Evaluate options for improvement in performance beyond minimum requirements. Although the use of inventions to meet minimum requirements is precluded, the potential for performance improvement due to the use of technology improvements is not ignored. Tasks are identified within the R&D program to address areas with high potential to provide significant improvements in facility performance. These R&D tasks provide the opportunity for big improvements from small research investments that would lower facility costs, lower operating costs and/or increase the usefulness of the facility. In addition, the gains obtained from these developments may apply to other facilities besides the ANS.

Provide prototype demonstrations for unique facilities. The ANS facility includes a number of one-of-a-kind components. Because some of these are considered to be unique, functional tests of the prototypes are necessary before their operation in the ANS facility. The tests of these prototypes are performed under the R&D program.

The remainder of this report presents (1) the process by which the R&D activities are controlled and (2) a discussion of the individual tasks that have been identified for the

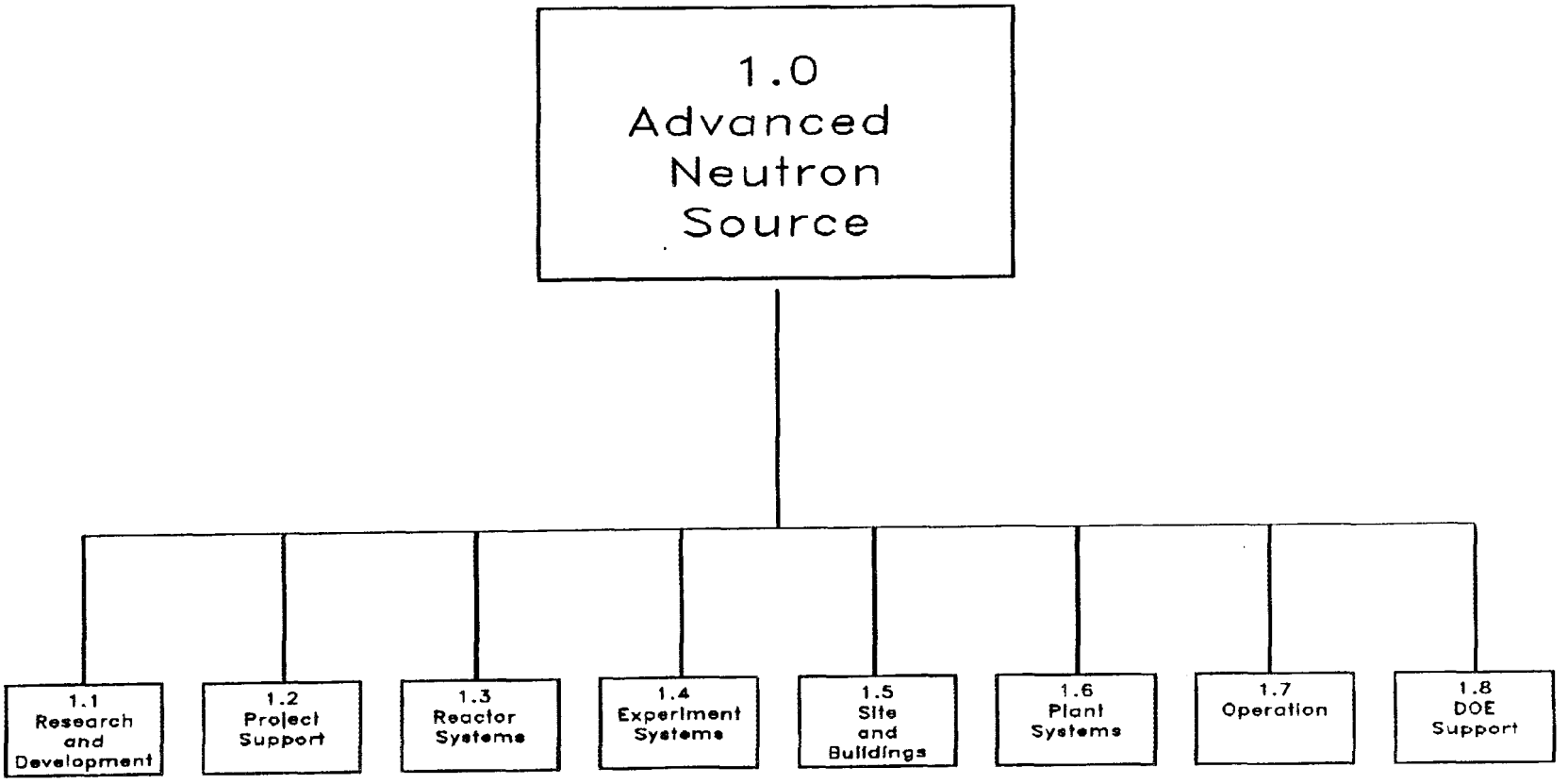

Fig. 1.1. Advanced Neutron Source level two work breakdown structure.

R&D program, including their justification, schedule and costs. The activities discussed in this report will be performed by Martin Marietta Energy Systems, Inc. (MMES) through the Oak Ridge National Laboratory (ORNL) and through subcontracts with industry, universities, and other national laboratories. It should be noted that in general a success path has been assumed for all tasks (i.e., no extra time has been allowed for unanticipated problems).

2. RESEARCH AND DEVELOPMENT PROGRAM SUMMARY

This section provides a summary of the R&D program. Subsection 2.1 identifies tasks at the third level of the WBS. Subsection 2.2 presents and discusses the costs and schedule associated with tasks. Subsection 2.3 summarizes the testing portion of the R&D program. Subsection 2.4 documents and discusses the major interfaces with other work areas of the ANS project. Subsection 2.5 presents and discusses the R&D management process. Finally, Subsect. 2.6 defines in more detail the R&D quality assurance (QA) activities alluded to in Subsect. 2.5.

2.1 RESEARCH AND DEVELOPMENT PROGRAM WBS LEVEL THREE TASKS

The R&D tasks have been grouped into the 14 WBS level three tasks that are identified in Fig. 2.1. Nine of these activities were part of an original R&D plan drafted during the early stages of the project in 1986. Shortly after this, in 1987, four additional tasks for the R&D program were identified—reactor core development (WBS 1.1.1); beam tube, guide, and instrument development (WBS 1.1.9); instrumentation and controls system development (WBS 1.1.12); and the reactor facility concepts (WBS 1.1.13). The present number of 14 tasks was reached in 1992 when a decision was made to move the safety-related tests into the R&D program, integrating the entire project test program. This new activity was designated safety-related R&D tests (WBS 1.1.14) and was scheduled to begin under the R&D Program in October 1992. A summary of the 14 tasks and their WBS level four subtasks is provided in Sect. 3 of this report.

2.2 RESEARCH AND DEVELOPMENT PROGRAM LEVEL THREE COST AND SCHEDULE

The R&D program schedule at WBS level three is shown in Fig. 2.2, and the total level three costs are presented in Table 2.1. Both the schedule and costs are separated into the three types of funding: expense, line-item, and capital equipment. Scoping analyses, parametric studies, code verification and validation (V&V), general data collection, and other concept development activities are performed using expense money. Analysis in direct support of Title I and Title II design, prototype development and testing, and documentation in support of items such as the Final Safety Analysis Report (FSAR) are performed using line-item money. Capital equipment money is used to (1) design and fabricate test facilities at ORNL that are not considered to be direct prototype tests and (2) purchase computers that are needed to perform analytical analyses and support experiment facilities. More detailed information on both schedule and costs is given in the level four subtask discussions presented in Sect. 3 of this report.

The schedule is driven by the need to meet major milestones (discussed in Subsect. 2.4) that are interfaces with other project activities. In developing this schedule, special efforts have been made to avoid making aspects of the R&D plan part of the project's critical path because the nature of the R&D activities makes their schedule more uncertain than that of the general engineering design activities. Although the R&D program (as shown in Fig. 2.2) extends into FY 2001, 80% of the work identified within

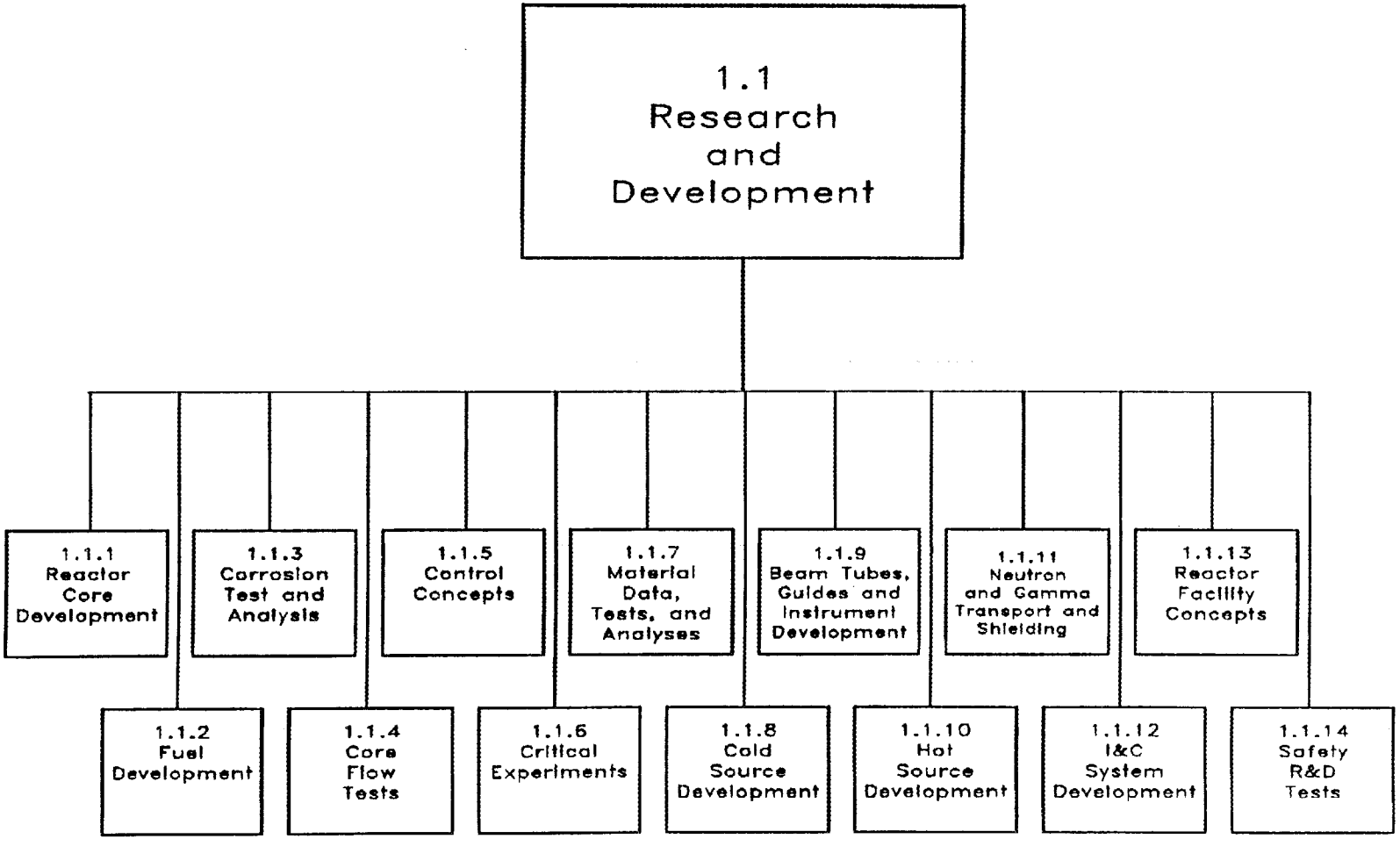

Fig. 2.1. Advanced Neutron Source Research and Development program level three work breakdown structure.

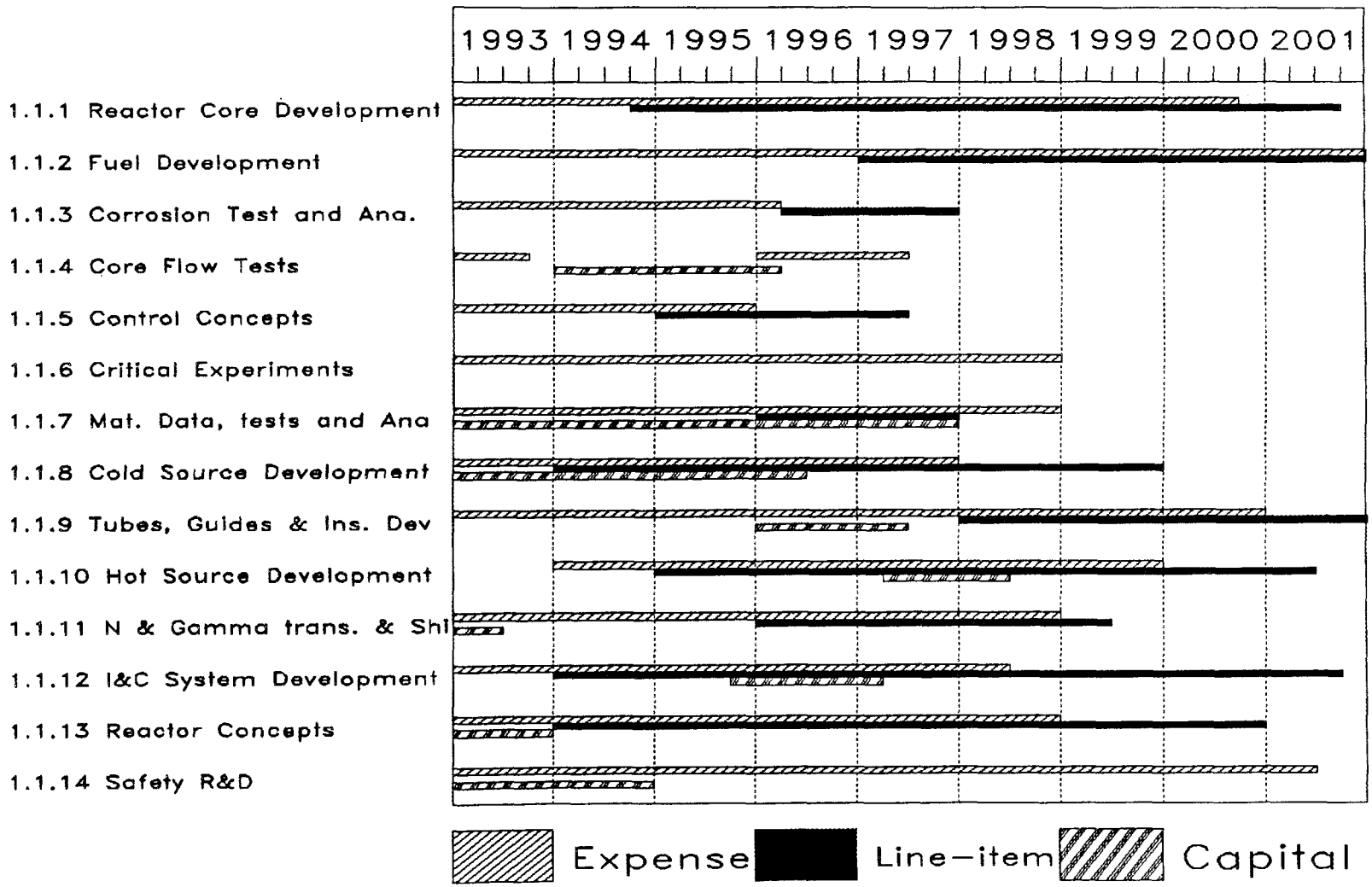

Fig. 2.2. Work breakdown structure 1.1 Research and Development program.

Table 2.1. Integrated Research and Development Program costs (FY 1993 through FY 2001) (all costs in 1992 dollars)

Work breakdown structure	Expense ($K)	Line item ($K)	Capital ($K)
1.1.1 Reactor core development	11,241	5,002	
1.1.2 Fuel development	8,194	3,114	
1.1.3 Corrosion tests and analyses	2,155	471	
1.1.4 Core flow tests	842		1,639
1.1.5 Control concepts	294	238	
1.1.6 Critical experiments	6,134		
1.1.7 Material data, structural tests, and analyses	9,183	1,231	1,844
1.1.8 Cold source development	2,367	6,590	2,743
1.1.9 Beam tube, guide, and instrument development	14,920	2,537	1,260
1.1.10 Hot source development	1,900	2,109	1,243
1.1.11 Neutron and gamma transport and shielding	2,498	751	45
1.1.12 Instrumentation and Controls System Development	2,114	1,471	599
1.1.13 Reactor facility concepts	7,484	12,518	703
1.1.14 Safety Research and Development tests	22,030		525
Totals	91,356	36,032	10,601

the R&D program is expected to be completed by the end of FY 1997, and more than 90% by the end of FY 1998. Only four activities have significant efforts beyond FY 1998:

1. The reactor core development activity extends into FY 2001 to provide the reactor physics and thermal-hydraulic analyses and documentation necessary to support the FSAR.

2. The fuel development task extends into FY 2001 to provide the fuel element production mode fabrication methodology and procedures.

3. The beam tube, guide, and instrument development task extends into FY 2001 to supply the final specifications and testing of the instruments before installation in the ANS facility

4. Much of the safety R&D task extends into FY 2000 to provide the final testing and analyses of test data needed to support the FSAR.

The costs presented in Table 2.1 represent a best estimate of the funding levels needed to support the various R&D tasks. A contingency to cover the uncertainty in estimating the cost of completing each activity was determined by examining each task and evaluating the uncertainty in the estimated costs. Individual contingency values range from as high as 150% to as low as 5%; the average is about 20%. This contingency would be used to cover cost overruns in individual tasks and to perform new tasks determined to be necessary by the staff and/or project review committees.

The schedule and costs are considered to be relatively tight; therefore, problems must be identified and resolved as early as possible. Early resolution of problems is a recurring theme throughout the tracking of the R&D tasks (discussed in Subsect. 2.5) and is considered critical to avoid having R&D tasks become part of the project's critical path. The relationship between the R&D program schedule and the total project schedule can be seen by comparing Fig. 2.2 with the project schedule as presented in Fig. 2.3

2.3 RESEARCH AND DEVELOPMENT TEST PROGRAM

The testing portion of the R&D program includes tests to establish correlations applicable to the ANS conditions, tests to validate analyses, tests to evaluate alternative design choices, and tests to demonstrate the functional performance of prototype components. Currently, about 60% of the total R&D costs are associated with the testing activities listed in Table 2.2. These tests were identified as being necessary by project staff and project review teams. Detailed explanations of these tests are included as part of the task descriptions included in Sect. 3 of this report.

The schedule for the testing activities is shown in Fig. 2.4. This schedule has been developed to make maximum use of experimental facilities whenever possible and to provide timely generation of data to support the design and other R&D activities. It is important to note that some of these tests require long lead times to plan, procure test sections, and perform tests. Therefore, delays in the activity due to funding shortfalls inevitably will lead to schedule slippage that cannot be made up by increased funding later.

All test programs and their resulting test facility requirements are reviewed by the R&D test facility manager, who has final approval authority over all test program cost estimates. In addition, as discussed in Sect. 2.5, all test programs must be approved by the project's Test Facility Review Board before any procurements, fabrications, or tests are carried out.

2.4 MAJOR PROJECT INTERFACES WITH THE RESEARCH AND DEVELOPMENT PROGRAM

Although all R&D tasks have direct or indirect ties to the facility design, 17 specific R&D milestones have been determined to be major project interfaces. Due to their potential to impact the project schedule's critical path, these tasks have been designated critical milestone events within the R&D Program and they receive special attention. These critical milestone events are presented in Table 2.3 by level three WBS along with the WBS level two ties. As indicated in this table, the present R&D schedule for these events is consistent with the completion dates required to support project needs. However, it is important to note that some tasks have little margin (as low as 3 months).

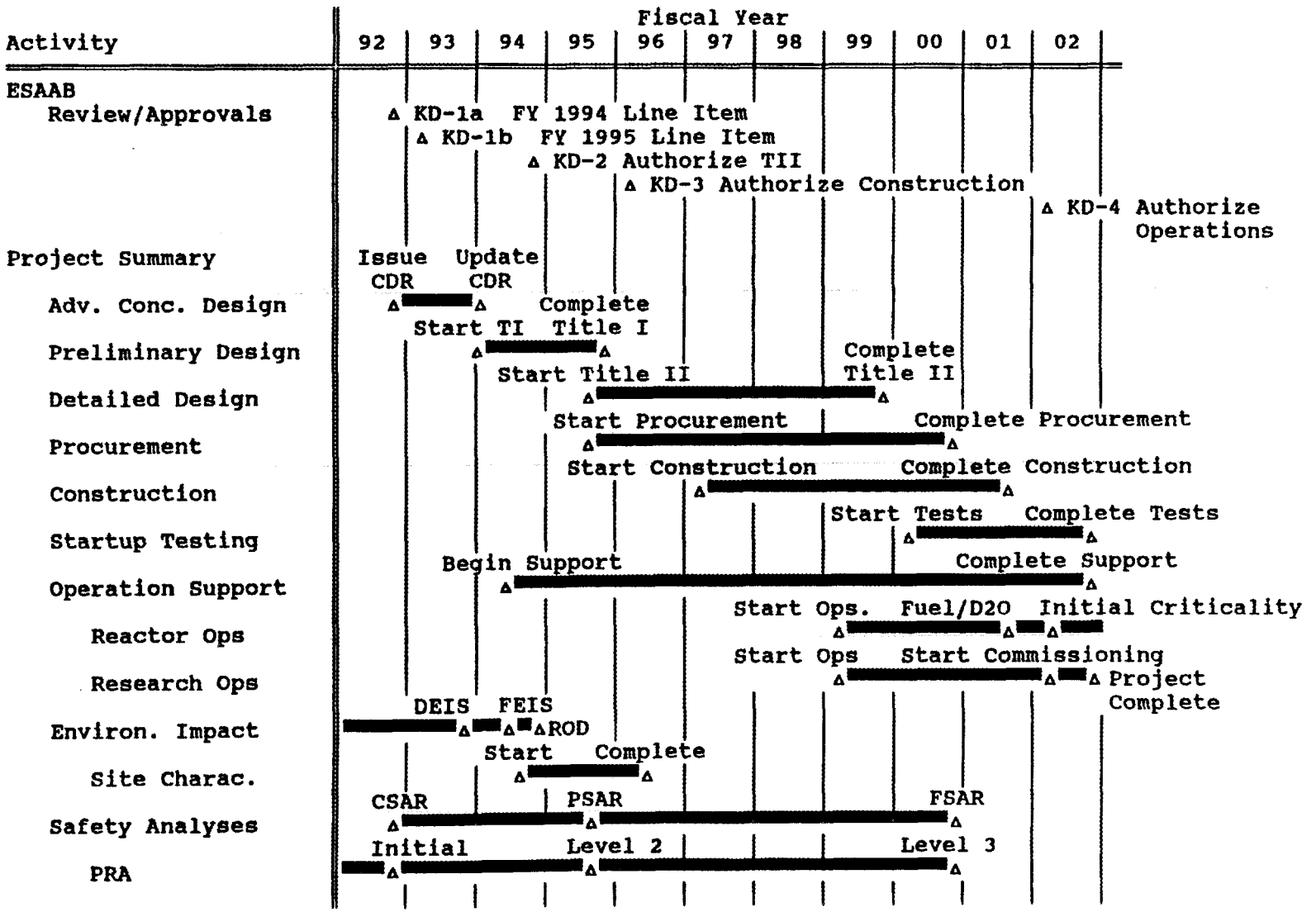

Fig. 2.3. Advanced Neutron Source planning schedule—summary.

Table 2.2. Test program activities

Work breakdown structure	Plan testing activity	Status
1.1.2	Capsule irradiation tests	In progress
1.1.2	Miniplate irradiation tests	In planning
1.1.2	Full-size plate irradiation tests	Future
1.1.2	Burnable poison fabrication tests	Future
1.1.2	Fuel fabrication tests	In progress
1.1.3	Corrosion loop tests	In progress
1.1.3	Pool boiler tests	Completed
1.1.3	Irradiation/corrosion tests	Future
1.1.4	Thermal-hydraulic tests	In progress
1.1.4	Natural circulation tests	Future
1.1.6	Critical experiments	In planning
1.1.7	Fuel plate stability tests	In progress
1.1.7	Fuel plate thermal deflection tests	Future
1.1.7	Fuel element and poison element vibration tests	Future
1.1.7	Material irradiation tests	In progress
1.1.7	Core pressure boundary tube fracture tests	Future
1.1.8	Two-phase cryogenic thermal-hydraulic tests	Completed
1.1.8	LH2/LD2 pump tests	In planning
1.1.8	LH2/LD2 loop tests	In planning
1.1.8	Cold source prototype tests	Future
1.1.9	Beam tube and guide assembly tests	Future
1.1.10	Hot source prototype tests	Future
1.1.12	Reactor protection system prototype tests	Future
1.1.13	Vessel seal tests	In planning
1.1.13	Locking ring/bolt torque tests	In planning
1.1.13	Latch tests	In planning
1.1.13	Beam tube thimble collapse tests	Future
1.1.13	Outer shutdown rod hydraulic tests	Future
1.1.13	Reflector vessel and core flow tests	Future
1.1.13	Refueling components test facility	Future
1.1.13	Irradiation capsule disconnect assembly tests	Future
1.1.13	Closure elbow refurbishing assembly tests	Future
1.1.13	Servomanipulator	Future

Table 2.2. (continued)

Work breakdown structure	Plan testing activity	Status
1.1.14	Fuel/water explosive interaction tests	In progress
1.1.14	Fission product release tests	In planning
1.1.14	Fuel transient performance tests	In planning
1.1.14	Fuel/concrete interaction tests	Future
1.1.14	Aerosol chemistry tests	Future
1.1.14	Containment release tests	Future
1.1.14	Flow blockage tests	In progress
1.1.14	Transient two-sided channel tests	In planning
1.1.14	Transient single-sided prototype span tests	Future
1.1.14	Low mass flux-positive quality tests	Future
1.1.14	Blowdown pump performance tests	Future
1.1.14	Boiling natural circulation tests	
1.1.14	Safety hydraulic testing of nonfuel components	Future
1.1.14	Pipe break experiments	Future

Therefore, it is important to give special attention to monitoring the schedule of these tasks to minimize the potential for R&D task impact on the project's critical path.

2.5 RESEARCH AND DEVELOPMENT PROGRAM PROCESS

This subsection summarizes the process by which an R&D task is identified, initiated, and completed, as illustrated in the flow chart presented in Fig. 2.5. The flow chart shows clearly the amount of planning required before beginning a task. This planning effort is critical to the cost and schedule control of the R&D Program. The remainder of this subsection provides a summary of each major step in the process.

2.5.1 Identification of Major Research and Development Activity

Fourteen major R&D activities are identified and discussed in Subsect. 2.1 of this report. These activities appear to be general enough to cover any new tasks that might be added in the future. Therefore, the set of major activities is thought to be complete, and no new major activities in the R&D program are expected.

2.5.2 Identification of Tasks Within an Activity

An initial set of tasks, discussed in Sect. 3, has been identified for each activity. These tasks are distinct pieces of work that represent the R&D program WBS level four activities. Within the R&D program, all costs and schedules are tracked at this level. Although all tasks have been defined by project staff (including project subcontractors), some tasks have been developed from review committee comments and suggestions. Project staff may add tasks as deemed necessary (particularly as a result of review committee action items or suggestions), but the addition of new tasks must be minimized if cost and schedule control is to be maintained.

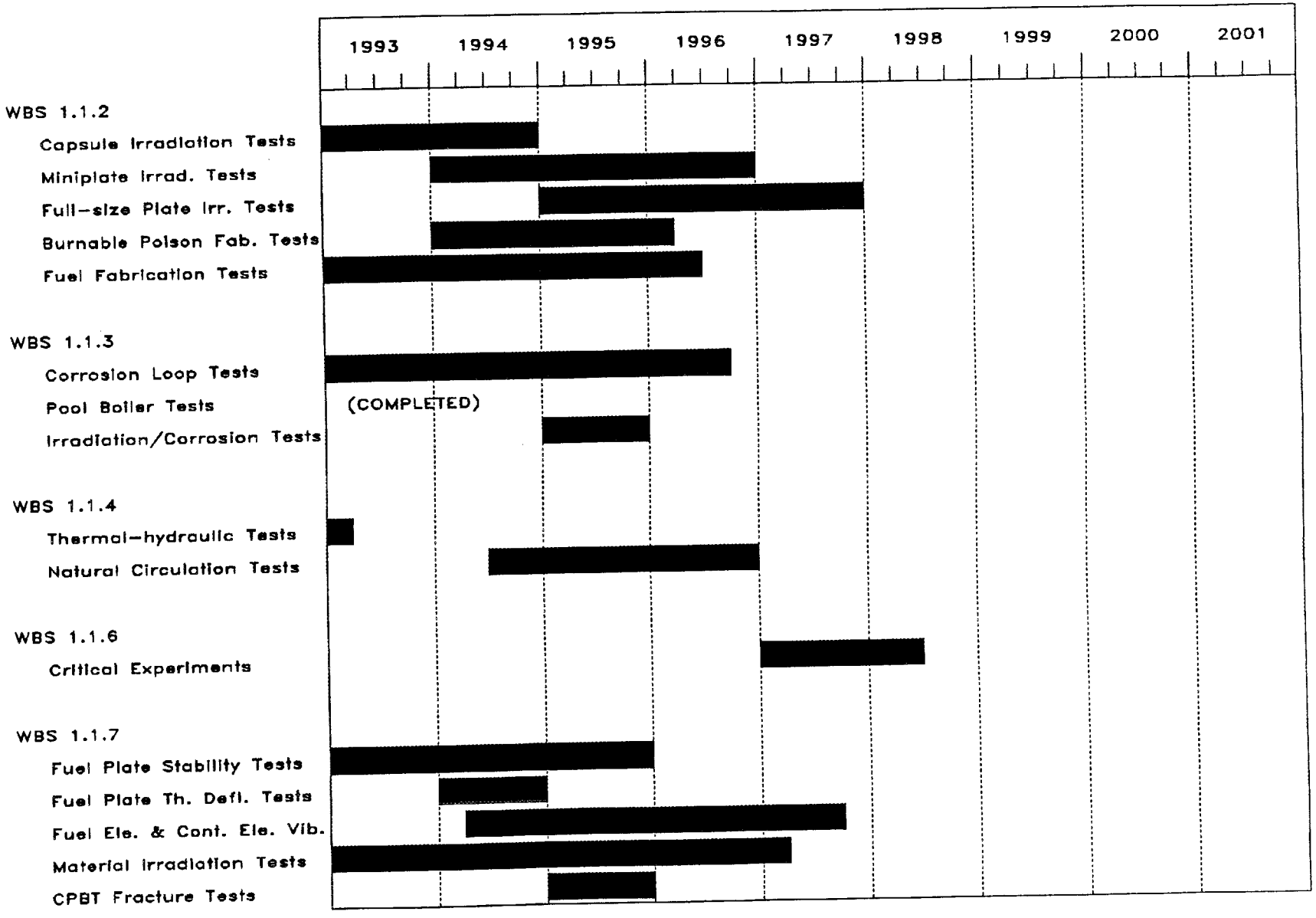

Fig. 2.4. Test program schedule.

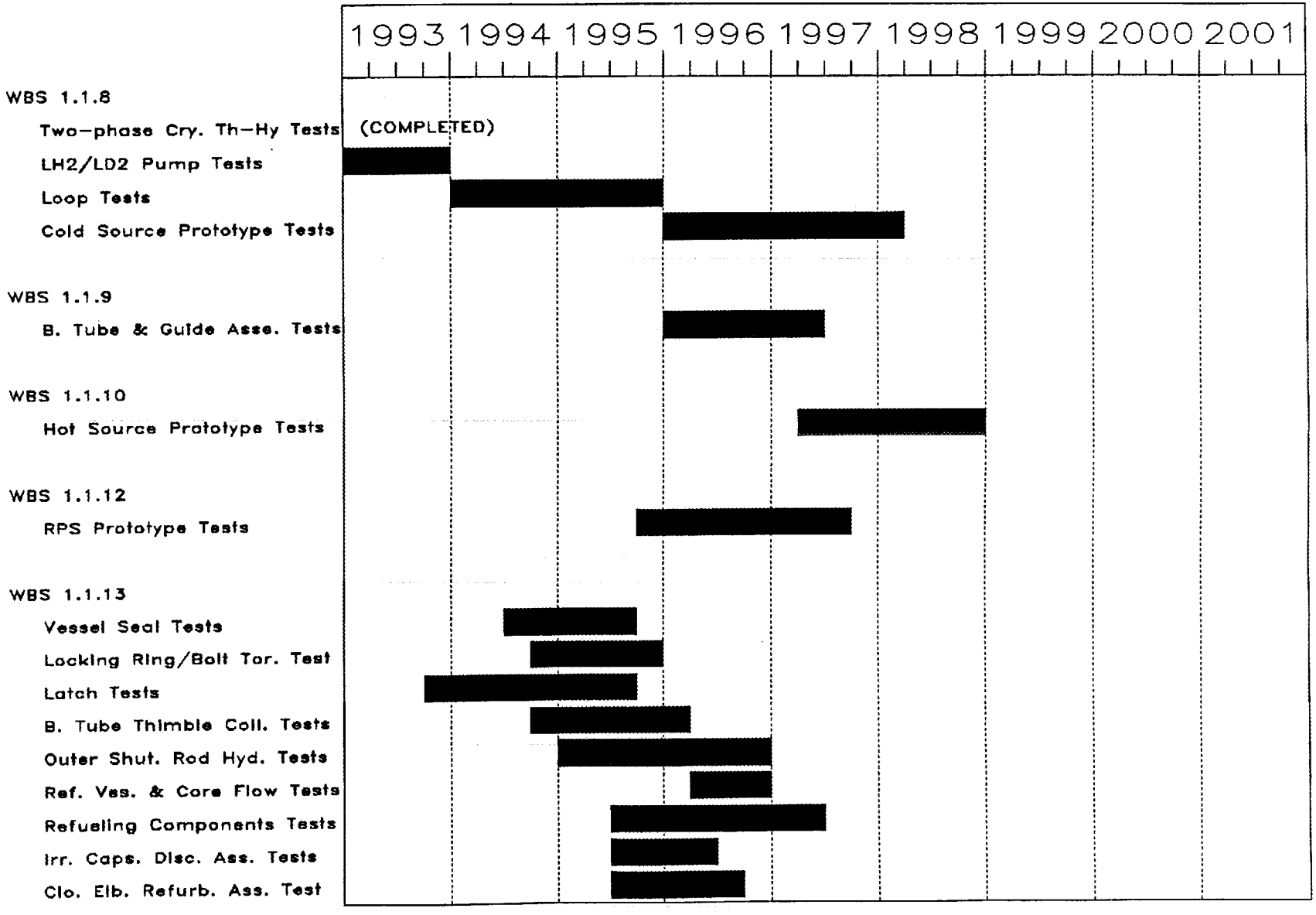

Fig. 2.4. (continued)

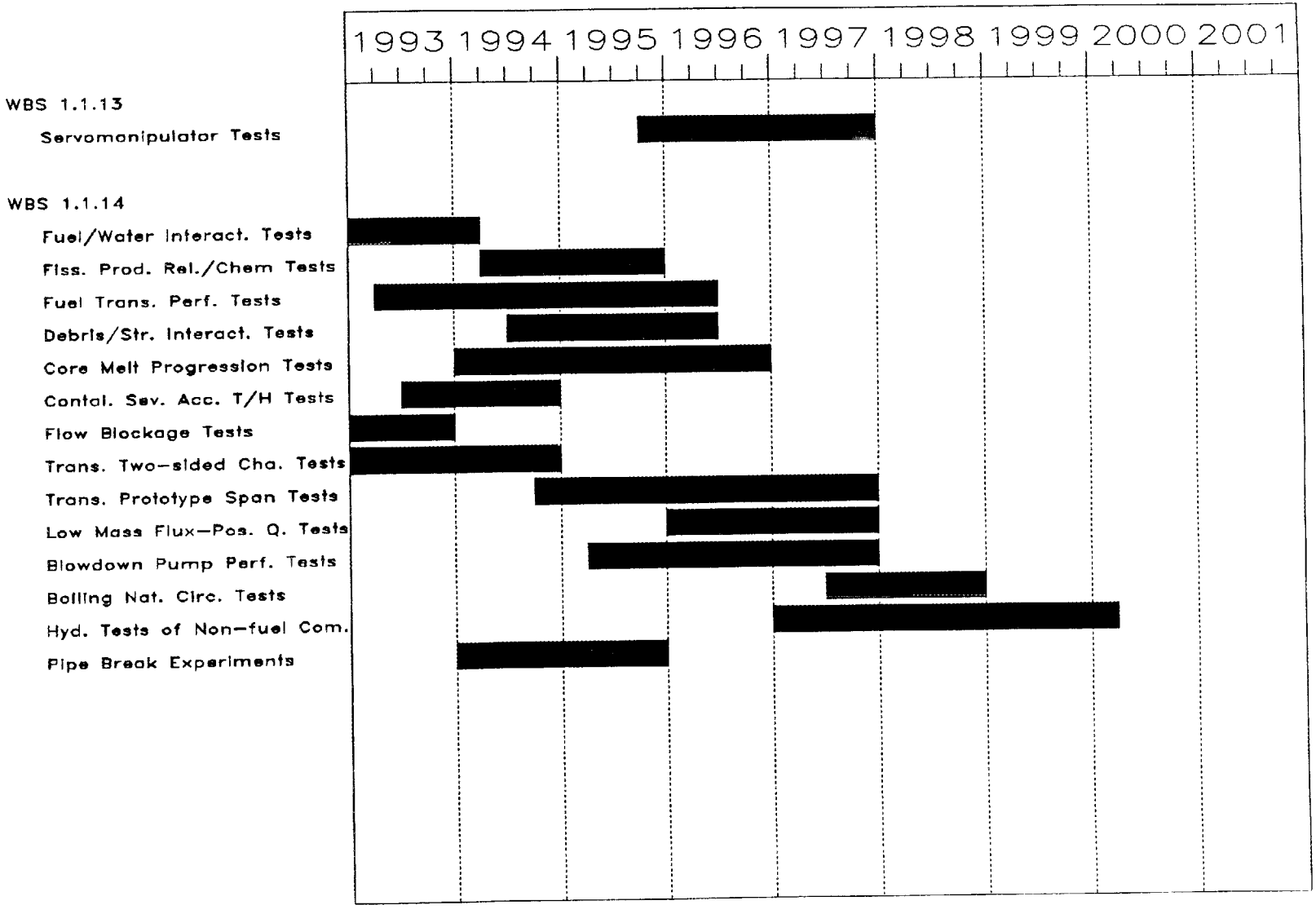

Fig. 2.4. (continued)

Table 2.3. Research and Development Program critical milestone events

	Research and Development Program critical milestone	Scheduled completion date	Project WBS element tie and required date				
			1.2	1.3	1.4	1.5/6	1.7
1.1.1	Core Development						
	Core design support analyses complete	4/96		7/96	7/96		
	Final Safety Analysis Report support analyses complete	9/99	9/00				10/00
1.1.2	Fuel Development						
	Fuel design specification complete	6/95		9/95			10/00
	Fuel performance report complete	9/98	9/00				
1.1.3	Corrosion Loop						
	Water chemistry requirements defined	3/96				9/96	
1.1.6	Critical Experiments						
	Critical experiments complete	6/98	9/00				
1.1.7	Materials Data, Tests, and Analyses						
	Core pressure boundary tube fracture tests complete	6/96		9/96			
	Materials properties data base complete	6/95		9/95	9/95		
	Component vibration tests complete	9/97		12/97			
1.1.8	Cold Source Development						
	Cold source prototype test complete	1/98			6/98		
1.1.9	Beam Tube, Guide, and Instrument Development						
	Beam tube assembly prototype test complete	3/97			9/97		
1.1.10	Hot Source Development						
	Hot source prototype test complete	9/98			6/99		
1.1.12	Instrumentation and Controls System Development						
	Reactor protection system prototype testing complete	6/97		1/98			
1.1.13	Reactor Facility Concepts						
	Component evaluation tests completed	9/95		6/96			
1.1.14	Safety Research and Development Tests						
	Transient thermal-hydraulic testing complete	9/99	9/00				
	Severe accident analyses testing complete	9/96	9/97	12/96			

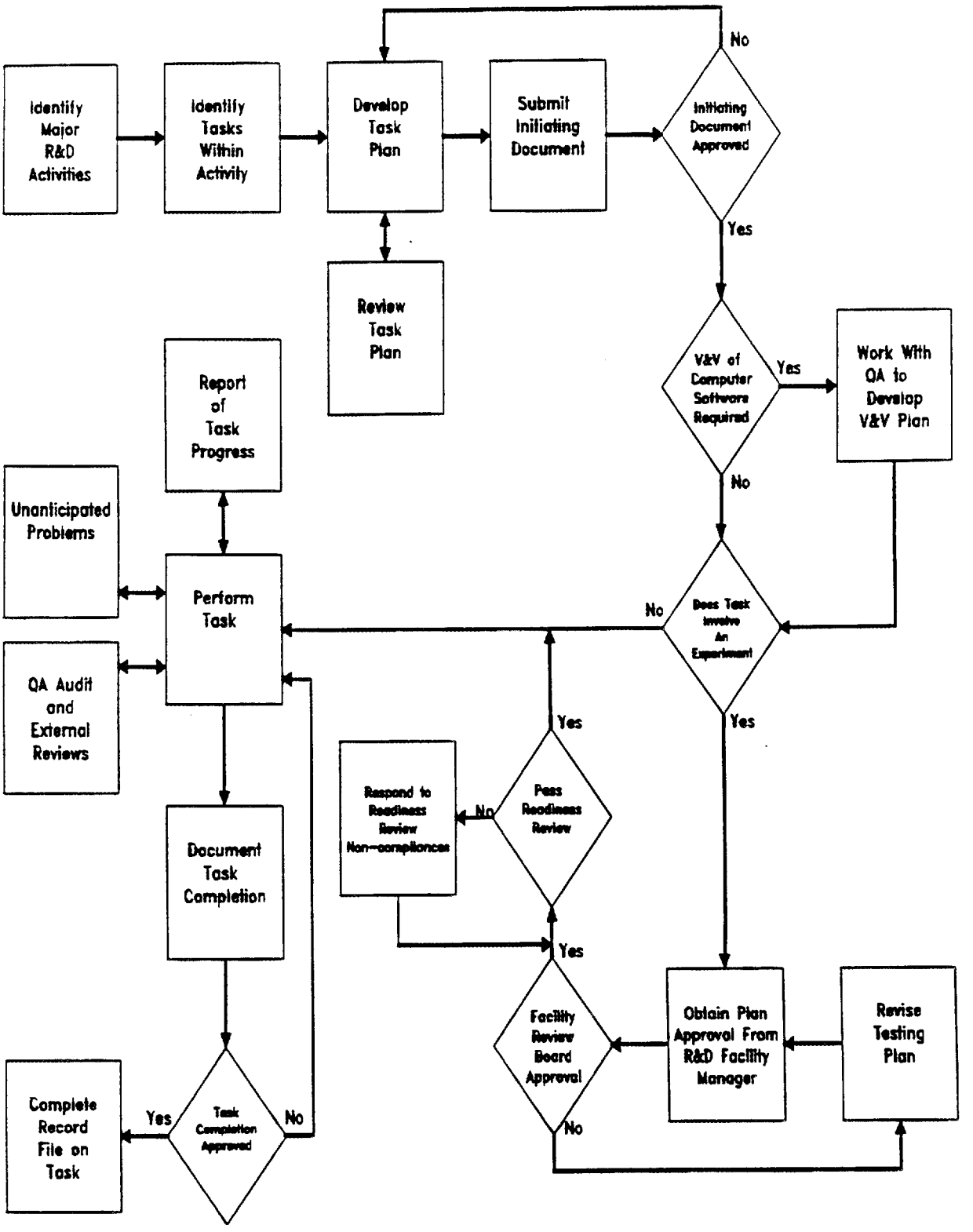

Fig. 2.5. Research and Development task flow chart.

2.5.3 Development of Task Plan

Once an R&D task has been identified, a plan for that task must be developed. The plan includes a general summary of the task, its justification, its required schedule, and its estimated cost. The present set of task plans is presented in Sect. 3 of this report.

Task plans are reviewed internally and externally at least once each year. External reviews may be performed as part of a project-wide review or as a focused review of a specific task. Changes are made to the plans to reflect scope changes, schedule changes due to funding shortfalls or overall project schedule changes, and cost changes due to scope and/or schedule changes. This R&D Plan will be reissued whenever significant changes in the task plans occur.

2.5.4 Submittal of Initiating Document

Just before the start of an R&D task (level 4 WBS item), an initiating document is required. As defined in the ANS procedures, the initiating document records specific information for a defined task, including task-related communication between the task leaders, other project technical staff, and project management. Development of this information also provides an opportunity for a mini-readiness review to determine whether the particular task should begin. The single-page initiating document notifies project management, QA, and other interested staff of the start of a new task.

A submitted initiating document may be rejected for several reasons: lack of an indication of readiness to begin the task, identification of a perceived flaw in the proposed plan, review comments indicating that the plan as proposed will not supply all the information needed to complete the task, or concerns raised by project QA. A rejected initiating document may be resubmitted when issues have been resolved.

2.5.5 Development of the Software Verification and Validation Plan

Software to be used in each task is identified as part of the task initiating document. Each software item will be reviewed to determine V&V needs based on its application within the R&D program. A graded approach will be used, based on the end use of the data obtained from the software in question. If the subject software previously has undergone V&V to an equal or higher level, it is necessary only to verify the appropriateness of the anticipated application of the software within the R&D task. A full V&V plan will be developed for all software that is considered to have a direct impact on the safety analysis of the ANS facility. Each V&V plan will be developed by the appropriate task leader with support and guidance provided by the ANS QA Program. Final approval of each V&V plan must be obtained from the R&D manager and the QA Program. Once created, the V&V plan becomes part of the appropriate WBS level four subtask plan and must be finalized before the task is considered complete.

2.5.6 Review of Test Facilities and Testing

As indicated in Sect. 2.3, a number of test facilities and testing activities are planned within the R&D program. A test plan is developed prior to commitment to any testing activity and is then reviewed by the R&D facility manager. Once the plan is approved by

the R&D facility manager, it is presented to a facility review board composed of a peer group of project and nonproject personnel. The testing activity proceeds if the plan is approved by this board, but a readiness review is conducted before any actual testing can begin.

The reviews performed by the R&D facility manager and the facility review board are intended to be close examinations of the need to perform the tests; the proposed way of performing the tests (e.g., building a new facility or using an existing facility, performing tests inside or outside ORNL); and the estimated costs of performing the tests. The purpose of these two reviews is to ensure that only required testing is performed and that tests are performed efficiently and cost effectively.

The readiness review is considered a QA activity to ensure that all aspects of the testing process have been considered and that no problems preclude beginning the testing process. This process ensures that all environmental, safety, and health (ES&H) considerations have been made, that all procedures are in place, and that all approvals have been obtained. The testing process cannot begin until the readiness review has been passed.

2.5.7 Performance of Task

An active task is tracked in three ways: monthly progress reports, QA audits or surveillances, and external reviews. Monthly reports of technical progress and costs are obtained for each WBS level four R&D activity. These reports are tracked against the expected schedule and costs for each subtask. Significant deviations in either schedule or costs are identified and are entered into the "unanticipated" problem category. These problems, along with other unanticipated problems that arise (e.g., surprise data, new compliance rules, and equipment failures), are resolved by informal discussions between project staff and project management to minimize impacts on project schedule and costs.

QA audits or surveillances are performed periodically by the ANS project QA department and by the QA staff of the organization performing the work. These QA activities check for compliance with project procedures and task plans. Severe noncompliance could lead to a halt of all work on the task until compliance has been ensured.

External reviews of the quality, planning, and progress of technical tasks also are performed for the significant R&D tasks or task areas. These external reviews provide project management with important information on the status of a particular task and provide the opportunity to identify potential problems before a task is completed. In addition, these reviews provide project staff an opportunity to debate alternative approaches with peer groups.

2.5.8 Documentation of Task Completion

A task completion document must be completed at the end of each WBS level four task. This completion document, explained in the project procedures, documents the completion of a task, summarizes the results of the task, and provides the references to reports or other documentation associated with the task. The task is not considered complete until the R&D manager signs the completion document form. Upon completion of a task, all records and files are stored in compliance with project procedures.

2.6 RESEARCH AND DEVELOPMENT QUALITY ASSURANCE ACTIVITIES

Four subjects are discussed in this section. Subsection 2.6.1 addresses the applicability of QA activities to the R&D program. Subsection 2.6.2 discusses the management of the QA process within the R&D program. Subsection 2.6.3 discusses the performance of QA activities. Finally, Subsect. 2.6.4 presents the methods for assessment of R&D activities.

2.6.1 Applicability of Quality Assurance to the Research and Development Program

R&D for the ANS Project is distinguished from the type of work involved in design or construction activities by the differences in the direct product. The direct product of a planned R&D chain for the ANS Project is the development of equipment, devices, design input information, or technology for application to the ANS.

The ANS Project is committed to using the guidance set forth by DOE Order 5700.6C. As part of the implementation of this guidance, the project uses the national consensus standard ASME NQA-1, *Quality Assurance Program Requirements for Nuclear Facilities*, to describe and define the management controls to be used in the design and construction of the ANS facility.

The project has developed and established a series of management controls that are documented in the ANS Quality Assurance Plan.

2.6.2 Management of Quality Assurance Within the Research and Development Program

The implementation of the management control system elements of NQA-1 starts with R&D task planning and QA program application reviews. The results of these activities are technical program objectives and QA program elements for the R&D task. Task planning constitutes a portion of the project's quality improvement strategy.

Training and qualification for R&D personnel participating in the project follow the guidance set forth in ASME NQA-1, Supplement 2S-4. The principal focus of these training/indoctrination sessions is areas such as knowledge of working procedures and a working knowledge of NQA-1, project terminology, and organization relationships. Technical expertise is established by the laboratory through the laboratory recruitment and personnel retention programs. The ANS R&D manager matches the project's needs with the laboratory's skill mix to arrive at the appropriate research personnel for a particular task.

Nonconformances and reportable occurrences are documented and handled by the originating ORNL R&D organizations, using their respective plans and procedures for these activities. The project also concurs with dispositions affecting project experiments or data interpretations.

For project R&D activities, the task leaders define those activities to be documented in instructions, procedures, and drawings early in the task. The task leader is responsible for the review of drawings, procedures, and instructions to ensure that appropriate qualitative or quantitative test criteria (or acceptance criteria) have been incorporated. Procedures or instructions prepared by the research divisions are reviewed by the division QA specialist or coordinator. Division quality plans specify the documents subject to document control measures, including preparation, review, approval, and issue. The researchers and the ANS task leader together identify the records to be generated and maintained. Dual storage methods are used for those records considered to be quality records.

2.6.3 Performance of Research and Development Quality Assurance

The task leader conducts training of R&D personnel in test facility operation, data collection, etc., in the form of orientation/indoctrination sessions. In certain cases, as defined by the task leader, walk-downs of the test facility are conducted.

R&D task and subtask readiness reviews are performed by project personnel and ORNL research personnel to ensure that all test prerequisites are met and that procedures, safety initiatives, and documentation needs are prepared and resolved.

Design controls are applied to the construction of new test facilities or modifications to existing facilities. They include such areas as materials selection, facility structural integrity, and test article design. Experiments for the project have current drawings showing the configuration of the test facility, including instrumentation and data gathering locations and types. Processes used in project R&D activities are evaluated for impact on test activities and the reliability of the data produced. If processes are identified as requiring control, procedures will be established for process implementation, including any necessary personnel qualifications.

Sample management practices used in project R&D activities include sample identification and control, traceability, and archival considerations. Special needs are documented in task initiation documents, task review meeting minutes, workshop meeting minutes, or project correspondence.

R&D standard operating procedures identify the measuring and test equipment requiring control and calibration. The task leader, in conjunction with ANS personnel, makes this determination.

Procurement of R&D items is subject to the controls of their respective division and the provisions of the ORNL QA manual. These controls specify that items and materials will satisfy all procurement requirements before release to the researchers for use.

R&D experiments and tests involve hardware and test facilities that are subject to inspection. These inspections are limited to those required to ensure that the test apparatus and test specimen are in accordance with their design for the performance of the test.

2.6.4 Research and Development Assessments

The R&D program's achievement of quality is measured through two types of assessment activity: (1) self-assessments and (2) independent assessments.

Self-assessments occur both at the project director level and at the line management level. Self-assessments at the project director level measure the overall R&D program's success in achieving the R&D goals for the project. Input into this level assessment activity comes from peer reviews, project workshops, and other program-level activities. Self-assessments at the line manager level measure the success of a particular task, or group of tasks, in meeting the specific objectives of the task. Input into this level of assessment comes from internal surveillances, task leader meetings, and other task-specific activities.

Independent assessments of R&D organizations and activities are conducted by organizations independent of the work activities, including customer oversight. Independent assessments are, in part, externally organized reviews/workshops and ORNL, MMES, and DOE audits and surveillances. The results of these assessments establish a baseline of acceptable performance. The corrective measures and actions provide input into the organization's self-assessment program.

R&D organizations are subjected to independent assessments consisting of peer reviews, ORNL Quality Department assessments, award fee assessments (conducted by Central Management Offices, ORNL, MMES, or corporate), ORNL ES&H assessments, and MMES technical audits. The results of these assessments establish a baseline of acceptable performance. The corrective measures and actions provide input into the organization's self assessment program.

3. DESCRIPTIONS OF THE RESEARCH AND DEVELOPMENT ACTIVITIES AND THEIR ASSOCIATED TASKS

Subsections 3.1 through 3.14 provide detailed information on the various R&D activities, including cost, schedule, justification, and activity description. Each subsection provides a summary of a WBS level three activity, which is then followed by more detailed information at WBS level four.

3.1 REACTOR CORE DEVELOPMENT—WBS 1.1.1

The technical objectives of the ANS Project, as listed in the Plant Design Requirements document[1], require that different regions of the reactor be designed with neutron fluxes of various magnitude and spectrum. No existing reactor design meets these requirements, and a new reactor core concept was required. This R&D activity provides for the development of a basic ANS reactor concept that meets or exceeds the technical objectives, along with the identification of appropriate analysis methods necessary to evaluate the concept. In addition, it provides the nuclear and thermal-hydraulic analyses necessary to support the design tasks. This WBS element contains two major project milestones:

1. Complete core design support analyses by April 1996. This will ensure that the Title II design activity can proceed with limited risk of design changes at a later date.

2. Complete FSAR support analyses by September 1999. Under the present schedule, the FSAR is to be completed by September 2000. To meet this schedule, the analyses supporting the documentation must be completed 1 year earlier.

The reactor core development activity is divided into five WBS level four tasks that are summarized in Table 3.1. As indicated in that table, this activity includes the tasks associated with the definition of the reactor core concept, the development of the process for treatment of uncertainties, and the performance of reactor physics and thermal-hydraulic analyses necessary to support the reactor systems design effort (WBS 1.3). Most of this work will be performed at ORNL and the Idaho National Engineering Laboratory (INEL), with support from various universities and industry groups. The total estimated costs of the activity over the 9-year period covered by this R&D plan are given in Fig. 3.1, and the associated schedules are shown in Fig. 3.2. Note that the line-item costs are used for analyses and documentations that directly support preliminary (Title I) and detailed (Title II) design activities, while the expense studies work is associated with perturbation studies, examination of design alternatives, and validation efforts. The line item costs also include the various tasks needed to support the nuclear and thermal-hydraulic sections of the FSAR and the effort needed to package and document the analytical tools in a form required by the operational staff for a transition to the operation of the facility. Subsections 3.1.1 through 3.1.5 provide more detailed information on the WBS level four tasks under this activity.

Table 3.1. Summary description of reactor core development work breakdown structure level four tasks

WBS	Task description
1.1.1.1	Methods development—This task is to identify/develop methods, codes, and models necessary to perform neutronics and thermal-hydraulics analyses of the ANS reactor core. Specific activities needed to support the neutronics work include cross section development, transport and diffusion methods comparisons, and analyses deemed necessary to benchmark methods against data from existing high flux reactors. Thermal-hydraulic work includes the selection and development of thermal-hydraulic correlations, the development of an ANS steady state thermal-hydraulic model for normal and operational transient conditions, development of a statistical methodology for combination of uncertainties, development of a coolant flow distribution model, and development of core component thermal-hydraulic models.
1.1.1.2	Preconceptual core development—This task was to provide the data necessary to define a reference core to start the conceptual core design. It included evaluations of various core geometries, H_2O vs D_2O cooling, power level, and power peaking. This task is considered to be complete and is documented in the ANS Preconceptual Core Design Report[2].
1.1.1.3	General design criteria development support—This task is to identify and document the requirements for margins and uncertainties to be used in the nuclear and thermal-hydraulic evaluations of the reactor core.
1.1.1.4	Reactor physics support to design—This task will provide the neutronics support to the development of the reactor core. Subtasks include the development of the fuel grading to provide desired power shapes, evaluations of reactivity worths, burnup analyses, perturbation analyses (e.g., voiding, light water contamination), refueling analyses, and startup analysis.
1.1.1.5	Thermal-hydraulics Support to Design—This task will include thermal-hydraulic analysis of various reactor core geometries and conditions and reactor components, as well as the detailed evaluations of coolant flow distribution. Analyses will include the use of the steady state thermal-hydraulics code, the core flow split codes, and other thermal analysis models developed under WBS 1.1.1.1. This task provides the thermal-hydraulic analysis support for the design of the reactor region components including the fuel elements, the core pressure boundary tube, control rods, and various experiment systems. It also provides the thermal-hydraulic support to the design of other R&D facilities, analytical examinations of natural circulation phenomena, and preoperational thermal-hydraulic support to operational planning and training.

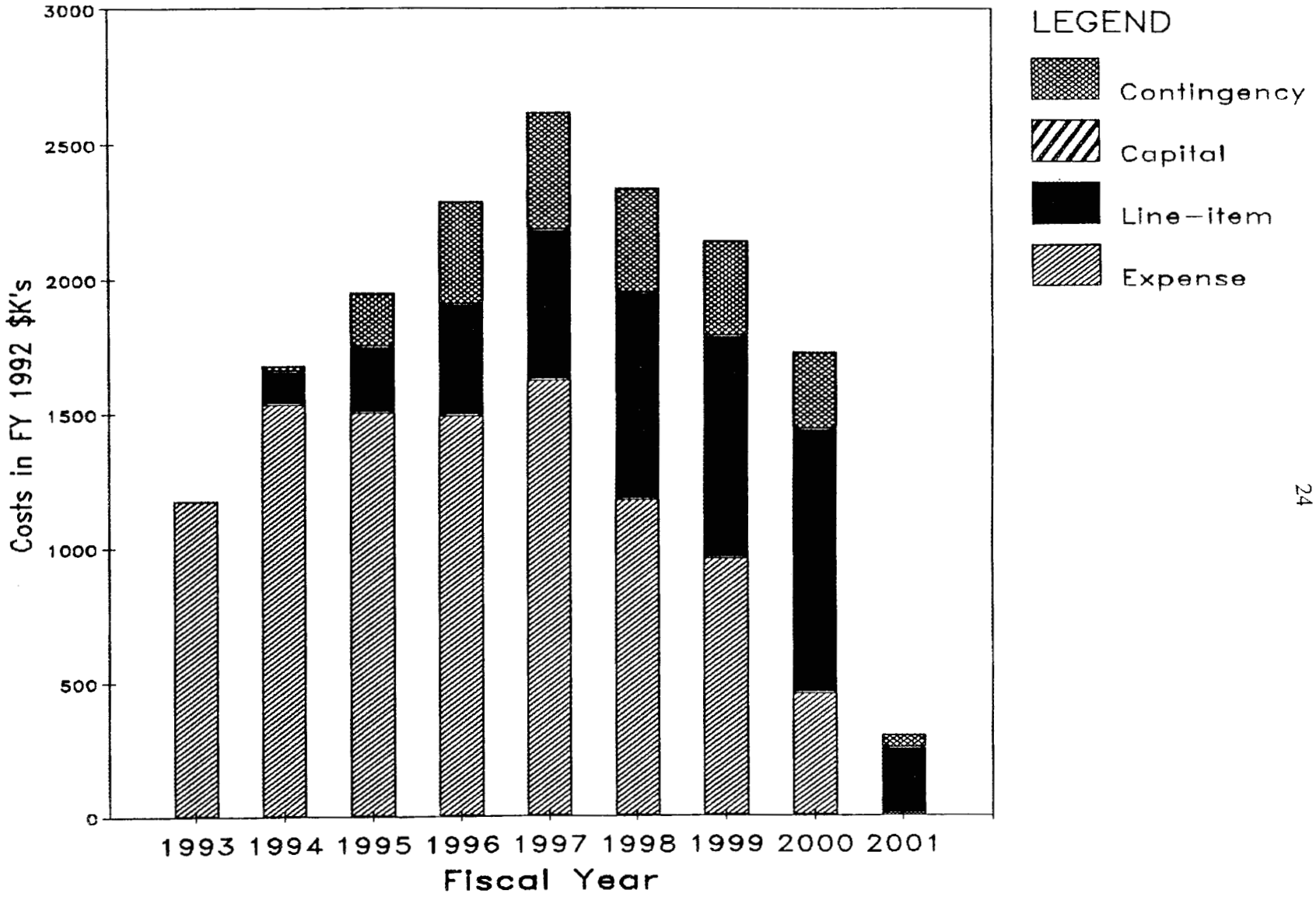

Fig. 3.1. Summary of cost estimates for work breakdown structure 1.1.1, reactor core development.

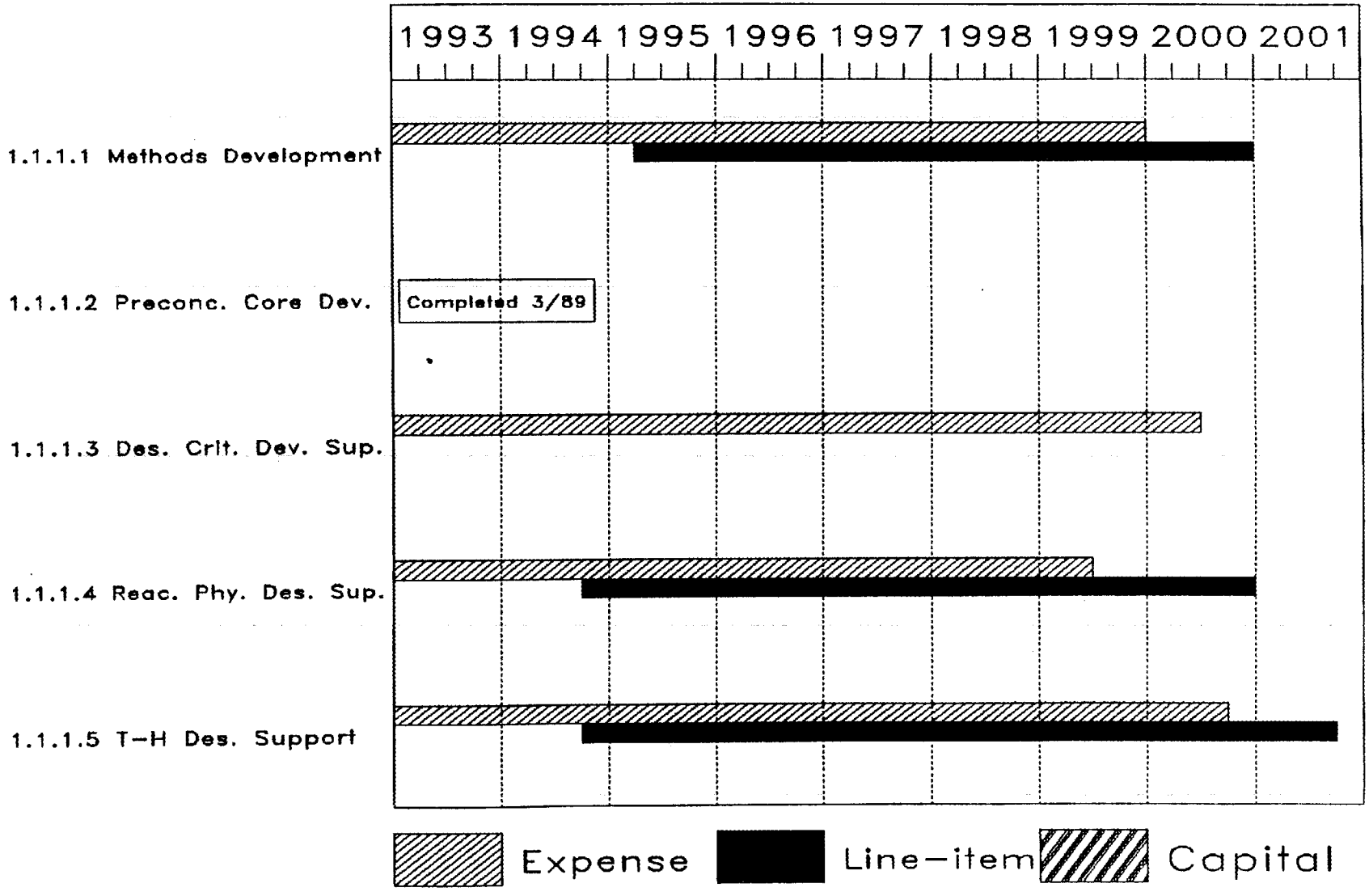

Fig. 3.2. Schedule for work breakdown structure 1.1.1, reactor core development.

3.1.1 Methods Development

3.1.1.1 Justification for the Methods Development Task

Although the reactor core nuclear performance will be verified later in the R&D program by critical experiments (WBS 1.1.6), these experiments occur late in the project schedule and are used for confirmation rather than basic core design development. In addition, the full thermal-hydraulic performance of the reactor core cannot be fully verified until the actual ANS startup tests. Therefore, the development of the ANS reactor core will depend heavily on analytical methods and models. This makes the selection/development and validation of both neutronic and thermal-hydraulic analytical methods a critical task that must be performed.

3.1.1.2 Description of the Methods Development Task

The work can be divided into three major areas:

1. Identification of basic methods—For reactor physics analyses, this subtask includes selection of cross sections (energy treatment, spatial dependence, and burn dependence); evaluation of transport vs diffusion methods; and identification of the approach for treating fuel burnup and fission product buildup. For the thermal-hydraulic analyses, this subtask covers issues related to correlation selection, identification of the physical property data to be used in analyses, determination of the method for treatment of uncertainties, identification of and treatment of fuel defects, and examination of one-dimensional (1-D) vs two-dimensional (2-D) vs three-dimensional (3-D) heat transfer treatments.

2. Development of appropriate ANS models—If the systems being evaluated were not modeled correctly, the results would be invalid however good the theoretical basis. Therefore, this subtask was formed from other subtasks to provide adequate emphasis on the need to develop appropriate models. The work includes gathering from drawings, sketches, or word of mouth the information necessary to complete both the neutronic and the thermal-hydraulic models to the necessary level of detail. Modeling assumptions, such as treatment of a curved surface with a rectangular mesh, are identified and evaluated as part of this subtask. Other areas, such as the selection of appropriate sizes for mesh or nodes, also are supported.

3. Validation of methods used—This subtask was formed to examine the validity of the neutronics and the thermal-hydraulic and theoretical methods used to evaluate the ANS reactor core performance. The validation of the neutronics work is performed in three phases: (1) modeling of clean criticals (simple critical experiments of uranium in heavy water) and comparison with existing measured data, (2) modeling of existing reactors or critical experiments that are similar to ANS [e.g., Institut Laue Langevin (ILL), FOEHN criticals] and comparison of the results with measured data reported in the literature; and (3) a final validation obtained by performing and analyzing our own critical experiments

prototypic of the ANS reactor core (discussed later in this report under WBS 1.1.6). Each of these phases provides an increased level of confidence consistent with the level desired for the corresponding stage of the project.

The thermal-hydraulic validation effort focuses on confirming the soundness of the various correlations by comparisons with measured data. The data needed for validation have been identified, and experiments have been set up to obtain the required data that are not covered by the limited existing data bases. Support for external reviews of both the neutronics and the thermal-hydraulics methods is also included in this subtask. The last action under this subtask will be to develop a plan of tests to be performed during the ANS reactor startup test program that will provide a final fine tuning of the analytical methods.

3.1.2 Preconceptual Core Development

Although preconceptual core development was completed in March 1989, material is included in this document for the purpose of completeness.

3.1.2.1 Justification of Preconceptual Core Development

This task initiated the development of the new reactor core concept and led to the general reactor core concept that has been developed further in the conceptual design phase. Without this task, the reactor core would look much more like existing facilities but would not achieve the performance requirements of the ANS Project.

3.1.2.2 Description of the Preconceptual Core Development Task

The preconceptual core development task consisted of numerous (hundreds) of parametric and scoping studies performed to understand the interactions and importance of the various major design parameters (e.g., core height and height-to-diameter ratio, core volume, power level, peak thermal flux, core life, coolant velocity, reflector size, moderator type, coolant type, and fuel density). Some of the detailed results of these studies are presented in ref. 2.

3.1.3 General Design Criteria Development Support

Early in the ANS Project, a decision was made to use statistical methods to account for uncertainties. The general design criteria development support task has provided efforts to develop statistical treatment of design variances (e.g., statistical variances in fuel plate thickness) and to perform the analyses to determine margins in operating conditions necessary to compensate for the variances.

3.1.3.1 Justification for General Design Criteria Development Support

The statistical data generated under this task are important to the support of the development of general design criteria. Without this task, the operational limits on the core needed to support the safety criteria for the five design bases could not be defined. These five criteria, discussed in detail in ref. 3, are expressed in terms of probability levels; thus, core performance must be converted to a probability function to test for compliance.

3.1.3.2 Description of the General Design Criteria Development Support Task.

Analytical, process, and manufacturing uncertainties must be accounted for in evaluating the performance of the ANS core. This task provides for the identification of the key parameters that must be evaluated statistically, the development of statistical distributions for the key parameters, the conversion of analytical tools to perform statistical rather than single value analysis, and the actual statistical analyses necessary to determine margins necessary to meet safety design criteria. Much of the work will focus on identifying and quantifying the manufacturing uncertainties because previous experience suggests that these uncertainties will dominate. Information obtained under this task also will be used to examine the benefit of tightening manufacturing tolerances and the cost effectiveness of additional experiments to reduce other parameter uncertainties.

3.1.4 Reactor Physics Support to Design

This task provides the reactor physics support to the development of the reactor core.

3.1.4.1 Justification for the Reactor Physics Support to Design Task

Without reactor physics support for design there would be little understanding of the nuclear performance characteristics of the ANS reactor. Thus the safety case could not be made and the project's ability to meet performance objectives would not be known.

3.1.4.2 Description of the Reactor Physics Support to Design Task

The work under this task encompasses all reactor core neutronics analysis necessary for evaluation of design options and reactor core performance over the fuel cycle. Not included in this particular task are evaluations of control concepts, because they were felt to be important enough to be treated as a separate R&D activity. Although a wide range of studies fall within the scope of this task, most of the analyses fall within five subtasks.

1. Development of the fuel grading—This subtask will provide the information necessary to determine the combination of possible fuel gradings and burnable poison distributions that produce power density distributions over the fuel cycle that most closely match the optimal distribution identified from thermal-hydraulic considerations. The effects of fuel burnup and movement of the central control rods over cycle life are included explicitly in the analysis.
2. Calculation of reactivity worths—Many experimental facilities are planned for the ANS reactor. The effect upon overall core reactivity of the various components making up these facilities will be calculated under this subtask. This calculation includes the reactivity effects of the transuranium production facilities, isotope production rods, fast flux irradiation targets, and reflector irradiation targets.

3. Burnup analysis—The analysis under this subtask includes those calculations associated with burnup that are not covered by subtask 1. Issues such as tritium production, silicon production, and production of other isotopes of concern to

the various technical disciplines are addressed by this subtask. Source terms and decay heat distributions also would be evaluated under this subtask.

4. Perturbation analysis—System perturbation effects are evaluated in terms of their impact on reactivity and/or fluxes. Examples of perturbations to be examined under this subtask are element movement, fuel expansion, fuel plate losses, heavy water voiding, and light water contamination.

5. Refueling and startup analysis—This task evaluates the refueling and startup options in terms of criticality. Calculations will be performed to track the fuel elements through all phases of the refueling process. In addition, monitoring schemes and startup procedures will be examined analytically under this subtask.

Other activities covered by this task that are not directly associated with the analyses discussed include documentation support to the Preliminary Safety Analysis Report (PSAR) and FSAR, external reviews of analyses, and other documentation activities.

3.1.5 Thermal-Hydraulic Support to Design

3.1.5.1 Justification for the Thermal-Hydraulic Support to Design Task

Various design parameters such as fuel loading, system pressure, coolant velocity, coolant temperature, and component geometry will have significant impact on the allowable operating power and safety margins of the reactor. To optimize these design parameters before final design and construction of the reactor, thermal-hydraulic calculations must be performed to investigate the importance of these design elements and determine ultimately which parameters may allow improvements in the reactor design.

3.1.5.2 Description of the Thermal-Hydraulic Support Task

This task includes the thermal analysis of various fuel loadings, coolant flow distributions within the core region, and cooling of other reactor components. It encompasses an array of reactor operating states including normal operation, pony motor flow, natural circulation, and the analysis of the refueling process and fuel storage.

The analysis of the core region will use the methodologies developed in WBS 1.1.1.1 to perform parametric calculations investigating the impact of each design parameter [i.e., coolant pressure, temperature, flow rate, cooling state (i.e., forced or natural flow), etc.] on core thermal performance. Calculations also will be performed to assess various fuel loading designs and to influence the fuel grading via. an interface with the neutronics analysis task (WBS 1.1.1.4). The thermal-hydraulic calculations will use appropriate limiting phenomena (e.g., incipient boiling, critical heat flux, and oxide spallation) as the basis for accessing core performance and will include appropriate uncertainty treatment.

Thermal-hydraulic analysis of nonfuel reactor components [e.g., the core pressure boundary tube (CPBT), control rods, reflector tank wall, and beam tubes] also will be performed under this task. Parametric and design calculations will determine appropriate operating conditions for these components considering limiting conditions such as incipient boiling and maximum component temperature.

Flow splits (between the two cores, the CPBT annulus, the central hole region, and the control rod coolant channels) will be determined considering the specific geometries,

the frictional pressure drops and form losses associated with those geometries, and the component cooling and structural requirements. Initial calculations will assume isothermal conditions and eventually will be extended to the nonisothermal case. These calculations will be used to size and locate orificing within the core region and to determine the total flow and pressure drop necessary for safe reactor operation.

This task also provides as-needed thermal-hydraulic support to other R&D tasks and preoperational support to the reactor operations staff.

3.2 FUEL DEVELOPMENT—WBS 1.1.2

The use of silicide fuel, grading in both axial and radial directions, the burnup rates encountered in the ANS, and the placement of burnable poisons in the fuel endcaps represent changes to the existing High Flux Isotope Reactor (HFIR) fuel fabrication process. In addition, improvements in the quality control process used during fuel fabrication can be translated into additional safety margin. The fuel development program develops fabrication requirements, demonstrates the fabrication process, qualifies candidate ANS materials to ANS conditions, and provides dummy and prototype plates and elements for testing under other R&D task areas. This WBS element contains two major project milestones:

1. Complete the fuel element design specification by June 1995. This allows the Title II design of the reactor system to proceed on schedule.

2. Complete the fuel performance report by September 1998. This allows ~2 years of review before the publication of the FSAR.

The fuel development activity is divided into 11 WBS level four tasks summarized in Table 3.2. Most of this work will be performed by Argonne National Laboratory, Babcock and Wilcox, and ORNL. The total estimated costs for this activity over the 9-year period covered by this R&D plan are given in Fig. 3.3, and the associated schedules are shown in Fig. 3.4. Note that the expense costs are associated with initial concept developments, including fabrication and performance restrictions, while the line-item costs are used for analyses and documentations that directly support preliminary (Title I) and detailed (Title II) design activities. In addition, the line item costs include the various tasks needed to define the production mode fabrication process; support for the fuel development sections of the FSAR; and the effort needed to package and document the fuel performance, including analytical tools, into a form required by the operational staff as a transition to the operation of the facility. Subsections 3.2.1 through 3.2.11 provide more detailed information on the WBS level four tasks under this activity.

3.2.1 Selection and Verification of Fuel and Cladding

3.2.1.1 Justification for the Selection and Verification of Fuel and Cladding Task

At the start of the ANS Project, there were several fuel and cladding options. However, it was determined that the performance for none of the fuels had been demonstrated for the expected ANS operating conditions. Thus an R&D program was determined to be necessary to qualify the fuel for the range of operating conditions expected in the ANS. Therefore, it was important that prime candidate materials be

Table 3.2. Summary description of the fuel development work breakdown structure level four tasks

WBS	Task description
1.1.2.1	Selection and verification of fuel and cladding—The purpose of this task is to review the existing data and select the fuel type and clad type most amenable to the objectives of the ANS[a] Project. The product of this task would be the identification of the primary and backup fuel and clad materials. This task is presently assumed to be complete. Primary fuel and clad materials and backup fuel materials have been identified. At present no backup cladding material is considered. If a backup cladding material or a change in the primary fuel or cladding material is determined to be necessary, this task will be reopened.
1.1.2.2	Capsule irradiation tests—Capsules containing small quantities of individual fuel particles will be irradiated in the target region of the HFIR[b]. The very small amount of fuel and the high thermal flux will allow burnup rates approaching those expected in the ANS. Temperatures typical of the ANS will be achieved with gamma heating and gas gaps. Evaluation of the fuel particles for structural stability and morphology of the fission gas bubbles will be done by metallography and scanning electron microscopy.
1.1.2.3	Miniplate irradiation tests—Miniplates will be tested in the reflector of the HFIR. This will allow examination of plate swelling, fuel-matrix, burnable poison, and meat-cladding interactions under conditions similar to those expected in the ANS.
1.1.2.4	Irradiation damage simulation studies and fuel performance modeling—Although the average burnup rates can be approached in the irradiation tests discussed, the peak burnup rate in the ANS fuel cannot be duplicated in any existing irradiation testing facility. The irradiation damage simulation by ion bombardment at very high dose rates will allow a determination of whether the irradiation behavior will change drastically at these higher burnup rates. This will allow use of modeling to predict the behavior of the fuel beyond regimes in which we can directly test the irradiation performance.
1.1.2.5	Final performance report—This task provides the integration of the data from the various fuel development subtasks in a documented form.

Table 3.2. (continued)

WBS	Task description
1.1.2.6	Plate and fuel element fabrication—The ANS uses a newly developed fuel compound, U_3Si_2. Development is required to determine the homogeneity levels achievable for the volume fractions of interest. At present the ANS requires a radial and axial fuel loading gradient. The achievement of the desired gradients with sufficient confidence and verification requires demonstration of new fabrication techniques.
1.1.2.7	Dummy plate and element fabrication—This task is to fabricate a dummy fuel element (Al-6061 plates with no fuel) for use in flow experiments. In addition, this task will provide prototype fuel plates for structural testing.
1.1.2.8	Prototype fuel element fabrication for critical experiments—This task is to fabricate a fully loaded prototype ANS core for use in performing critical experiments. The activities of this task include powder die and fixture development, engineering, and manufacture of an upper and lower fuel element.
1.1.2.9	Full-size plate irradiation—This task will complete the irradiation testing program for the ANS fuel with the irradiation of a few full-size plates.
1.1.2.10	Burnable poison selection and testing—This task will supply the material evaluations necessary to qualify the burnable poison material selection. Some demonstration of techniques to fabricate the burnable poison sections also will be included in this task.
1.1.2.11	Fuel element fabrication—This task will provide the effort needed to establish the production mode fabrication process. Although some cores may be fabricated under this task, it is not clear that they will be suitable for use in the ANS reactor.

[a]ANS=Advanced Neutron Source
[b]HFIR=High Flux Isotope Reactor

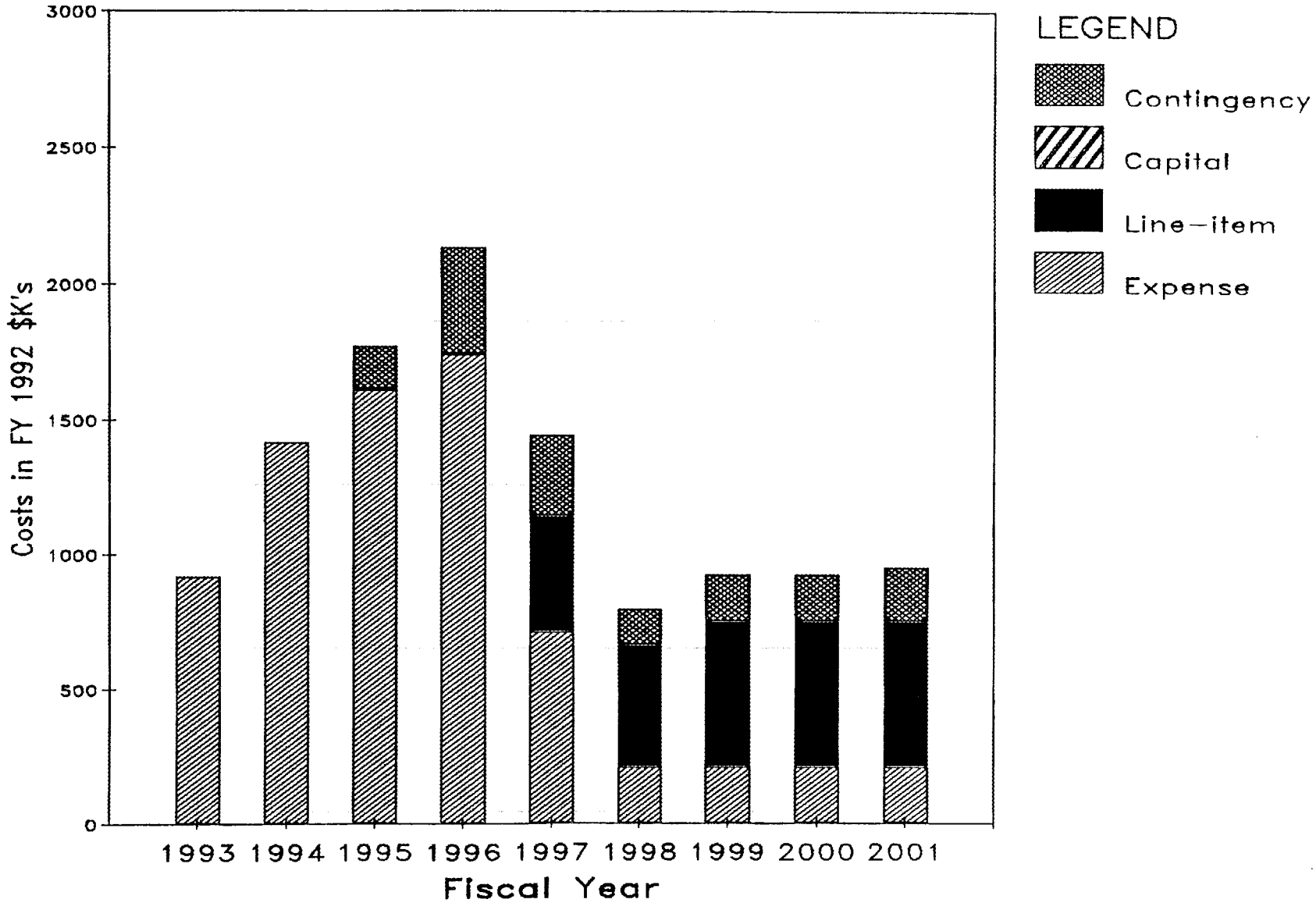

Fig. 3.3. Summary of cost estimates for work breakdown structure 1.1.2, fuel development.

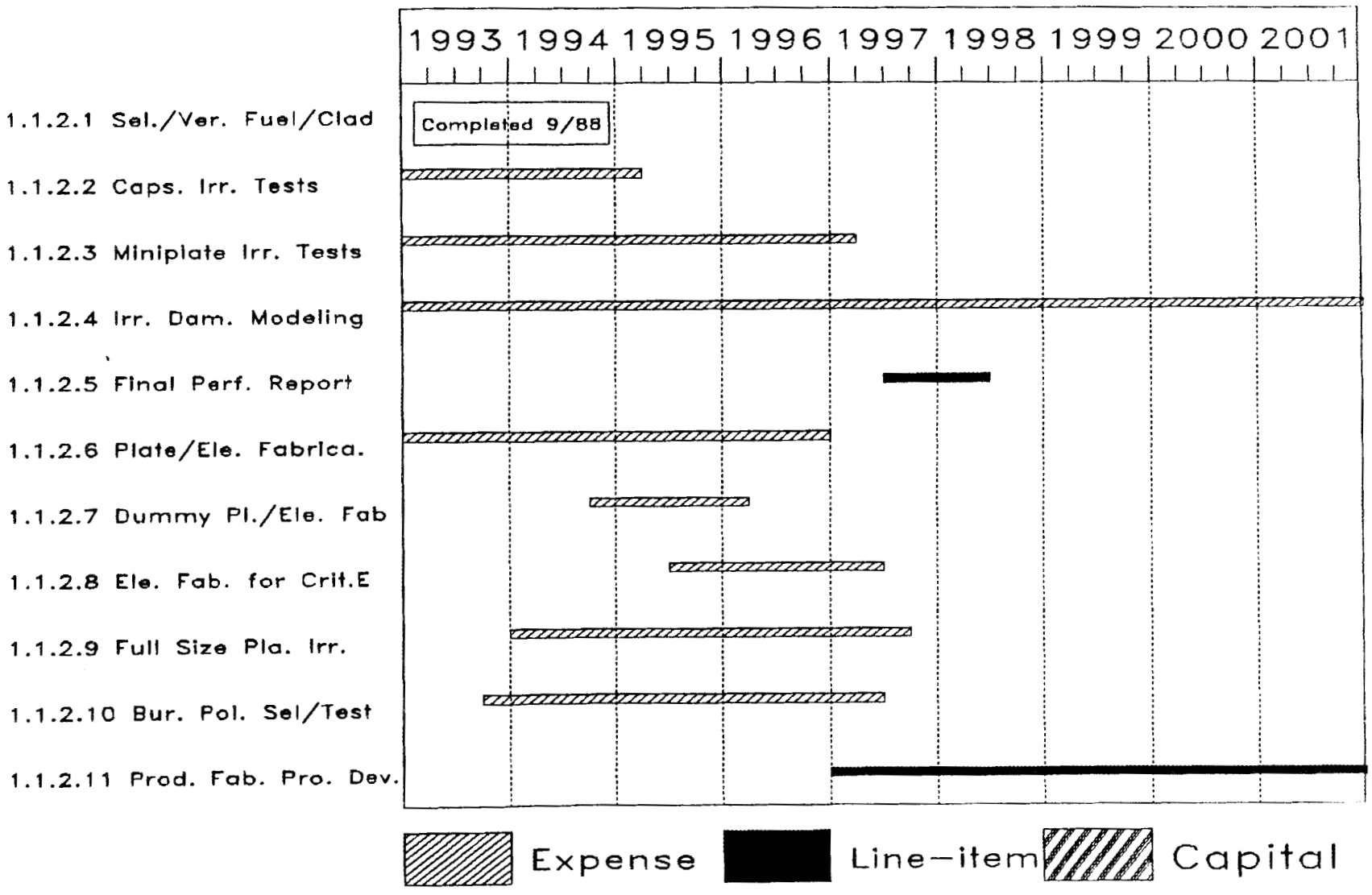

Fig. 3.4. Schedule for work breakdown structure 1.1.2, fuel development.

selected for the fuel and cladding early in the program so that scarce resources could be concentrated on their development and on verification that they meet the performance requirements of the reactor.

3.2.1.2 Description of the Selection and Verification of Fuel and Cladding Task

This task is considered to be complete. The purpose was to select the fuel and cladding materials thought to have the highest probability of meeting the requirements for the reactor. This task was completed early in the program so that resources could be concentrated on the chosen materials. The materials were chosen based on reviews of the materials used in research reactors through literature searches and consultations with experts in the fields of fuel fabrication, irradiation performance, and corrosion.

The premise was that the prime candidate materials should include those used successfully for many years in the HFIR, but that new developments also should be reviewed for potential improvements. For the cladding, the decision was made to stick with the 6061 aluminum alloy because the ANS Project has extensive experience with it in both fabrication and corrosion performance. Some newer alloys have the potential to improve fabricability, and others have the potential to improve corrosion performance. However, the 6061 alloy appeared to be the best cladding material for the ANS. A different situation existed for the fuel material. The reference core at the time this selection was made required a much higher specific uranium loading than was practical for the U_3O_8-Al dispersion used in the HFIR. The U_3Si_2-Al dispersion was chosen based on its higher density and proven good irradiation performance in tests in lower power reactors. Since this selection, the specific uranium density in the ANS has decreased to the level where U_3O_8 is also a viable option. However, U_3Si_2-Al remains the prime candidate fuel because of its higher thermal conductivity and stable irradiation performance at high burnups and temperatures. With the current core design, both U_3O_8 and UAl_x are viable backup fuel materials as dispersoids in aluminum.

Note that although this task is considered complete, it may be reopened if tests imply that the U_3Si_2 fuel or Al-6061 cladding do not perform as well as expected.

3.2.2 Capsule Irradiation Tests

3.2.2.1 Justification for Capsule Irradiation Tests Task

No irradiation data are available at the high temperatures and extremely high burnup rates of the ANS. Capsule irradiation tests can obtain these data at a variety of conditions relatively inexpensively because of the very small sample size.

3.2.2.2 Description of the Capsule Irradiation Tests Task

The capsule irradiation tests are to determine the irradiation behavior of fuel samples under temperatures and burnup rates approaching those expected in the ANS. Extremely small samples can be tested in the target region of the HFIR, where the burnup rates will be high. The desired temperature will be attained mainly from gamma heat in the capsule and a gas gap to limit conductivity.

Three tests are included in the program. The first has been irradiated and is about halfway through postirradiation examination. The capsule specimens consisted primarily of the prime candidate fuel at various temperatures and burnup rates (32 of 36 specimens). The second test will consist of the backup fuels U_3O_8 and UAl_x and the prime fuel U_3Si_2

under conditions as representative of the ANS as possible. The first two tests contain passive temperature monitors. The third test may be instrumented for temperature and will contain a combination of the three fuels irradiated under various conditions. The primary evaluation will be microstructural (metallography and scanning electron microscopy) to evaluate the compatibility of the fuel and matrix and the ability to retain fission gas in a stable bubble configuration.

3.2.3 Miniplate Irradiation Tests

3.2.3.1 Justification for the Miniplate Irradiation Tests Task

Although the fuel capsule irradiations provide much useful information about the behavior of the fuel, miniplates are preferred for accurate fuel swelling data and postirradiation heating tests. Therefore, plates typical of the ANS must be irradiated to beyond the burnup levels actually expected in order to determine structural stability.

3.2.3.2 Description of the Miniplate Irradiation Tests Task

These tests are to measure plate swelling and fuel-matrix, fuel-burnable poison-matrix, and meat-cladding interactions under irradiation to burnups typical of the ANS. Two tests are planned for the reflector region of the HFIR: Each test consists of a module containing several plates and each will be instrumented for temperature. The modules can be irradiated to include plates with a total burnup typical of the average and peak burnups expected in the ANS; however, the burnup rate, and possibly temperature, will be lower than in the ANS. The plates will be evaluated for general condition, swelling, microstructural stability, fission product retention, and stability during postirradiation heating.

3.2.4 Irradiation Damage Simulation Studies and Fuel Performance Modeling

3.2.4.1 Justification for Irradiation Damage Simulation Studies and Fuel Performance Modeling Task

There is no facility capable of testing the ANS fuel under its expected operating conditions of burnup rate, operating temperature, and burnup level. The planned tests have one or more parameters as close to these conditions as possible but cannot reach all three simultaneously. A fuel performance model is required to combine the results of the various tests and predict performance in the reactor regime.

3.2.4.2 Description of the Irradiation Damage Simulation Studies and Fuel Performance Modeling Task

This task is to develop a fuel performance model so that the behavior of the fuel system can be predicted in regimes beyond which facilities are available to test it. The mechanistic model will be based on the body of existing irradiation data for the U_3Si_2-Al system and will be continually tested and updated as the irradiation data are generated for the ANS Project. In addition, small specimens will be bombarded by krypton ions in the electron microscope and by neutrons in the Intense Pulsed Neutron Source at Argonne National Laboratory. These bombardments will provide information on the structural and crystallographic changes that occur during irradiation to determine the mechanisms of the

damage. Peer reviews of the models will be held periodically; they will be supported by this task.

3.2.5 Final Performance Report

3.2.5.1 Justification for the Final Performance Report Task

Because the fuel performance is an important safety aspect of the reactor design, the expected fuel performance must be well documented and reviewed. This task develops a formal fuel performance report that has undergone peer review and acceptance. The report produced will facilitate reviews of the ANS fuel performance and will provide a reference document for safety evaluations. It also will be needed for training of operational staff.

3.2.5.2 Description of the Final Performance Report Task

The final performance report will summarize the results of the fuel development and verification tests in a report for formal documentation. The report will be the primary responsibility of the ORNL task leader, but it will be a collaborative effort with the subcontractors, Argonne National Laboratory and Babcock and Wilcox. The report will contain detailed information on the fuel and cladding performance under all anticipated ANS conditions, including information on uncertainties. In addition, some data will be supplied on fuel behavior under conditions beyond those anticipated for the ANS for use in examining fuel behavior under severe accident conditions.

3.2.6 Plate and Fuel Element Fabrication

3.2.6.1 Justification for the Plate and Fuel Element Fabrication Task

The ANS uses a relatively new fuel compound with more stringent requirements than those for any fuel previously fabricated. The fabrication and inspection techniques must be developed and verified. In addition, since the ANS Project is using a probabilistic approach for treatment of uncertainties, distributions of variances for the major fabrication parameters must be established.

3.2.6.2 Description of the Plate and Fuel Element Fabrication Task

The plate and fuel element fabrication task is to develop the fabrication and inspection techniques that will result in fuel elements meeting the requirements of the ANS. The fabrication and inspection methods used for HFIR will, of course, be the starting point. However, some major differences are involved in fabricating the ANS elements. The U_3Si_2 fuel is pyrophoric and must be handled in an inert atmosphere up through the point of pressing it into compacts. The chemical reactivity also means that some special care must be taken to prevent oxidation of the fuel on the compact surfaces, especially during heating for hot rolling. In addition to the radial fuel grading as required in HFIR, the ANS design specifies a fuel gradient in the axial direction. The higher density of the silicide particles compared with the oxide particles also may mean that it will be more difficult to obtain homogeneous blends with the low-density aluminum powder. Because of the higher power density in the ANS, more rigid specifications have been assumed for the fuel distribution, bonding quality, and cooling channel spacings. These

development programs will be conducted at Argonne National Laboratory (where the silicide fuel was developed for lower powered reactors) and Babcock and Wilcox (the U.S. fabricators of fuel for HFIR and eventually for ANS).

3.2.7 Dummy Plate and Element Fabrication

3.2.7.1 Justification for the Dummy Plate and Element Fabrication Task

The high flow velocities in the ANS require flow testing of a prototypic element. Measurement of the structural properties of prototypic involute plates is required for accurate property input into the structural models for the elements. Therefore, dummy fuel plates and later complete dummy fuel elements need to be fabricated to support test programs in other portions of the R&D program. It should also be pointed out that even if these materials were not needed to support the testing program, their fabrication would still be a natural part of the fabrication development process.

3.2.7.2 Description of the Dummy Plate and Element Fabrication Task

Prototype dummy plates (aluminum alloy plates with no fuel) for structural testing and at least one upper core and one lower core dummy elements (fuel elements composed of dummy plates) will be fabricated for use in flow experiments. The fabrication task will be used in the fuel element fabrication development to arrive at die configurations for forming plates, tools and fixtures for assembling the element, and fixtures and procedures for welding the plates into the element while maintaining coolant channel dimensions.

3.2.8 Prototype Fuel Element Fabrication for Critical Experiments

3.2.8.1 Justification for Prototype Fuel Element Fabrication for Critical Experiment Task

Critical experiments are planned to verify physics performance of the ANS reactor core. This critical experiment, performed under WBS 1.1.6, requires a prototype upper and lower fuel element, which will be provided under this task. Without this task, the critical experiment could not be performed.

3.2.8.2 Description of the Prototype Fuel Element Fabrication for Critical Experiment Task

This task is to produce a fully loaded prototypic ANS core for use in critical experiments. Several plates will be removable or will have removable portions so that power profiles can be determined. Powder production, compact die design, plate manufacture and inspection techniques, and element assembly techniques need to be prototypic. A complete upper and lower element will be manufactured and shipped to the site where the critical experiments will be performed.

3.2.9 Full-Sized Plate Irradiation

3.2.9.1 Justification for the Full-Sized Plate Irradiation

Although the capsule and miniplate tests supply a good data base on the performance of the fuel under irradiation, review committees have indicated that irradiations of full-sized plates should provide a final verification of the satisfactory performance. The testing of full-sized plates will ensure that no warping or bowing occurs because of burnup or temperature variations. In addition, the full-sized plate irradiations will provide a final validation of the fuel performance models.

3.2.9.2 Description of the Full-Sized Plate Irradiation Task

This task is to verify the satisfactory irradiation performance of full-sized ANS plates in as near to prototypic conditions as possible. The tests are planned for the target region of HFIR. Evaluations of the fuel capsule and fuel miniplate tests will be performed before final machining of the full-sized plates. This task will complete the fuel irradiation testing program for the ANS fuel.

3.2.10 Burnable Poison Selection and Testing

3.2.10.1 Justification for the Burnable Poison Selection and Testing Task

Boron carbide is the burnable poison included in the conceptual design based on its satisfactory performance in HFIR. However, technical staff evaluations and fuel review committee comments indicate that the helium production may impact the structural performance of the plate. As a result, other burnable poisons that do not produce any gaseous products during operation are to be examined with respect to their compatibility with the fuel and cladding. This task provides the appropriate evaluations and testing necessary to qualify the burnable poison chosen for the ANS.

3.2.10.2 Description of the Burnable Poison Selection and Testing Task

This task is to select the appropriate material for use as the burnable poison and verify its satisfactory performance in the fuel plates. Boron carbide has served satisfactorily in the HFIR and other reactors, but it lowers the resistance to blistering in postirradiation heating tests. Poisons that have no gas generation may offer additional margin. The selection and testing task includes out-of-pile compatibility testing and fabrication feasibility testing for poison materials that have been selected based on neutronics desirability. The selected materials would be included in at least some of the miniplates and full-sized plates in the irradiation test program discussed under actions 3.2.3 and 3.2.9.

3.2.11 Production Mode Fabrication Process Development

3.2.11.1 Justification for the Production Mode Fabrication Process Development Task

Continued interaction between the fabricator and the reactor project will be necessary as production gets under way to ensure that the production process gives elements of adequate quality. Obtaining maximal quality and productivity while minimizing costs undoubtedly will require many process changes in the early years of production.

Without the process development task, the tolerance levels that can be obtained under a production mode condition cannot be ensured.

3.2.11.2 Description of the Production Mode Fabrication Process Development Task

This task is to provide the interaction between the fabricator and the reactor project while the production process for the fuel elements is being established, assist the fabricator in meeting the specifications, and modify the specifications if needed when modification will not sacrifice element performance or when process changes can improve element performance. The statistical data necessary to judge the quality of the plates and elements in a production mode will be collected and evaluated in this task, which will be the final phase in the fuel development program.

3.3 CORROSION TESTS AND ANALYSES—WBS 1.1.3

Aluminum alloy fuel cladding for high power density research reactors [e.g., HFIR, the Advanced Test Reactor (ATR)] has performed satisfactorily during long-term operation without serious problems due to oxide growth or corrosion. However, because some parts of the ANS fuel plates may operate under thermal-hydraulic conditions not previously encountered or explored experimentally, corrosion tests and analyses were initiated to investigate those processes and to develop a means to determine their importance and the magnitude of their effect on ANS reactor performance.

The basic objectives of the corrosion test program are to ensure that excessive fuel and clad temperatures due to corrosion product buildup do not occur during the lifetime of the ANS core and to ensure that the corrosion/erosion processes do not compromise the structural properties and containment capabilities of the fuel cladding.

To meet the above objectives, a corrosion test loop has been built that is capable of operation over the complete range of thermal-hydraulic conditions of importance to ANS operation. This test facility is being used to provide a data base for aluminum corrosion behavior.

The specific information that will be obtained includes

1. a set of experimental determinations of the extent and kinetics of corrosion product buildup and the associated corrosion of the basis 6061 aluminum alloy under thermal-hydraulic and coolant water chemistry conditions appropriate for ANS operation;

2. a mechanistic understanding of the corrosion process to support experimental data correlations and the evaluation of the sensitivity of the corrosion process to the ANS system variables;

3. the means to predict oxide film thicknesses and thus to judge the influence of oxidation and corrosion on the heat transfer and integrity of a fuel plate;

4. assurance that spalling and erosion of the protective product layer will not be a problem during the lifetime of the ANS core;

5. assurance that the use of heavy water as primary coolant and the intense radiation fields in the ANS core will not adversely impact the cladding corrosion behavior;

6. testing of alternate alloys or of the effectiveness of surface treatments if unmodified 6061 aluminum proves unacceptable or has marginal properties for a satisfactory fuel clad.

This WBS element contains one major project milestone:

Define water chemistry requirements by March 1996 to allow the Title II design of the reactor water systems and chemical control systems to proceed on schedule.

The corrosion tests and analysis activity is divided into five WBS level four tasks that are summarized in Table 3.3. With the exception of some ATR analysis work performed at INEL, all of this work will be performed at ORNL. The total estimated costs for this activity over the 9-year period covered by this R&D plan are given in Fig. 3.5 and the associated schedules are shown in Fig. 3.6. Note that the expense costs are associated with the development of the data base and empirical correlations, while the line-item costs are used for prototype testing, analysis, and documentation that directly support preliminary (Title I) and detailed (Title II) design activities. Subsections 3.3.1 through 3.3.5 provide more detailed information on the WBS level four tasks under this activity.

3.3.1 Design/Procurement and Installation of Test Loop

3.3.1.1 Justification for the Design/Procurement and Installation of Test Loop Task

No existing facility was found that could provide the testing capabilities needed to generate the required test data. Therefore, a facility capable of testing aluminum alloy specimens over a wide range of heat flow, coolant flow rates, and coolant properties was required to generate an acceptable experimental data base supporting the task objectives.

3.3.1.2 Description of the Design/Procurement and Installation of Test Loop Task

Almost all the work under this task has been completed. The design, construction, installation, and startup testing of the ANS corrosion test loop were essentially finished in FY 1988. One significant modification to the loop is planned, and funding under this subtask is allocated for FY 1994: The loop will be modified for heavy water use to allow confirmatory heavy water tests.

3.3.2 Performance of Loop Experiments

3.3.2.1 Justification for Performing the Loop Experiments Task

The justification for performing these tests was established in the introductory paragraphs of Sect. 3.3. The experiments involved in meeting the test requirements are difficult ones. Thus, extreme care is required in performing the tests to obtain accurate data that will provide meaningful and consistent results.

Table 3.3. Summary description of the corrosion tests and analysis work breakdown structure level four tasks

WBS	Task description
1.1.3.1	Design/procure and install test loop—This task includes all activities needed to design and construct the facility needed to perform the Advanced Neutron Source corrosion tests. Most of this activity has been completed using capital equipment money, but future upgrades to the system are expected that would be performed using expense funding.
1.1.3.2	Loop experiments—This task includes all the activities necessary to perform the planned out-of-pile corrosion tests. Operation staff, maintenance, and test specimens are covered under this task.
1.1.3.3	Analysis of experimental data—This task includes all the analytical analysis, nondestructive testing, and evaluations necessary to understand the data obtained from the out-of-pile corrosion tests.
1.1.3.4	Analysis of ATR[a] oxide data—This task was established to examine exiting data on oxide film growth obtained in the ATR under ATR conditions. This task was completed in FY 1989, and no additional work is planned.
1.1.3.5	Comparison tests under irradiation conditions—This task includes the design, preparation, and evaluation of an oxide growth test performed in-pile in the HFIR[b] facility, along with a comparable test performed out-of-pile in the corrosion test loop facility.

[a] ATR=Advanced Test Reactor
[b]High Flux Isotope Reactor

3.3.2.2 Description of the Performance of the Loop Experiments Task

This task includes all the activities to procure and prepare test specimens, fabricate the test sections, and perform the tests. In addition to test surveillance, operation and maintenance of the test loop are included.

More than 50 corrosion tests are planned. A typical ANS corrosion test loop experiment will involve exposure of the specimen to carefully controlled resistance heating and cooling water flow conditions for up to 350 hours. During this time, the outer surface temperature of the specimen will be monitored at several points along its length to determine the added resistance to heat flow due to any growing corrosion product. Therefore, for a typical experiment, it is anticipated that the temperature will vary as a

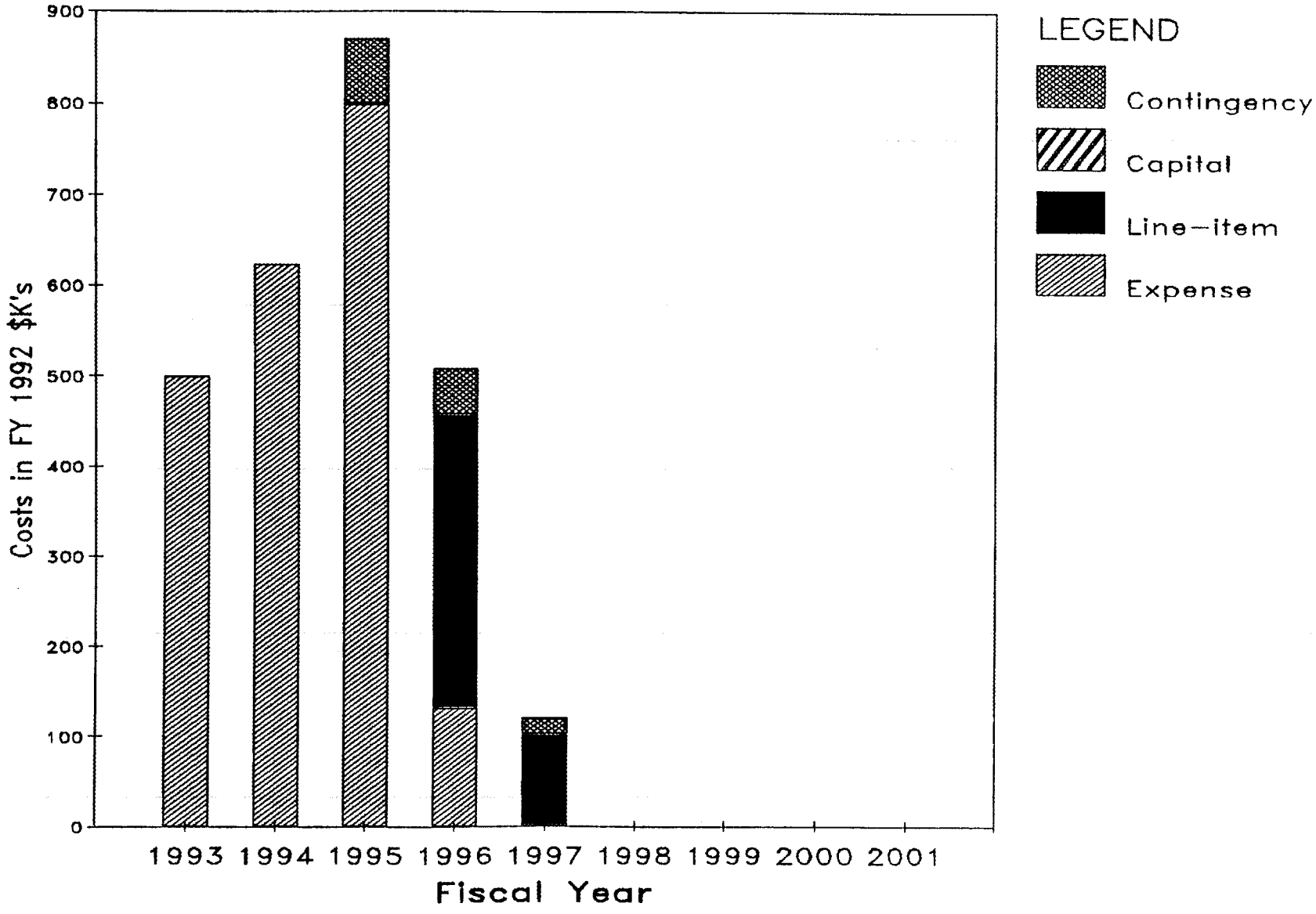

Fig. 3.5. Summary of cost estimates for work breakdown structure 1.1.3, corrosion tests and analysis.

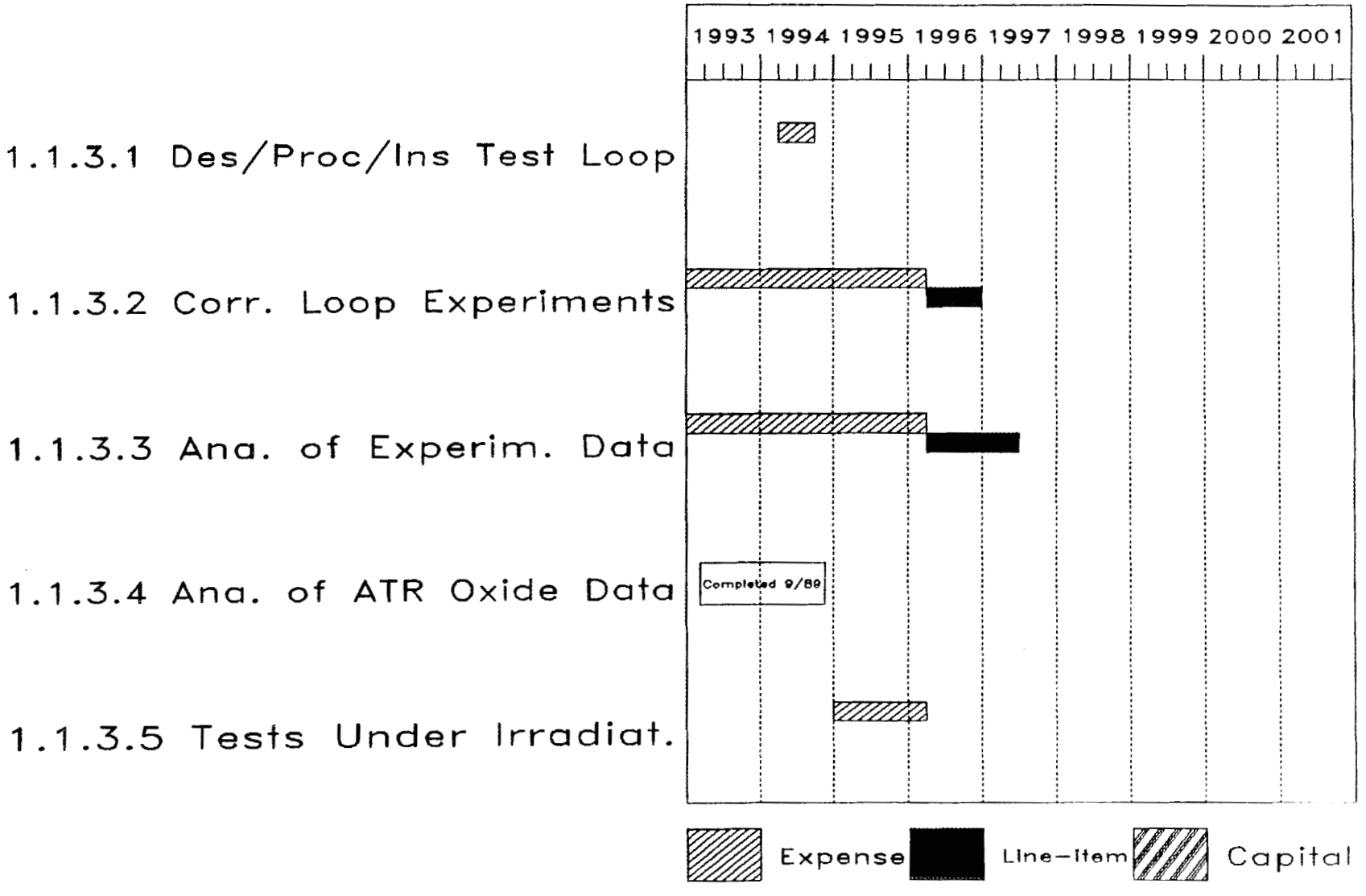

Fig. 3.6. Schedule for work breakdown structure 1.1.3, corrosion tests and analysis.

function of position along the specimen in the direction of the water flow, across the specimen in the direction of the heat flow, and over time if a low thermal conductivity corrosion product develops. In addition, the rate of growth of the corrosion product will itself be a function of temperature so that the gradients in specimen temperature also will change over time.

3.3.3 Analysis of Experimental Data

3.3.3.1 Justification for the Analysis of Experimental Data Task

The examination of the experimental data generated during the tests and of the aluminum specimens after testing constitutes a critical part of the program and serves as the basis for all observations and results. The establishment of appropriate data correlations is a necessary contribution to reactor design. Correlations to support analytical reactor performance models must be developed because all potential operating conditions can not be tested.

3.3.3.2 Description of the Analysis of Experimental Data Task

This task includes the disassembly of the test section and the various examinations of the specimens and the corrosion products. Calculations are performed on the various data obtained during the tests to describe the film growth process in terms of the thermal-hydraulic and water chemistry parameters. In addition, all activities necessary to evaluate the results and establish data correlations are accomplished under this task.

3.3.4 Analysis of Advanced Test Reactor Oxide Data

3.3.4.1 Justification of the Analysis of Advanced Test Reactor Oxide Data Task

Measurements of oxide thicknesses on aluminum cladding in many ATR cores were made by INEL throughout years of operation of this reactor. These data constitute an important source of information for oxide product characterization for ATR conditions and could, like certain HFIR data, serve as additional information important to ANS applications.

3.3.4.2 Description of the Analysis of Advanced Test Reactor Oxide Data Task

This task was completed in 1989, and no future work is planned. The film thickness data for a series of ATR fuel loadings were compiled. Calculations based on reactor operation information were used to associate each film measurement with local reactor conditions, and the results were compared with those predicted by the Griess correlation.

3.3.5 Comparison Tests Under Irradiation Conditions

3.3.5.1 Justification for Comparison Tests Under Irradiation Conditions Task

The potential effect of nuclear radiation on the kinetics of oxide film growth on aluminum clad fuel cores has not been resolved with certainty. While the operating

experiences with aluminum cladding seem to confirm that no large effects exist at the neutron fluxes and fluences involved, a more direct comparison to furnish additional evidence would be useful.

3.3.5.2 Description of Comparison Tests Under Irradiation Conditions Task

This task presently includes the design and preparation of comparison tests and their exposure in the HFIR facility under conditions to simulate (as closely as possibly) typical ANS operation. Duplicate out-of-pile tests will be made in the ANS corrosion test loop using the same parameters in an attempt to identify differences in the film growth behavior as a result of radiation, per se.

3.4 CORE FLOW TESTS—WBS 1.1.4

The core flow tests activity includes the tests and analyses necessary to validate the thermal-hydraulic analytical models. Test facilities will be constructed to perform tests that will include examining the thermal-hydraulic parameters for nominal and hot channel fuel plates for ranges of normal and anticipated ANS operating conditions. These tests will provide the data base necessary to validate/develop the correlations and models used to analyze the reactor core thermal hydraulics.

This WBS element does not contain a major project milestone. The work is divided into two WBS level four tasks summarized in Table 3.4. All this work is expected to be performed at ORNL. The total estimated costs for this activity over the 9-year period covered by this R&D plan are given in Fig. 3.7, and the associated schedules are shown in Fig. 3.8. It should be noted that the capital equipment money is associated with the design and construction of a natural circulation test loop. Subsections 3.4.1 and 3.4.2 provide more detailed information on the WBS level four tasks under this activity.

3.4.1 Thermal-Hydraulic Test Facility

The thermal-hydraulic test facility task was initiated in FY 1991 with the design of a test loop. Since then the test loop has been built and tests have been initiated.

3.4.1.1 Justification for the Thermal-Hydraulic Test Facility Task

The availability of critical heat flux and flow instability data at normal ANS operating conditions (e.g., flow rate, pressure, geometry) is very limited. The primary purpose of the thermal-hydraulic test loop is to acquire such data over a range of conditions representative of the ANS reactor operating conditions. If these data are not obtained, uncertainties and additional margins would have to be included in the design that could preclude meeting performance requirements.

3.4.1.2 Description of the Thermal-Hydraulic Test Facility Task

To characterize the thermal-hydraulic performance of the ANS reactor, certain key parameters and/or the accuracy of their prediction must be known. This task will include the measurements and analyses necessary to validate thermal-hydraulic models that, in

Table 3.4. Summary description of the core flow tests work breakdown structure level four tasks

WBS	Task description
1.1.4.1	Thermal-hydraulic test facility—To characterize the thermal-hydraulic performance of the Advanced Neutron Source reactor, key parameters and/or the accuracy of their prediction must be known. This task will include the measurements and the analysis necessary to validate thermal-hydraulic models that, in turn, will be used to assess the Advanced Neutron Source thermal-hydraulic conditions under normal and anticipated transient conditions.
1.1.4.2	Natural circulation tests—These tests will be used to examine issues related to natural circulation within the primary system. The design will encompass phenomena such as loop-to-loop interactions under free convection conditions, as well as accumulator-to-accumulator interactions. Initial conditions within the flow loops will be varied (i.e., one pump off with the others operating, two pumps off) to determine the worst case conditions for core and component cooling.

turn, will be used to assess the ANS thermal-hydraulic conditions under normal and anticipated transient conditions. (Note that assessments of thermal-hydraulic conditions for unlikely events and severe accidents will be performed as part of the safety R&D tests in WBS 1.1.14.) The loop is designed to test a full-length aluminum coolant subchannel of the ANS core. Tests in this facility will provide both single- and two-phase thermal-hydraulic information. The specific phenomena that are to be examined in this loop are (1) single-phase heat transfer coefficients and friction factors, (2) the point of incipient boiling, (3) nucleate boiling heat transfer coefficients, (4) two-phase pressure drop characteristics in the nucleate boiling regime, (5) flow instability limits, and (6) critical heat flux limits. Currently, 50 tests are planned under this subtask.

3.4.2 Natural Circulation Tests

3.4.2.1 Justification for the Natural Circulation Tests

Natural circulation in plate-fueled research reactors is not well understood and deserves a careful experimental evaluation because decay heat removal after loss of flow is associated with many initiating events. The ANS reactor is designed to provide natural convection cooling after shutdown for these initiating event scenarios. Multi-loop designs with the potential for different flow conditions in different loops may cause asymmetries that can affect the stability and behavior of the total system. Experimental validation and/or development of models and correlations that can predict the system response under natural convection conditions is necessary to ensure the capability for cooling under shutdown conditions.

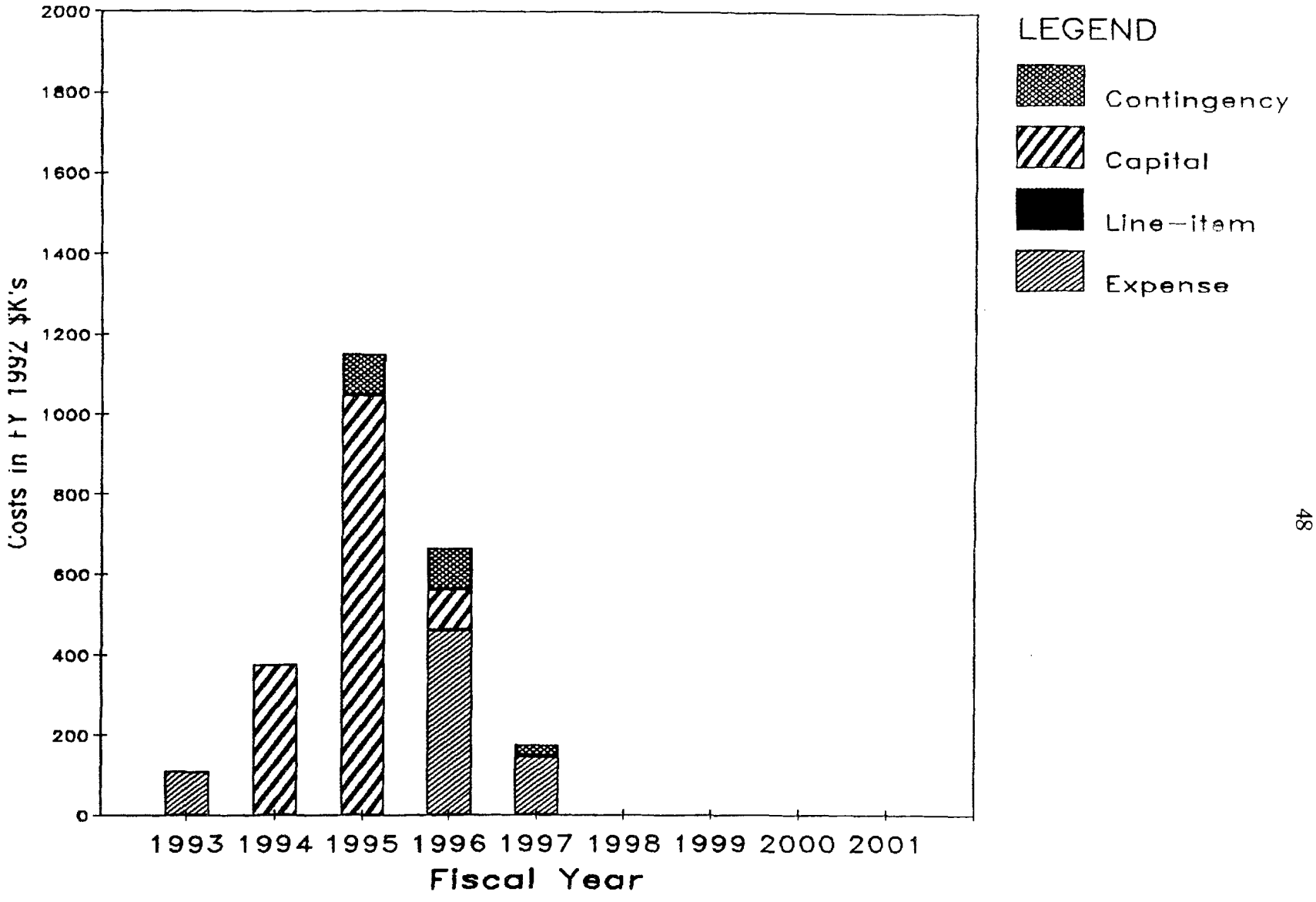

Fig. 3.7. Summary of cost estimates for work breakdown structure 1.1.4, core flow tests.

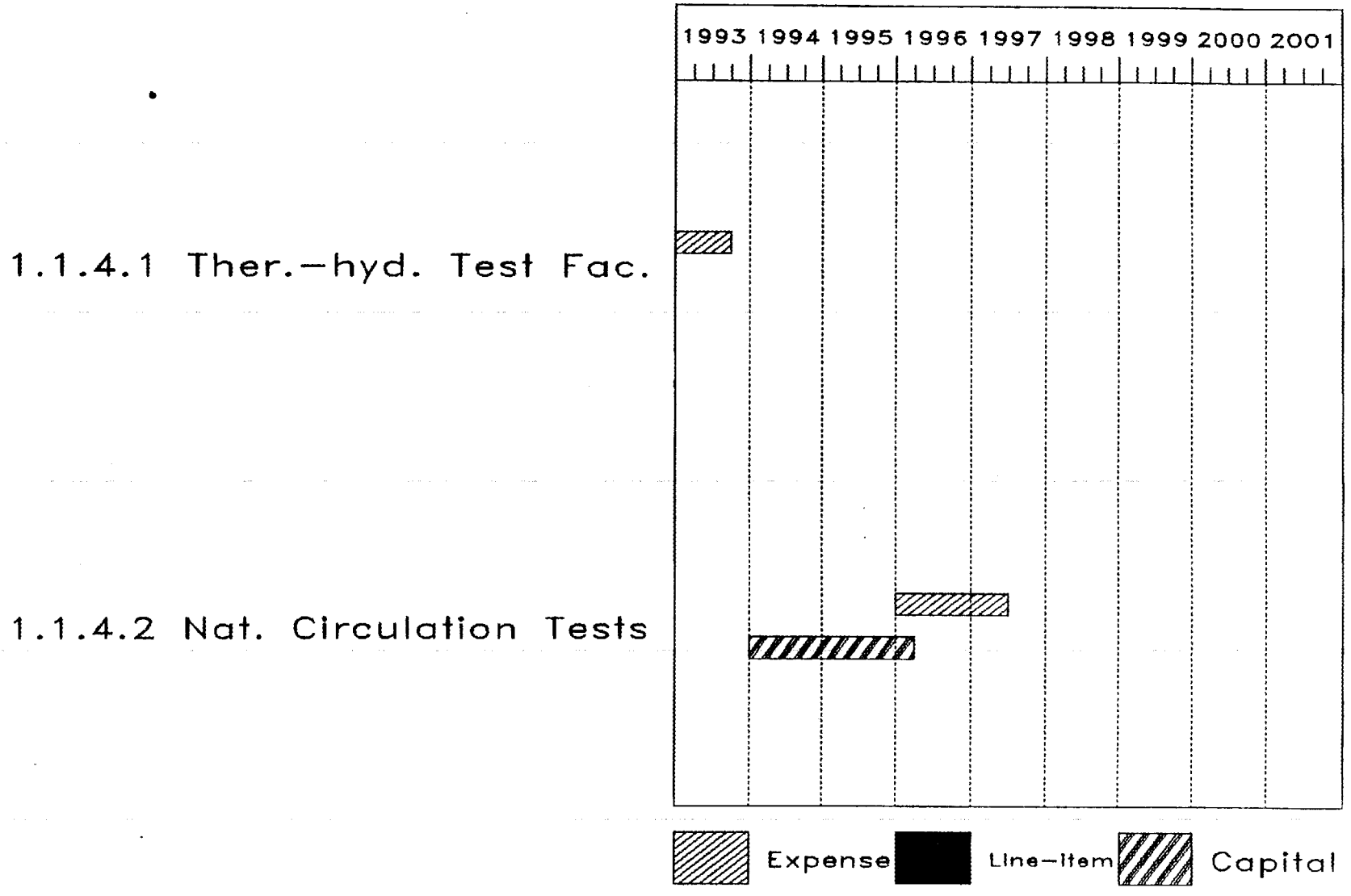

Fig. 3.8. Schedule for work breakdown structure 1.1.4, core flow tests.

3.4.2.2 Description of the Natural Circulation Tests

The tests proposed under this subtask will be used to examine issues related to natural circulation within the primary system. The facility design will encompass such phenomena as loop-to-loop interactions under free convection conditions and accumulator-to-accumulator interactions. It is not expected that this facility will have to be full scale, but it will have to incorporate a heated region to simulate the natural circulation conditions properly within the reactor primary coolant loops. Initial conditions within the flow loops will vary (i.e., one pump off with the other operating, two pumps off) to determine the worst case conditions for core and component cooling. Interactions between accumulators will be studied by incorporating scaled accumulators in the facility. Analysis of data from these tests will be used to evaluate natural circulation models. Other issues, such as natural circulation internal to the coolant piping, will be addressed as necessary.

3.5 CONTROL CONCEPTS—WBS 1.1.5

The development of a concept for reactivity control of the reactor core is an integral part of the development of the reactor core. However, it was recognized that reactivity control issues could receive more emphasis and visibility if they were a separate activity from reactor development (WBS 1.1.1). Thus the control concepts activity (WBS 1.1.5) was created. It focuses on the reactor physics aspects of reactivity control and still is considered to be highly coupled to the reactor development activity. The work includes the reactor physics analyses necessary to develop the concept for reactivity control and to perform an initial detailed analysis of the concept. Further reactor physics analyses considered necessary to support the design of the reactivity control system and the FSAR are covered under WBS task 1.1.1.4.

The control concepts WBS element does not contain a major project milestone, although the work performed under this WBS is in direct support of the major project milestones previously discussed for WBS 1.1.1. The work is divided into the two WBS level four tasks in Table 3.5, and it will be performed at INEL, the Massachusetts Institute of Technology, and ORNL. The total estimated costs over the 9-year period covered by this R&D plan are given in Fig. 3.9, and the associated schedules are shown in Fig. 3.10. The line-item money is associated with the detailed analyses of the developed concepts that directly support preliminary (Title I) and detailed (Title II) design activities. Subsections 3.5.1 and 3.5.2 provide more detailed information on the WBS level four tasks under this activity.

3.5.1 Static Analyses of Control and Safety Poison Systems

3.5.1.1 Justification of Static Analyses of Control and Safety Poison Systems

The analyses of control and safety poison systems provide the reactor physics evaluations needed to support the design of the reactivity control and safety shutdown systems. To proceed with the reactivity control system design, it is necessary to know both the control materials and their performance in a neutron and temperature environment similar to the expected ANS reactor operating conditions. Without this R&D activity, the ability to control the nuclear reaction during normal operation cannot be demonstrated.

Table 3.5. Summary description of the control concepts work breakdown structure level four tasks

1.1.5.1	Static analysis of control and safety poison systems—This task includes all the analyses necessary to characterize both the reactivity control and the reactivity safety systems. Activities under this task include the static neutronics analysis necessary to make decisions on control geometry, poison material selection, and control and shutdown poison location. In addition, this task will supply the neutronics support to the design of the control and safety poison systems.
1.1.5.2	Reactor kinetics—This task is to determine appropriate kinetics analysis methods and to supply the kinetic parameters necessary for the analyses. Included is the examination of the limits to point kinetics and quasi-steady state kinetics analysis, as well as the need for one-dimensional, two-dimensional, and/or three-dimensional time dependent analyses in the special reactor physics environment of the Advanced Neutron Source. Methods for treating photoneutron production also will be identified. The parameters to be determined include neutron lifetime, effective delayed neutron fraction, and photoneutron production.

3.5.1.2 Description of the Static Analyses of Control and Safety Poison Systems Task

The environment in which the control and safety shutdown systems are expected to operate is expected to be more hostile than that of either the ILL or the HFIR reactor. Therefore, the systems used in those reactors are not directly transferable to the ANS. Various design options for reactivity control and shutdown (use of burnable poison, reflector control, conventional mechanical poison, liquid poison, and combinations) will be assessed under this task, including the geometry and location alternatives. Once the preferred option for the control system and the preferred option for the shutdown system have been identified, the poison material must be chosen. Issues to be addressed for material selection include reactivity worth, burnup rate, behavior of daughter products, amount of nuclear heat deposited during operation, decay heat, fabrication techniques, mechanical properties, and corrosion resistance. The reactor physics issues will be addressed under this task, and the remaining issues will be examined under other elements of the R&D program. The materials considered initially will include all poison materials previously used in reactors, with emphasis on those previously used in other high flux reactors. Once the control and safety shutdown systems have been defined, the performance of each system will be evaluated in detail. By the end of FY 1993, the basic analyses needed to make advanced conceptual design decisions are expected to be completed. The work in FY 1994 and beyond will be devoted to developing realistic worth curves for mechanical poison systems throughout the fuel cycle, determining heat loads over the lifetime of the rods, analyzing control activation, determining the impact of vibration in mechanical poison systems on the core reactivity, and determining transmutation products over the life of the poison systems for use in material property

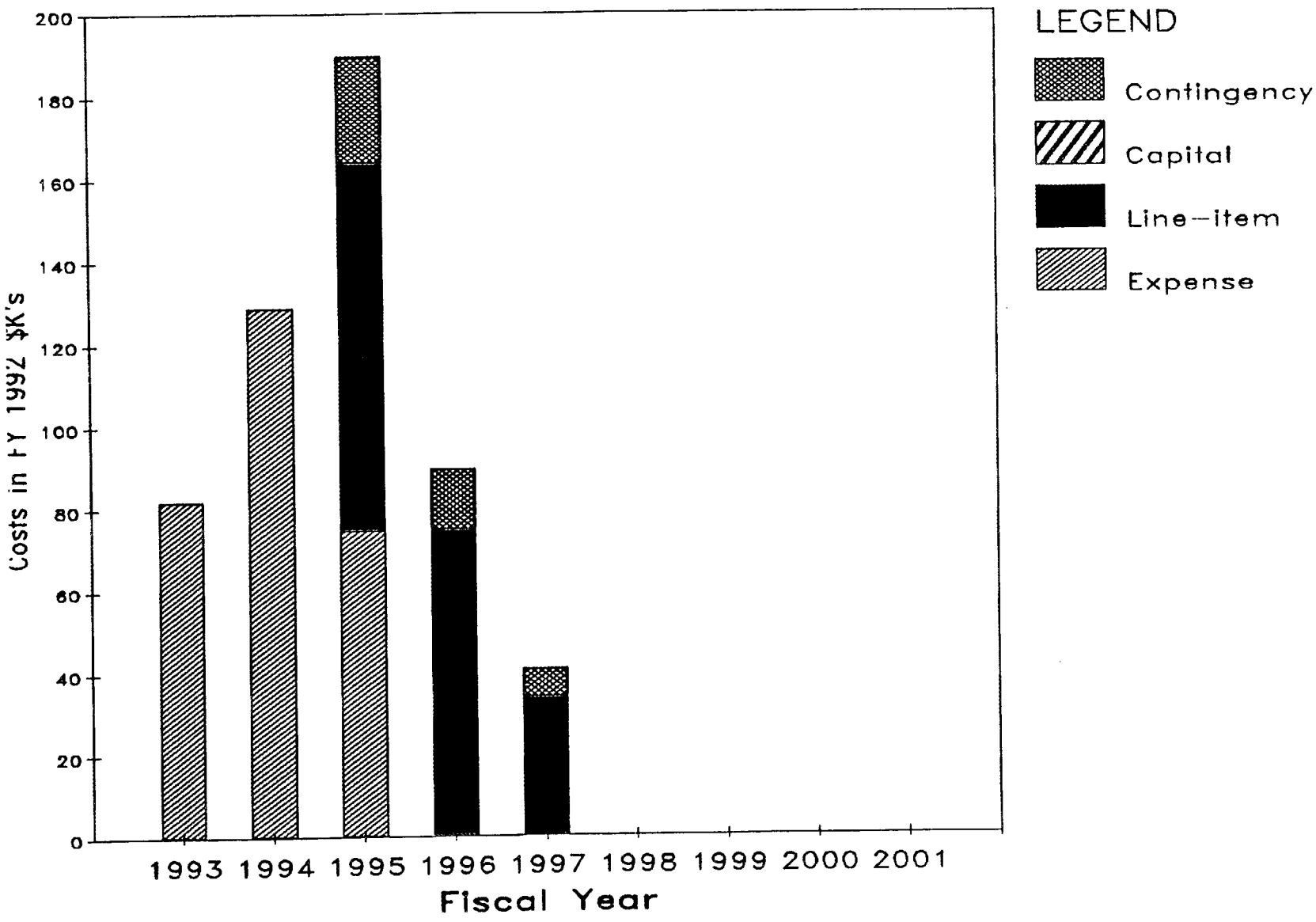

Fig. 3.9. Summary of cost estimates for work breakdown structure 1.1.5, control concepts.

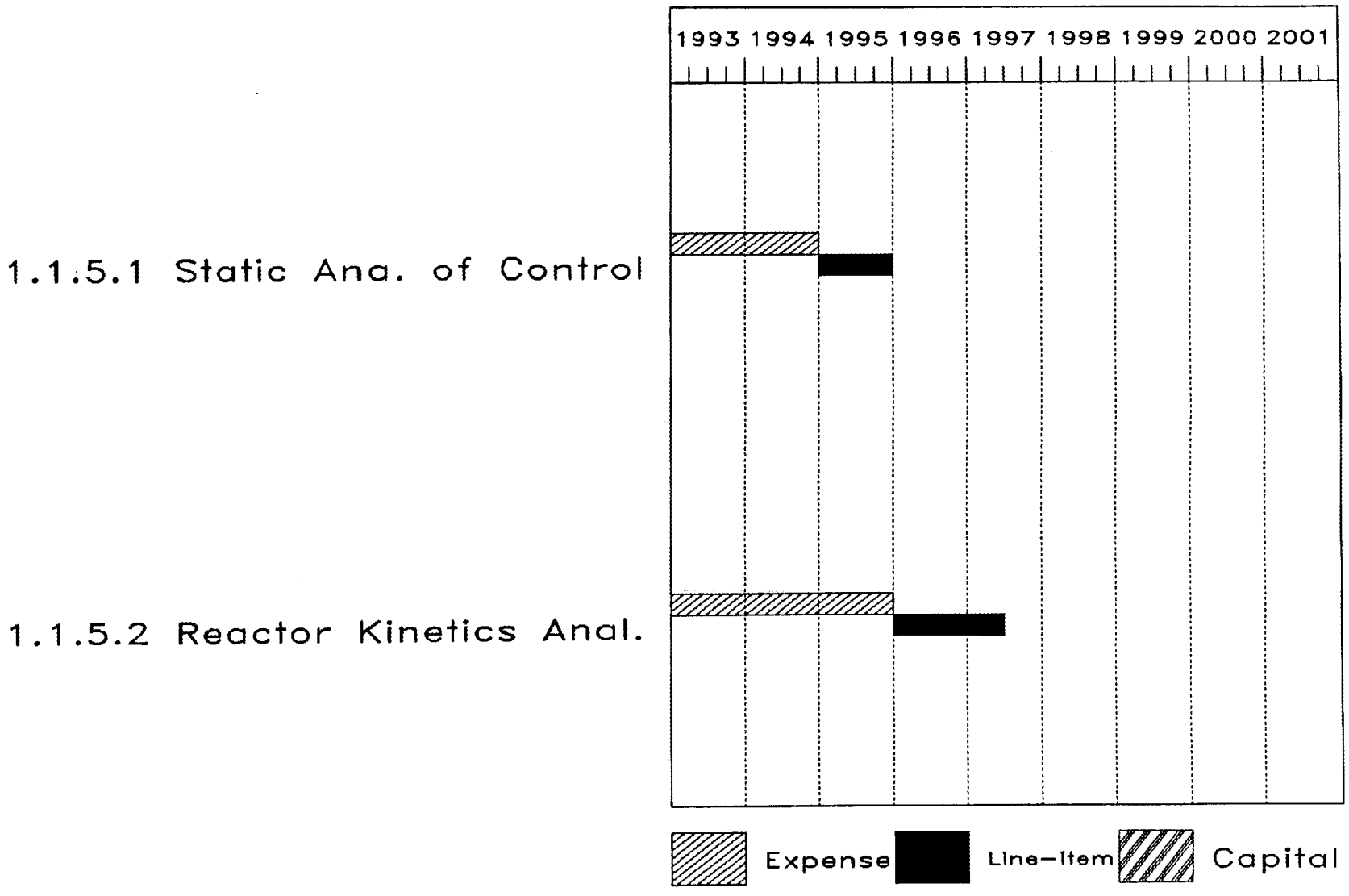

Fig. 3.10. Schedule for work breakdown structure 1.1.5, control concepts.

evaluations. Any additional work (e.g., documentation, responding to questions) beyond the scheduled completion of this task in FY 1997 will handled by the general support task under WBS 1.1.1.

3.5.2 Reactor Kinetics Analyses

This task is to perform appropriate kinetics analyses and to supply the kinetic parameters necessary for the analyses.

3.5.2.1 Justification of the Reactor Kinetics Analyses

Because of its short reactor core, high coolant velocity, large heavy water reflector, and high fuel burnup over short periods of time, the modeling methods needed to simulate the kinetics behavior of the ANS reactor adequately is not known. Therefore, this task was identified to develop and demonstrate a methodology for kinetics analysis of the ANS core. Without this task, the neutronic behavior of the ANS reactor core during transient conditions could not be adequately determined.

3.5.2.2 Description of the Reactor Kinetics Analyses Task

This task compares the various approaches for treating reactor kinetics (i.e., point kinetics; quasi-steady state kinetics; and 1D, 2D, and 3D space-time treatment) and provides a technique that is considered appropriate for ANS analyses. This task also includes the calculations necessary to determine the kinetics parameters—such as neutron lifetime, effective delayed neutron fraction, and photoneutron production—that are needed to perform the kinetics analyses.

3.6 CRITICAL EXPERIMENTS—WBS 1.1.6

The critical experiments activity includes all tasks needed to plan, perform, and analyze a set of critical experiments that will be used to benchmark the performance of the ANS core. As discussed in the section on methods development for reactor physics, there are three phases to the process being used in the ANS Project to validate the reactor physics models. The third phase of the validation process is the performance of critical experiments with a geometry as close to the ANS geometry as is reasonably practical. Prototypic fuel elements, control rods, and other key components would be an important aspect of these experiments. In addition to the actual tests, critical experiment design, preanalysis, and postanalysis all would be performed under this activity.

This WBS element contains one major project milestone:

Complete critical experiments by June 1998. This schedule allows at least 2 years to evaluate the results before the loading of fuel into the ANS is expected.

The critical experiment activity is divided into five WBS level four tasks summarized in Table 3.6. Most of the work under this activity would be performed at an as yet undefined location (facility chosen to perform the critical experiments), and some analysis

Table 3.6. Summary description of the critical experiments work breakdown structure level four tasks

WBS	Task description
1.1.6.1	Develop experimental plan for critical experiments—The purpose of this task is to provide a detailed experimental plan for a critical experiments program for the ANS core. This plan includes selection of site to perform experiments, detailed cost estimates of facility modifications required to conduct experiments, description of experimental parameters to be measured, description of organizational structure for completing the critical experiments task, and a detailed total experiment cost and schedule.
1.1.6.2	Preanalysis of critical experiments—Estimates of the parameters to be measured in the critical experiments will be generated using selected computational methods and data. This analytical analysis will be used to better define the actual measurement steps and to provide the experimenters with indications of the measurement values expected.
1.1.6.3	Modifications of facilities for critical experiments—Regardless of the site selected for conducting the critical experiments, modifications to existing facilities will have to be made. An experimental rig must be fabricated and necessary paper work including safety analysis studies must be completed. The costs for the fabrication of the fuel elements to be used in the experiments are included as a subtask in WBS 1.1.2.
1.1.6.4	Perform critical experiments—The purpose of this task is to conduct critical experiments and measure physical parameters necessary to confirm the ANS core performance.
1.1.6.5	Analysis of critical experiments—The purpose of this task is to provide analytical analysis of the critical experiments. Activities under this task include identification of bias factors and resolution of any significant differences between calculated and measured parameters.

work would be performed at INEL and ORNL. The total estimated costs (all expense money) for this activity over the 9-year period covered by this R&D plan are given in Fig. 3.11, and the associated schedules are shown in Fig. 3.12. Subsections 3.6.1 through 3.6.5 provide more detailed information on the WBS level four tasks under this activity.

3.6.1 Development of an Experimental Plan for Critical Experiments

3.6.1.1 Justification for Development of an Experimental Plan for Critical Experiments

Before a critical experiment program can be initiated, a good experimental plan must be developed. The parameters that will be measured, along with the experimental conditions, must be known before legitimate discussions of a site for the experiments can be held. Without this task, the risk of not getting the required data from the tests is greatly increased.

3.6.1.2 Description of the Development of an Experimental Plan for Critical Experiments Task

A detailed list of measurements that could be made as part of the critical experiments will be prepared under this task. The measurements on the list then will be divided into three categories: (1) those that are absolutely required to validate reactor physics performance, (2) those that will greatly reduce the uncertainty and make the safety case more defensible, and (3) those that would be desirable if permitted by cost and schedule. That information then will be used to evaluate the various site location options for performing the tests. Any site considered for the tests must be capable of performing at least all the tests within category one.

Selection of the site for performing these tests will be covered under this task. Discussions will be held with the staff of proposed sites to determine the feasibility, practicality, and costs of performing experiments.

After a site has been selected, a final test plan will be developed specifically for the selected site based on discussions between the staff of the selected site and the ANS Project staff. This plan will be reviewed externally and internally before it is issued in 1995 as the reference plan.

3.6.2 Preanalysis of Critical Experiments

3.6.2.1 Justification for Preanalysis of Critical Experiments

The preanalysis of the critical experiments serves three purposes:

1. One of the intentions of the critical experiments will be to document how well the reactor physics methods can be used to predict behavior. Therefore, experiment steps are preanalyzed to provide a data base for comparison with measured data.

2. The practicality of various measurements must be determined. For instance, if we propose a small perturbation (e.g., temperature changes, flooding of a beam tube, removal of a fuel plate), it must be determined that the perturbation is large enough to measure with acceptable confidence. If the size of the perturbation is on the same level as the uncertainty in the measurement, the proposed experiment step may not provide much useful information. Thus each measurement must be pre-analyzed to determine the practicality of the test.

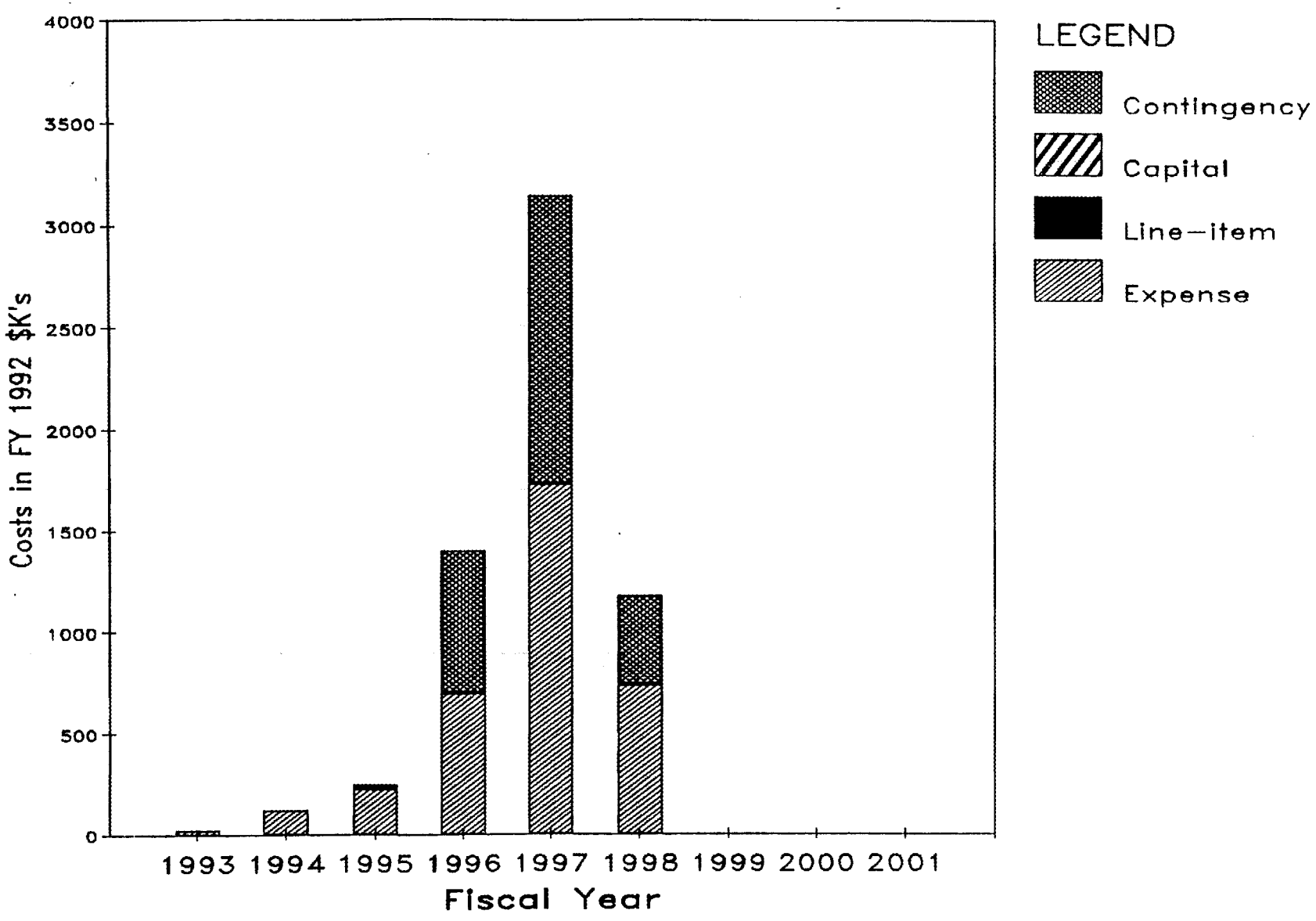

Fig. 3.11. Summary of cost estimates for work breakdown structure 1.1.6, critical experiments.

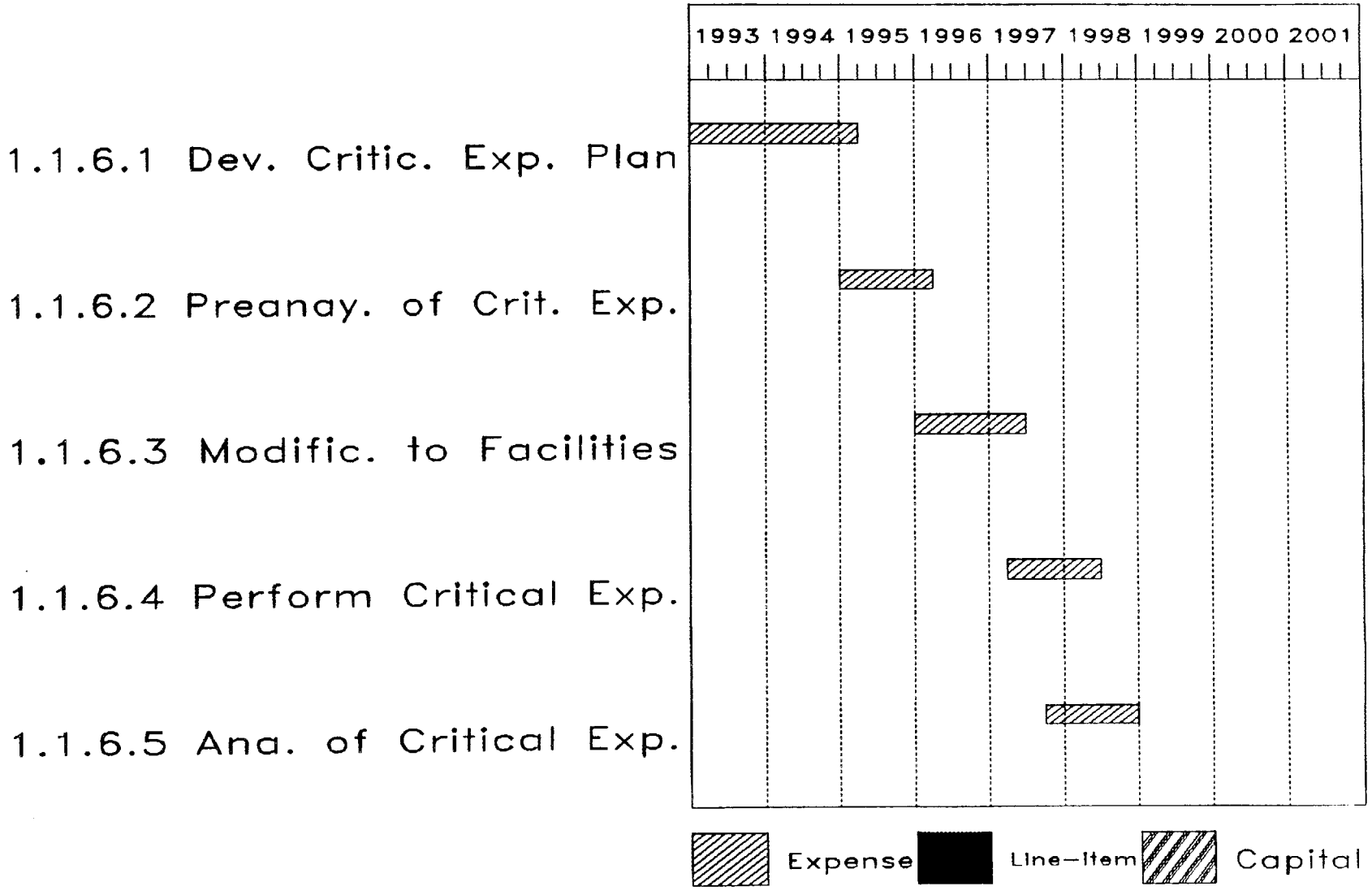

Fig. 3.12. Schedule for work breakdown structure 1.1.6, critical experiments.

3. The site performing the tests will require that certain analyses be performed to ensure that their facility's safety rules are not violated during the tests. These preanalyses will provide that information, also.

3.6.2.2 Description of the Preanalysis of Critical Experiments Task

This task will provide all the analyses necessary to understand the various proposed tests in the critical experiment plan. These analyses will be completed and documented before actual testing begins. Included in the studies covered by this task will be the comparison of perturbations introduced in the ANS geometry and in the critical experiments geometry to determine the level of fidelity associated with the critical experiments.

3.6.3 Modifications of Facilities for Critical Experiments

3.6.3.1 Justification for Modifications of Facilities for Critical Experiments

No facility exists that can directly accommodate the experiments expected to be proposed in the critical experiments plan. Therefore, some modifications to the facility at the chosen site are expected before the experiments begin. This task provides the effort to examine the types and extent of modifications needed to complete the critical experiment plan adequately.

3.6.3.2 Description of the Modifications of Facilities for Critical Experiments

Although the extent of modification required to perform the tests will be one of the criteria for selecting the site, it will not be the only one. Therefore, the extent of modifications required may range from simple to extensive. A plan for the modifications will be developed jointly by the staff of the proposed site facility and the ANS Project staff, and modifications will be made on a cost-effective basis. The proposed facility modification plan, including costs and schedule, will be reviewed internally and externally before modifications are initiated.

3.6.4 Perform Critical Experiments

3.6.4.1 Justification for Critical Experiments Task

This task provides the measured data that will be used as the third phase of the validation of the reactor physics analysis methods and provides a direct measurement of parameters (e.g., heat loads, fluxes) that are extremely important to the designers, safety reviewers, and users. Although these tests occur late in the project, they are considered critical to the confirmation of the design and the safety case.

3.6.4.2 Description of Critical Experiments Task

A series of reactor physics tests and measurements will be performed in a geometry that is as prototypic of the ANS as possible. A full prototype core will be fabricated at Babcock and Wilcox, under WBS 1.1.2, for use in these tests. In addition, prototype

mechanical poison systems and detector systems will be used as part of the test geometry. Approximately 6 months of tests are expected, and an additional 6 months will be set aside for potential followup tests. Although the experiment plan has not been completed, five categories of tests are expected:

1. Fuel loading and startup—includes tests covering areas such as subcriticality monitoring, reactivity measurements during refueling, and simulation of startup;

2. Reactor criticality measurements—includes criticality measurements at beginning-of-life and possibly simulated end-of-life configurations;

3. Tests for direct validation of calculational methods—includes identification of critical rod configuration at beginning of life, reactivity coefficients, relative fission density distribution, neutron spectrum, light water contamination, and kinetics parameters;

4. Reactor performance measurements—includes neutron and gamma ray fluxes at various experimental facility locations per fission in the core; and

5. Reactor operation experiments—includes measurement of control and safety rod worths, evaluation of prototype detectors, and examination of the impact of various perturbations on reactor control.

3.6.5 Analysis of Critical Experiments

3.6.5.1 Justification of Analysis of Critical Experiments Task

Analysis of critical experiments provides the evaluation and interpretation of the data obtained from the critical experiment measurements. Thus this task determines the implications of the data and generates feedback to the various project disciplines. Without this task, much of the effort put into the critical experiments program would be wasted.

3.6.5.2 Descriptions of Analysis of Critical Experiments Task

The measured data obtained from the various tests in the critical experiment plan will be compared with the preanalysis data and with data obtained from similar tests in heavy water high flux reactors. Evaluations will be performed to resolve any significant discrepancies between measured data and preanalysis evaluations. If the evidence indicates that the measured data may be in error, tests may be identified to provide additional information. If the preanalysis is considered to be at fault, models will be examined for errors and additional calculations will be performed as deemed necessary. The product of this task will be a report that documents the expected biases and uncertainties associated with specific reactor physics calculations for the ANS. The implications of the results of the comparisons between the calculated and measured data will be examined and discussed with the various ANS Project disciplines as appropriate.

3.7 MATERIAL DATA, STRUCTURAL TESTS, AND ANALYSIS—WBS 1.1.7

Tests and analyses are needed to ensure the structural adequacy of the ANS, including its fuel plates, control elements, CPBT, and other important components. ASME code cases will be developed under the material data, structural tasks, and analysis task, including any evaluations necessary to support the code case. Irradiation damage, material limitations, and vibration also will be evaluated under this activity.

This WBS element contains three major project milestones:

1. Complete the materials properties data base by June 1995. This provides a detailed materials data base for materials expected to be used in the ANS in time for designers to begin the Title II design effort.

2. Complete CPBT fracture tests by June 1996. These tests provide important information on the expected performance of the CPBT in time to have an impact on the Title II design of the component.

3. Complete component vibration tests by September 1997. These tests are performed late in the R&D program, when designs are complete enough to test. The tests are intended to be confirmation of analytical models, but they still will occur early enough in Title II to provide input that can be used to fine tune the design.

The material data, structural tests, and analysis activity is divided into the eight WBS level four tasks summarized in Table 3.7. Most of the work under this activity would be performed at ORNL with support from various subcontractors. The total estimated costs for this activity over the 9-year period covered by this R&D plan are given in Fig. 3.13, and the associated schedules are shown in Fig. 3.14. Most of the work is performed with expense money. However, some capital equipment money is used in this task to construct test facilities, and line-item money is used when the work is in direct support of the Title I or Title II design activities. Subsections 3.7.1 through 3.7.8 provide more detailed information on the WBS level four tasks under this activity.

3.7.1 Structural Design Support

3.7.1.1 Justification for Structural Design Support

Detailed structural evaluations are required to demonstrate that the various components will not fail under any conceivable condition. The nature of certain components will require the development of special models for structural evaluation. In addition, certain components require a reliable basis for setting the maximum defect size to provide a basis for fracture mechanics analyses.

3.7.1.2 Description of the Structural Design Support Task

The thrust of this task is to provide expert assistance, and the latest knowledge in the field, to the designers of those reactor components that require structural analyses. Input for the "design report" will be provided for "code-stamped" components. Assistance

Table 3.7. Summary description of the material data, structural tests, and analyses work breakdown structure level four tasks

WBS	Task description
1.1.7.1	Design support—This task provides support to the design as needed to evaluate the structural design behavior of various reactor components.
1.1.7.2	Fuel plate stability tests and analysis—This task addresses the hydraulic stability of the involute fuel plates under high coolant flow rates. Analytical methods development and application are included. Also included are tests of epoxy plates to validate the analytical methods, as well as tests of aluminum plates and complete dummy fuel elements to verify that the design will meet ANS requirements.
1.1.7.3	Fuel plate deflection due to thermal expansion—This task is to determine fuel plate deflections needed to support the thermal-hydraulics analysis effort. Thermal tests of fuel plates are planned to benchmark the analytical methods.
1.1.7.4	Vibration tests and analyses—This task addresses vibration concerns. Components for which vibrations are identified as a concern include the fuel elements, inner control rods, outer shutdown rods, and the core pressure boundary tube. Experimental studies are necessary for the fuel elements and the inner control rods.
1.1.7.5	Material properties—This task will provide data on the effects of the extremely high neutron flux on the materials used in constructing ANS components. A major thrust is determining irradiation effects on the fracture toughness of 6061-T6 aluminum. Tests in HFIR will be supplemented by the development of a surveillance test program for the ANS.
1.1.7.6	ASME code case for 6061-T6 aluminum—This task is to gain ASME Boiler and Pressure Vessel Code Committee acceptance of 6061-T6 aluminum for the construction of Class I nuclear vessels.

Table 3.7. (continued)

1.1.7.7	Materials properties data base—This task will provide the project with a centralized accessible source of materials properties data qualified according to NQA-1.
1.1.7.8	Fracture assessment—This task provides a sound basis for fracture assessment. It includes the preparation of a deterministic methodology for preventing nonductile rupture in unirradiated and irradiated 6061-T6 aluminum. Adapting the probabilistic fracture methodology used for ferritic components to 6061-T6 aluminum also is included as a way of safely extending the operating life of ANS components. Fracture experiments are included to verify the margins promised by the methodologies.

also will be provided in the preparation of design specifications and review of design reports required by the ASME Boiler and Pressure Vessel Code.

Fracture-mechanics evaluations of critical components will be required. The development of the fracture mechanics methodology for 6061-T6 aluminum is the purpose of Task 1.1.7.8. However, one important ingredient of any fracture assessment methodology is characterization of the flaw distribution; that will be done under this task. Quantitative information about the size distribution of potential defects in aluminum components will be developed. The smallest defect size that can be detected with high certainty by nondestructive examination techniques will be determined. Nondestructive examination techniques and procedures will be selected to ensure that components placed in service will not have defects larger than some carefully determined size. That defect size, with appropriate safety margin, will then be used in fracture evaluations.

Because 6061-T651 aluminum has a relatively low fracture toughness that is degraded by neutron irradiation, special nondestructive examination methods will be required to ensure that components placed in service have a very small maximum defect size. Studies conducted under the National Aeronautics and Space Administration space shuttle program[4] show that it is currently possible to detect surface flaws reliably that are 0.4 mm (0.015 in.) deep and 2.3 mm (0.090 in.) long. The ANS Project will demonstrate such techniques and apply them to the reactor vessel (CPBT).

3.7.2 Fuel Plate Stability Tests and Analysis

3.7.2.1 Justification for the Fuel Plate Stability Tests and Analysis Task

Closely spaced arrays of fuel plates cooled by water flowing through the channels between the plates have long been used in research reactors. Since the work of Stromquist and Sisman in 1948[5], it has been known that very high flow velocities past fuel plates can cause the plates to deform, buckle, and collapse. Excessive fuel plate deformation can impede coolant flow and heat removal and thus must be avoided in the reactor design. Obviously, the fuel plates must not deflect to such an extent that they will impede the

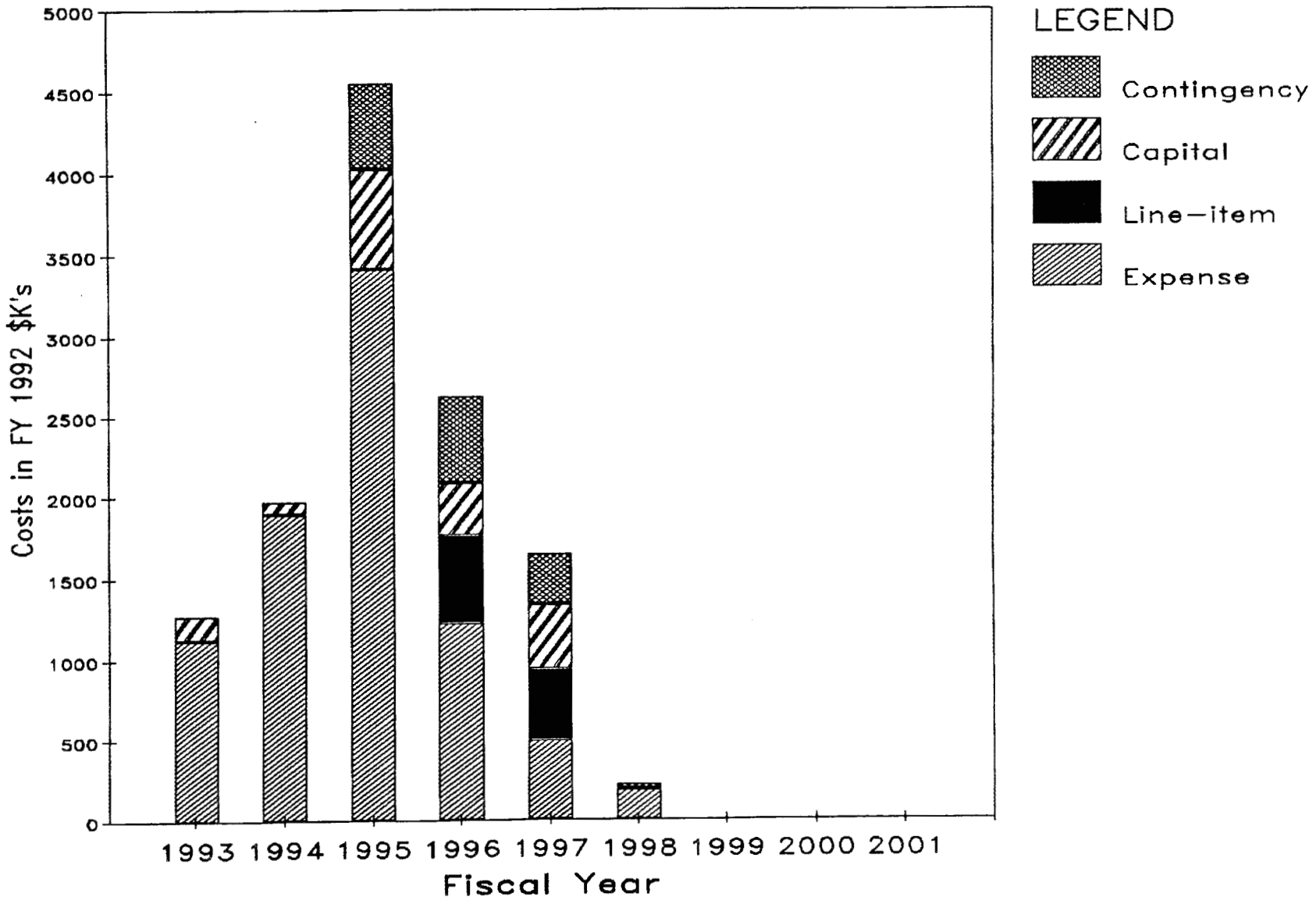

Fig. 3.13. Summary of cost estimates for work breakdown structure 1.1.7, material data, structural tests, and analysis.

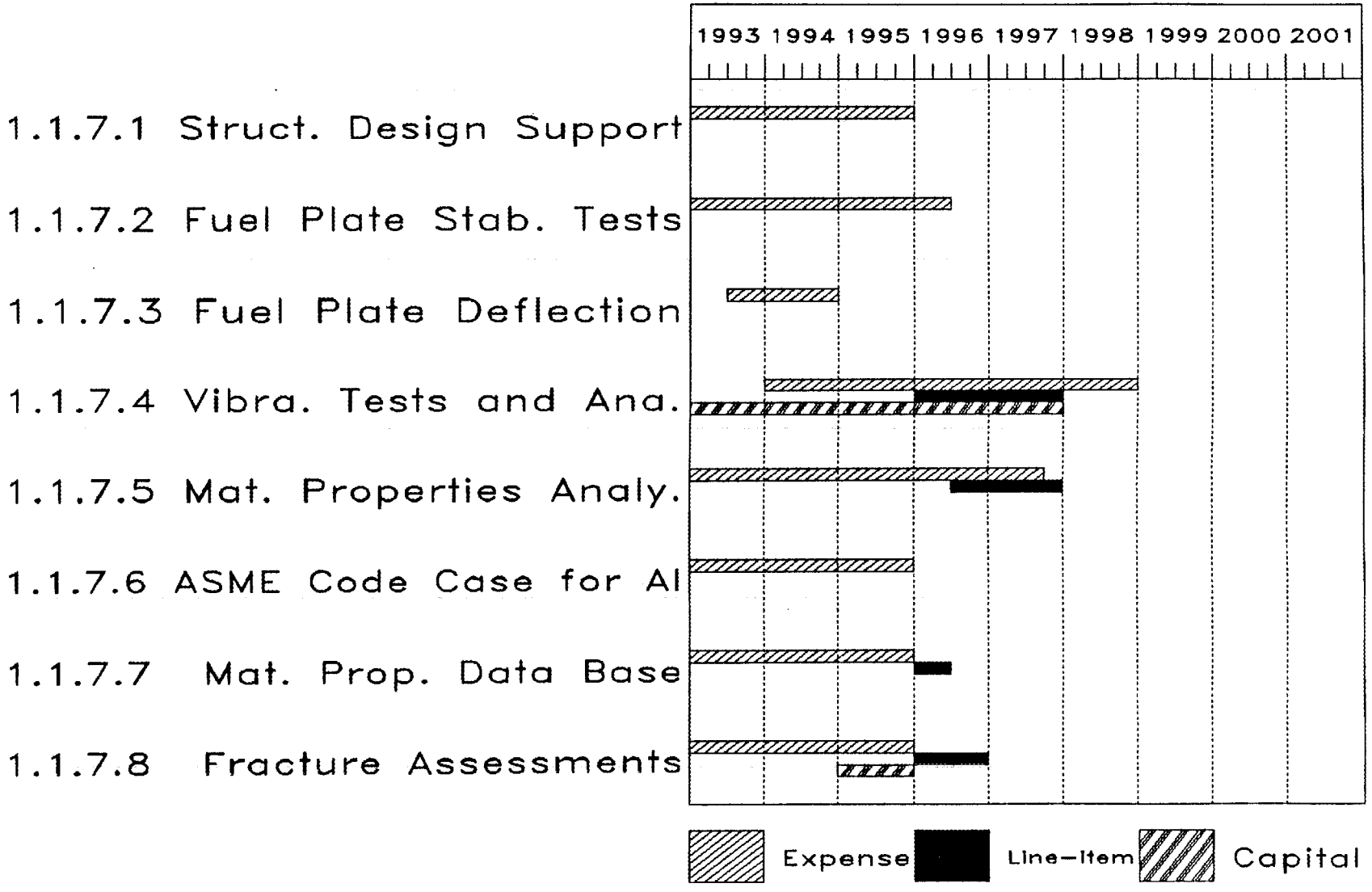

Fig. 3.14. Schedule for work breakdown structure 1.1.7, material data, structural tests, and analysis.

flow of coolant through the reactor core. Therefore, fuel plate stability tests and analytical model analysis must be performed to ensure that adequate margins exist between the coolant flow velocity and plate collapse velocity. Without this task, plate stability could not be ensured for any substantial variation from the HFIR flow and geometry conditions, which would greatly limit the performance of the ANS reactor core.

3.7.2.2 Description of the Fuel Plate Stability Tests and Analysis Task

To assess the allowable coolant flow velocity, a linearized involute shell model of the fuel plates has been coupled with a linearized hydraulic equation incorporating fluid friction.[6] Inertial and damping terms (time derivatives) are included in the fluid and plate equations, although computer cost limitations prevent the full use of the inertial terms. Incorporation of inertial terms and complete inlet and outlet boundary conditions allows the calculation of the normal modes of vibration of the coupled fluid-plate system under flow conditions. These vibrational modes and their associated frequencies and damping coefficients are of interest in the ANS design in their own right, in addition to their involvement in the Miller instability phenomenon.[7]

For nonisothermal plate stability analysis, ABAQUS[8] is used in combination with a locally developed "user element" to predict the behavior of the fluid in the coolant channel and to provide two-way coupling between the fuel plates and the coolant fluid. Design changes that are expected to strengthen the heated plates are being investigated.

The current hydraulic buckling analysis capability is limited to a linearized, infinitesimal perturbation about a large, nonlinear thermal deflection. Modifications will be made to the ABAQUS fluid user element that might allow calculations of fully nonlinear hydraulic buckling of fuel plates in the plastic regime.

The capability to perform hydraulic creep buckling calculations for the fuel plates has not been reached yet, but plans to develop that capability are under way. Additional measurement of the elevated-temperature creep and plastic behavior of the fuel plate materials might be required for further development of analysis capabilities.

In the past, tests conducted on arrays of flat plates demonstrated instability at high velocities, but such tests had never been done on arrays of involute plates. Benchmark tests of arrays of aluminum involute plates and proof tests of complete dummy fuel elements are planned. However, the flow rates and pressures required to reach plate instability are fairly large for aluminum plates. The lower modulus of elasticity of plastic (epoxy) plates, compared with aluminum, reduces the critical velocity so that modest flow rates and pressure can be used if epoxy is used to model the aluminum plates.

A single involute plate made from epoxy with flow channels formed by rigid walls has been tested. Because of facility limitations, the maximum velocity that could be achieved in the test was 121% of the critical velocity predicted using the Miller analysis[7], but it was only 95% of the critical velocity according to the Sartory[6] analysis. No instability was observed during the test.

Tests also have been conducted on arrays of five plates for both the upper and lower fuel elements to examine their structural response to coolant flow. The tests were conducted on full-scale epoxy models of the fuel plates, which allowed for reduced test pressures and flow velocities. The results have been related to the prototype aluminum clad aluminum/uranium silicide plates through model theory. Additional tests on epoxy plates are planned to validate the analyses further. The tests will be followed by tests on dummy aluminum plates to monitor the inertial response of the plates.

As a final confirmation, a stability test for a full dummy prototype core is planned as a proof test for the interaction of all core components, for the adequacy in fabrication of each component, and for the assembly of the core. The flow rate, inlet pressure, exit pressure of adjacent channels, and strain gage response of plates are to be monitored. A stainless steel flow loop with a maximum flow rate of 1250 L/s (20,000 gpm) at 2.4 MPa (350 psi) is proposed for these tests; testing is scheduled to begin in FY 1994.

3.7.3 Fuel Plate Deflection Due to Thermal Expansion

3.7.3.1 Justification for the Fuel Plate Deflection Due to Thermal Expansion

Mismatch in the thermal expansion of the fuel plates and the support cylinders (side plates) in the plane normal to the reactor axis are accommodated by the involute plate design. There is no such mechanism for accommodating the axial differential expansion. The ANS involute fuel plates will be distorted by the thermal expansion mismatch. If the deformation excessively reduces the thickness of the coolant gap between two adjacent plates, heat removal may be impeded, and the fuel plates may overheat and be damaged. Therefore, the changes in gap must be calculated and provided as input to the thermal hydraulic analyses. In addition, validation of analytical models must be obtained through analysis of benchmark tests.

3.7.3.2 Description of the Fuel Plate Deflection Due to Thermal Expansion Task

For structural analysis, ABAQUS[8] and ADINA,[9] two widely used and highly respected general-purpose finite element structural analysis computer programs, are being used to estimate the maximum change in gap between plates due to temperature variations alone. Temperature distributions in a hot and a cold ANS fuel plate will be used to determine maximum deflection due to thermal expansion. Displacements then will be imposed on the hot fuel plates and the cold fuel plates, and analyses performed to determine the change in gap between plates. The results will be input to the thermal-hydraulic analyses to make better temperature calculations for another iteration.

The deflection of the fuel plates due to temperature differences between them and the support cylinders must be determined experimentally to benchmark predictions. Two dummy fuel plates are to be mounted symmetrically in support cylinders and strain-gaged to monitor the thermal response of the plates to temperature differences in the support cylinders. One of the support cylinders will be required to float relative to the other depending on the final design. Thermal control of the support cylinders is also required.

3.7.4 Vibration Tests and Analyses

3.7.4.1 Justification for Vibration Tests and Analyses Task

Turbulent axial and cross flow around the inner control elements, or any other flexible component, can result in vibration amplitudes that cause concern about the safe operation of the reactor. The cross flow has been minimized by the inherent design, but several components will be exposed to turbulent axial flow. The inner control elements are the most flexible components because of their long unsupported spans and their small diameter. Large vibrations of the control elements could cause contact between these

components and with the core inner cylinder. This contact would result in wear and could result in a failure to scram or put impurities in the coolant flow stream. Large vibrations also could cause oscillations in the response of the nuclear monitoring devices, which could cause difficulty in the control system or could result in false scram signals. The criteria used for the design thus far are that the vibrations will be minimized and that no contact can be allowed between any of the elements. These criteria are also being applied to the CPBT, which has the potential, though less potential, to have flow-induced vibrations (FIV) large enough to be of concern. The requirement to keep sections thin to limit the amount of nuclear heat deposited in them makes vibrations more likely. Components of particular concern are the fuel elements, the primary control rods, the reflector safety rods, and the core pressure tube. Without this task, the extent of vibration of important components and its consequences would not be known.

3.7.4.2 Description of the Vibration Tests and Analyses Task

The two analysis methods for FIV are the empirical and the analytical modeling methods. The empirical model used thus far appears to be giving reasonable results. The empirical model with refinements in the inputs is a reasonable way to proceed with the design. The amount of conservatism in the empirical equation is unknown at this time and will be studied further to see how applicable the empirical data base is to the ANS configuration. The analytical model approach to the estimate of FIV is straightforward in that random structural response calculations are performed for a system of coupled beams. The major problem with the analytical model approach to FIV is that the vibratory component of the pressure that produces the forcing function is generally unknown. Some attempts have been made to synthesize the forcing function, but without much success so far. The best approach to the forcing function development would be a series of tests on rigid segments of the control rods contained in the inner core cylinder to measure the actual net random pressure fields across the control elements. The segments would be chosen on the basis of unique flow characteristics, and testing of the entire control element would not be required. However, measuring the pressure difference across the control elements with adequate frequency response in the instruments is not a trivial task. It is not in the current budget estimates because the full scale flow test to measure response directly is a more desirable approach.

Consequently, a series of simpler and less expensive analyses are planned to refine the current FIV estimates. The refinements that should be made first are the added hydroelastic coupling that exists in our configuration as a result of the close spacing and the tight confinement of the inner core cylinder. An estimate of this added inertial loading will be made for inclusion in our modal calculations. Alternate supports at the top of the control elements will be studied because we have eliminated the upper guide tubes in the current design: That is, the control element modes will be studied for a set of rollers fixed to the inner core cylinder to support the upper end of each control element. Additionally, the structural coupling due to the lower guide tube connections will be included. This task will need to be done with the NASTRAN model because the current approximate models are limited to simple boundary conditions and are not programmed for spring connections.

Finally, 3-D models of the control elements and the support structure will be developed to model the fluid coupling and added effective mass of the fluid. Eigenvalues and eigenvectors will be computed for this 3-D model. Direct correlation of these results will be made with measured values from the Control Element Test Facility system modal

test. Correlation of these analysis and test data will verify that the control element system dynamics are thoroughly understood and that the test hardware is configured and mounted in the manner intended. That is, the modal test will provide an acceptance test for the hardware and at the same time provide diagnostic information needed to interpret the results of the forced flow tests. Thus, the modal analysis and modal test will not predict directly the response of the structure, but will provide valuable design guidance and basic knowledge of the control element structure system. This knowledge will be used to interpret and resolve any discrepancies that appear in the full scale flow test data and will provide guidance for design changes to eliminate any FIV that might be encountered in the Control Element Test Facility forced flow test.

Three test facilities are expected to be needed to complete this task. Those test facilities are described in the following paragraphs.

Core element dynamic test facility—In evaluating the expected performance of the fuel plates, it is necessary to compare the plate's natural frequencies with potential exciting frequencies of the reactor system. If these frequencies match, then some modifications will be needed to avoid possible plate failures. To determine the natural frequencies of the fuel plates, a plate (or plates) with fixed boundaries is to be excited with variable frequency sounds and the natural frequencies recorded with time-average holography. A holographic lab with a real-time and a time-average system, in addition to a variable frequency audio oscillator for exciting the resonance, is required.

Control element test facility—The major purpose of this facility, as previously discussed, is to determine whether there will be unacceptable FIV of the control elements. The facility also will be used to establish assembly/disassembly procedures, determine operational characteristics, and determine operational reliability. The facility will include a flow loop that will enable testing of a full set of three control rod assemblies at reactor flow and pressure conditions. An assembly test stand will also be built so that functional and assembly checkout can be performed parallel with the flow test stand checkout and operation. The assembly test stand likely will be an inner core cylinder filled with static fluid and containing one or more complete control elements, including the drive systems. Assembly and disassembly checkout procedures will be developed on the flow test stand as well as the assembly test stand. Scram testing will be accomplished in both the assembly and forced-flow test stands. The assembly stand will be modified to perform the scram test under external heating, which simulates the neutron and gamma heating present in the reactor. This heating external to the inner core cylinder will induce a nonsymmetric thermal deformation that may cause bowing in the control elements. If excessive, the bowing could cause a failure to scram properly, depending on several factors such as manufacturing tolerances of bearings. This heated scram test will verify that the thermal deformations will not degrade or prevent normal scram performance. The required flow rate and pressure of the flow loop cannot be determined until the cooling requirements for the control elements in the reactor are established. The conceptual design phase has begun, and testing is scheduled in FY 1994.

Outer shutdown rod dynamic test facility—The purpose of this facility is to determine whether there will be unacceptable FIV of the shutdown rods. A single shutdown rod will be tested in a flow loop at reflector vessel flow and pressure conditions. A flow loop will be required, and operating conditions will be contingent on the design.

Testing is presently scheduled for FY 1994, but no planning of the actual tests has been performed at this time.

3.7.5 Material Properties Evaluations

3.7.5.1 Justification for the Material Properties Evaluations Task

The extremely high neutron flux in ANS means that certain components will be subjected to higher total fluences than previous reactors have been if reasonable component lifetimes are ensured. It is important to know the effect of this high fluence (including rate effects) on the various materials used in the ANS reactor. Embrittlement is an especially important characteristic that must be understood. Without material properties evaluations, the lifetime limits of various components can not be established.

3.7.5.2 Description of the Material Properties Evaluations Task

A test program is already under way for determining the effect of irradiation on the mechanical properties of structural materials for the ANS Project. The initial focus is the study of the effect of irradiation on 6061-T651 aluminum alloy, which has been selected for the CPBT, the reflector vessel, and possibly the cold source as well. Six capsules will be irradiated in the HFIR to study the response of base metal, weld metal, and heat-affected zone metal. Individual capsules will be irradiated to neutron fluence levels from 10^{26} to 10^{27} m^{-2}. This maximum fluence represents ~ 6 months of operation for the CPBT and ~ 30 years for the reflector vessel. Four capsules will contain material typical of the CPBT and will be irradiated at a thermal-to-fast neutron ratio of ~ 2, close to the predicted ratio for the CPBT. The other two capsules will be irradiated with as high a ratio of thermal-to-fast neutrons as possible. These capsules are intended to provide data for the reflector vessel, which will see a thermal-to-fast ratio of $\sim 10^7$. That ratio vastly exceeds the ratio at any existing irradiation facility. Existing information on the effect of irradiation on the mechanical properties of aluminum alloys suggests that the thermal-to-fast neutron ratio may be very significant, with higher ratios resulting in increased degradation of properties.

Some irradiation tests also are planned on the control material and the material used for springs in the control rod scram mechanism. Some testing also will be done on unirradiated materials to establish baseline properties.

An extensive materials surveillance program will be planned and conducted to assess the neutron-induced changes in the mechanical properties of the reactor vessel materials. Changes in properties, most importantly fracture toughness, will be evaluated by comparing these properties for irradiated and unirradiated materials. The capsules containing the test specimens will be irradiated under conditions most closely representing the radiation environment of the reactor vessels (outer and lower inner CPBTs and reflector vessel).

Test specification ASTM E-185-82, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," and 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," present criteria for monitoring changes in the fracture toughness properties of reactor vessels through surveillance programs. The ANS reactor vessel surveillance program will adhere to the requirements of ASTM E-18582 and will satisfy the intent of 10 CFR 50 Appendix H. Exceptions may be

necessary when these procedures and requirements are applied to nonferrous materials and/or when reasons make them unfeasible for an experimental research reactor.

The reactor vessel (CPBT) will be replaced at intervals so that the fracture toughness of the vessel does not fall below that required by the analysis conforming to Appendix G of the ASME Code Section III. The replacement interval will be determined from data obtained by measuring the fracture toughness reduction of the surveillance specimens.

3.7.6 ASME Code Case for 6061-T6 Aluminum

3.7.6.1 Justification for the ASME Code Case for the 6061-T6 Aluminum Task

DOE requires that, where applicable, the ASME Boiler and Pressure Vessel Code requirements be met. Section III of the ASME Code indicates those materials that already have been approved for use as nuclear pressure vessels. However, a material is needed for the ANS that has a very low absorption cross section. A review of Sect. III indicated that none of the materials included had a low enough absorption cross section to meet the criteria. Therefore, approval of an appropriate material, 6061-T6 aluminum, must be obtained. Without this task, the concept of an inner pressure boundary between the reactor core and the beam tubes could be used only if an argument were made for the equivalence of the ANS Project approach to the Sect. III requirements.

3.7.6.2 Description of the ASME Code Case for 6061-T6 Aluminum Task

After careful consideration of candidate materials, 6061-T6 aluminum was selected as the reference CPBT structural material. Although it has been used in previous research reactors, 6061-T6 aluminum has not been included in the ASME *Boiler and Pressure Vessel Code* for Class 1 nuclear construction. The project has formally requested the *ASME Boiler and Pressure Vessel Code* Committee to include 6061-T6 aluminum as an acceptable material for Class 1 components. That request must be considered and approved by a series of *Code* committees up through the main committee of the *ASME Boiler and Pressure Vessel Code* Committee.

The principal activities under this task are to ensure that the appropriate Code committees consider the use of 6061-T6 aluminum and to revise the proposed Code Case in response to the committee comments.

Two major areas that require development for the Code Case are (1) fatigue design curves and (2) rules for ensuring that nonductile fracture will not occur. Fatigue design curves now have been developed and approved by the appropriate Code committee. Additional work must be done to obtain Code committee approval of rules to ensure that nonductile fracture will not occur.

3.7.7 Materials Properties Data Base

3.7.7.1 Justification for the Materials Properties Data Base Task

The design team will need to address material limitations frequently. Therefore, a readily accessible centralized materials data base that covers the materials used in the ANS design will avoid delays in obtaining information and will ensure that all project staff use the same properties data.

3.7.7.2 Description of the Materials Properties Data Base Task

It is important that a reliable and consistent set of material properties be used by all of the ANS Project personnel. A data base format will be selected and the data required by the various ANS personnel will be input into the data base. Reviews of the data entered into the data base will be necessary to ensure that the data are accurate. A means of accessing the data base will be developed and made available to project staff. A procedure for updating the data base and informing staff of updates also will be developed and implemented under this task.

3.7.8 Fracture Assessment

3.7.8.1 Justification for the Fracture Assessment Task

This task will validate the performance of the CPBT. This work is required to provide an adequate basis for ensuring that catastrophic failure of the pressure boundary will not occur under any anticipated operating conditions.

3.7.8.2 Description of the Fracture Assessment Task

This task is to provide a sound basis for fracture assessment. It includes the preparation of a deterministic methodology for prevention of nonductile rupture in unirradiated and irradiated 6061-T6 aluminum. Adaptation of the probabilistic fracture methodology used for ferritic components to 6061-T6 aluminum also is included as a way of establishing a risk of failure. Fracture experiments are included to verify the margins promised by the methodologies.

3.8 COLD SOURCE DEVELOPMENT—WBS 1.1.8

One of the major facilities of the ANS is a source of low-energy neutrons for neutron beam experiments. These low-energy neutrons are obtained by placing a moderator at very low temperature in the reflector region of the ANS reactor. The cold source development task provides the analysis and testing program necessary to develop an ANS cold source concept and demonstrate its performance capabilities.

This WBS element contains one major project milestone:

Complete cold source prototype tests by the end of January 1998. This task provides input data to the design at least 1½ years before the end of the Title II design effort.

The cold source development activity is divided into the seven WBS level four tasks that are summarized in Table 3.8. Most of this work would be performed at ORNL, but some of it would be performed at INEL, the University of Virginia, and other subcontractors. The total estimated costs for this activity over the 9-year period covered by this R&D plan are given in Fig. 3.15, and the associated schedules are shown in Fig. 3.16. The initial development work would be performed using expense money. The more detailed design support efforts and the prototype tests would use line-item money. Capital equipment money would be used to construct test facilities during the development phase. Subsections 3.8.1 through 3.8.7 provide more detailed information on the WBS level four tasks under this activity.

3.8.1 Neutronics Analysis

3.8.1.1 Justification for the Neutronics Analysis Task

The cold neutron current entering the guide tubes depends upon the thermal flux entering the cold moderator, the cold moderator material properties, the geometry of the cold moderator and guide tubes, and the characteristics of materials between the cold moderator and the guide tubes. Cold neutron physics analyses are necessary to optimize features to meet or exceed to the best extent practical the performance requirements for the cold source. Without this task, the ability to meet performance requirements cannot be determined.

3.8.1.2 Description of the Neutronics Analysis Task

This task includes all activities necessary to optimize, within defined constraints, the cold neutron current entering the guide tubes. Subtasks will include identification of appropriate cold neutron cross sections and analysis methods, evaluations of moderator options, examination of the importance of geometry, and studies of the impact of structural material location and type. In addition, detailed evaluations of neutron and gamma heat loads will be performed to provide heat sources to the thermal-hydraulic analyses. Validation of analysis techniques will be obtained by modeling the ILL vertical cold source and comparing calculated parameters with those measured at the ILL.

Once a design concept has been agreed upon for the moderator vessel, more detailed physics calculations will be required to determine the overall thermal loads of the cold source system because the neutron and gamma heat loads are dominant in the sizing of the refrigerator. In addition, a greater understanding of radiation damage to the materials of construction will be required to plan replacement programs; thus, comprehensive mapping of gamma and neutron flux levels will be needed.

3.8.2 Thermal Analysis

3.8.2.1 Justification for the Thermal Analysis Task

Thermal-hydraulic analyses are necessary to develop a conceptual cold source design. The models developed under this task play an important role in understanding the thermal-hydraulic behavior of the system under normal steady state as well as transient

Table 3.8. Summary description of the cold source development work breakdown structure level four tasks

WBS	Task description
1.1.8.1	Neutronics analysis—This subtask will provide model development data such as cold cross sections and kerma factors necessary to analyze and optimize the cold neutron physics. Gamma and neutron heating at various parts of the system will be evaluated. Models of the ILL reactor and cold source will be used to compare calculations with measured data, providing a validation of analysis techniques.
1.1.8.2	Thermal analysis—This subtask will include modeling to simulate the thermal and hydraulic behavior of the cold source system both under stable conditions of single-phase operation and under conditions of two-phase operation such as cooldown or component failure. It will also include an analysis of the heat exchanger and deuterium pump performance.
1.1.8.3	Stress analysis and structural design—This subtask will examine under all anticipated conditions the integrity of the pressure vessels that make up the system.
1.1.8.4	Cryogenic model verification tests—This subtask was initiated to provide two-phase flow simulation in a boiling thermosyphon cold source model. Its initial emphasis was on examining bubble formation and vapor fraction as a function of heat load. With the rescoping to a pumped subcooled liquid system, emphasis has changed to examining heat transfer coefficients to the inside walls of the spherical vessel and the remaining two-phase operations such as cooldown or component failure.
1.1.8.5	Cold source prototype development and demonstration—This subtask covers development of the deuterium pumping system, development of the main loop components and refrigerator, and finally, construction of a complete prototype test facility.
1.1.8.6	Cold source materials selection—Cryogenic materials testing and irradiation evaluation will be covered by R&D task 1.1.7, but this subtask will cover the effort required to provide an interface between the cold source design and development team and the materials evaluation staff.
1.1.8.7	Instrumentation and control—This subtask will cover the design and development of a monitoring and control system for the cold source system, which will include temperature and pressure control together with comprehensive parameter indication throughout the loop. Also, evaluations will be performed to determine interlocks between the cold source control system and the reactor control system.

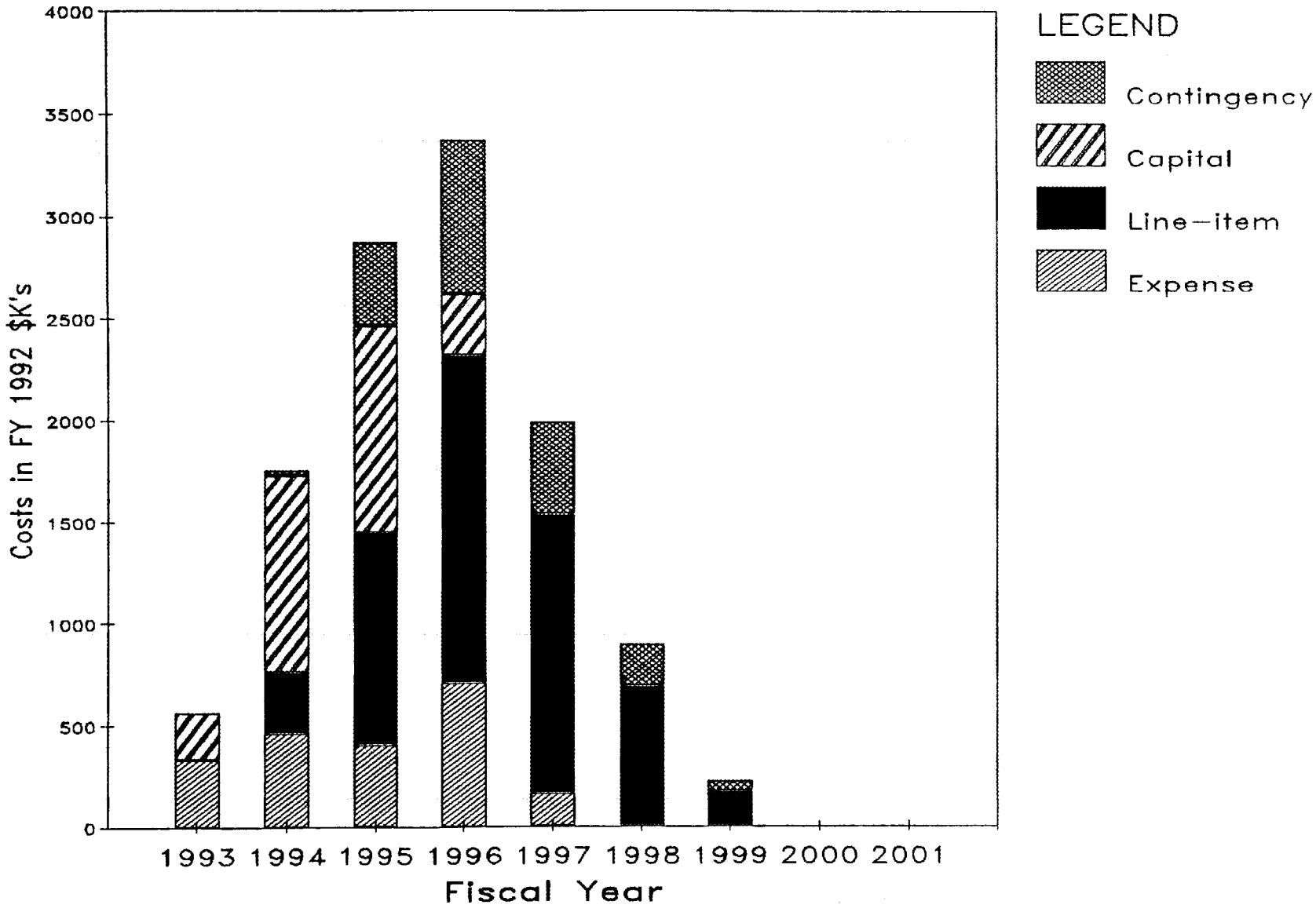

Fig. 3.15. Summary of cost estimates for work breakdown structure 1.1.8, cold source development.

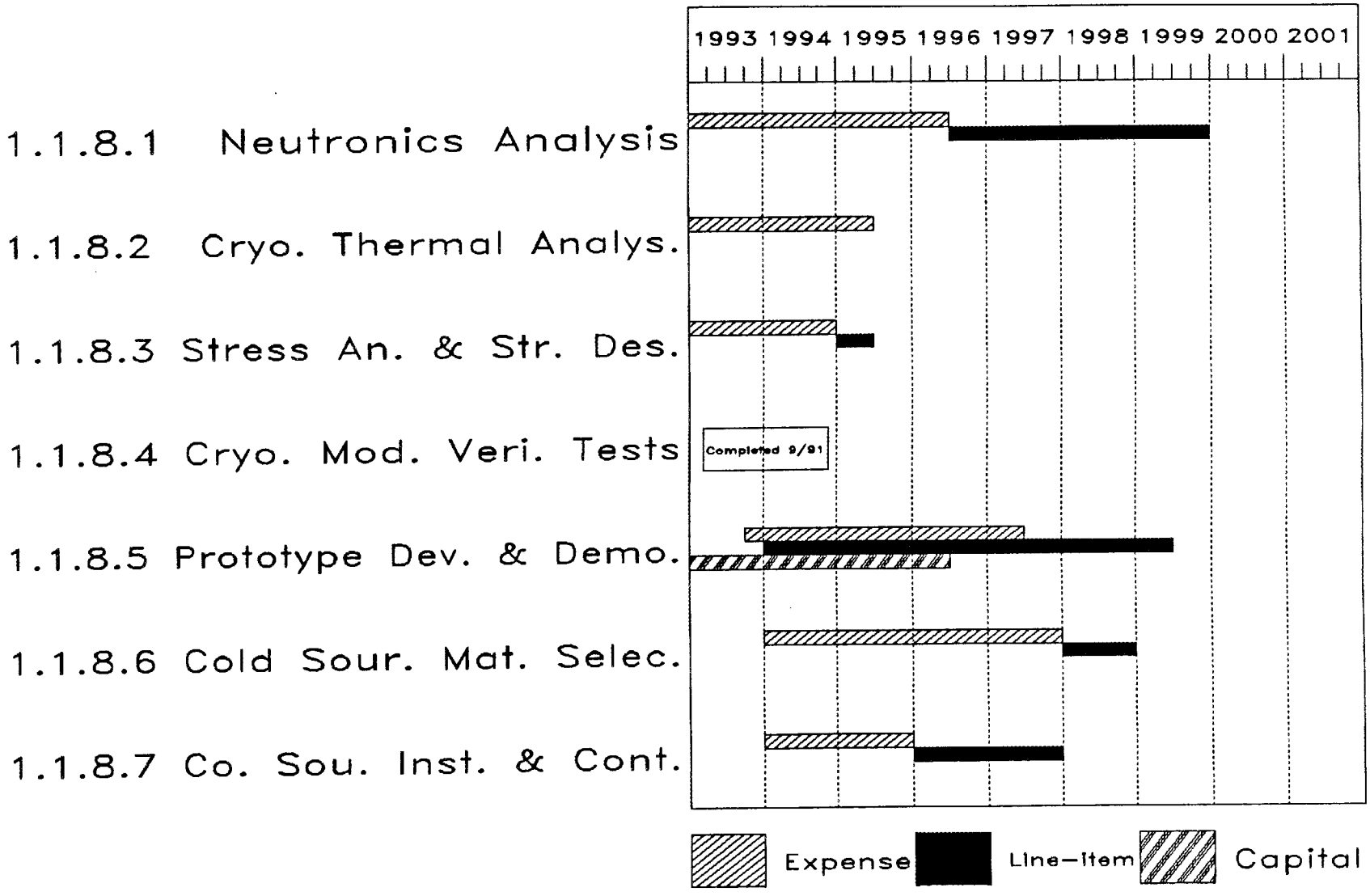

Fig. 3.16. Schedule for work breakdown structure 1.1.8, cold source development.

(e.g., startup or shutdown) conditions. Without this task, stability of the ANS cold source moderator cannot be ensured.

3.8.2.2 Description of the Thermal Analysis Task

Appropriate models will be developed and analyses performed to cover the complete cold source system and to determine the refrigerator, heat exchanger, pump, and other loop component characteristics necessary to maintain a single-phase steady state moderator condition while removing approximately 30 kw of heat. Thermal-hydraulic evaluations of normal nonsteady state (e.g., startup and shutdown) and anticipated off-normal conditions also must be performed to ensure that minimum cooling requirements are met. In addition, models must examine the potential for freezing within the system and the options to avoid it if it is determined to be of significant risk to the system.

Models developed will be benchmarked against data in the literature and ANS tests as necessary. A peer review group external to the project will be supported by this task to provide a periodic examination of the adequacy of the models and analyses.

3.8.3 Stress Analysis and Structural Design

3.8.3.1 Justification for the Stress Analysis and Structural Design Task

Five major factors require that careful thought be given to developing the design of the cold source structures: (1) Structures will be subjected to several types of stresses during operation, including pressure, flow, and thermal loads; (2) structures should be designed for a very low probability of deuterium release; (3) the addition of unnecessary structure adds more heat load and can reduce the number of cold neutrons reaching the guide tubes; (4) it is anticipated that the cold source structures will have to be replaced periodically; and (5) transient conditions during startup and shutdown may introduce important short term stress loads. The stress analysis and structural design task provides the detailed analysis necessary to address these issues. Without this task, a safety case for the cold source could not be made and performance of the cold source system could not be ensured.

3.8.3.2 Description of the Stress Analysis and Structural Design Task

The cold source will incorporate a number of pressure vessels, some of which will be enclosed by a vacuum chamber. A philosophy of double containment will be followed throughout the design development process; that is, if a deuterium vessel should fail, the vacuum chamber structure must contain the fluid or provide safe ventilation from the building.

Evaluations will be performed to determine stress loads and set structure geometry requirements. Although Al-6061 is the reference structural material, other materials also will be examined. A safe working stress value equal to or less than that recommended by the ASME code will be used for all analyses. Some iteration of the results will be required because factors such as heat loads change as the structural geometry changes. Proof tests will be performed as necessary to support the design.

3.8.4 Cryogenic Model Verification Tests

3.8.4.1 Justification for the Cryogenic Model Verification Tests Task

The cryogenic model verification tests task was originally created to examine and test the philosophy of a boiling thermosyphon cold source. Its emphasis was on the examination of bubble formation and liquid fraction as a function of the heat load, and it supplied the data needed to validate the two-phase analytical models. With the switch to a single-phase system, work under this task was greatly reduced and this task is now considered complete. However, the data are of continued value to the project because there will be a transition to two-phase conditions during startup and shutdown operations.

3.8.4.2 Description of the Cryogenic Model Verification Tests Task

The scope of this work was two-fold: (1) providing the data necessary to create a computer code to evaluate heat transfer coefficients for two-phase operation and (2) building and operating a test model to prove the system feasibility and to examine bubble propagation under various heat loads. A test facility with a spherical geometry cryostat was constructed and measurements were taken under various heat load conditions using a liquid nitrogen moderator. Testing was suspended shortly after the decision was made to pursue a single-phase liquid cold source system. This task will be reopened if it is determined that additional tests are needed to provide a better understanding of the transition to and from the two-phase system that occurs during shutdown and startup.

3.8.5 Prototype Development and Demonstration

3.8.5.1 Justification for the Prototype Development and Demonstration Task

The cold source system will be a unique design that should be tested in an environment much simpler than that of the completed ANS facility. These tests will have three major objectives:

1. to demonstrate the operation of the system as a single unit for the first time;

2. to identify weaknesses in the system and either make modifications or develop operational procedures as considered appropriate; and

3. to prove the reliability of the system under all operational conditions.

If performance flaws are identified, these tests will provide the opportunity for design change prior to installation at the ANS. The results also will be used to define a startup test program for the ANS cold source.

3.8.5.2 Description of the Prototype Development and Demonstration Task

Analytical models and tests of individual components will be used to develop the prototype cold source for the ANS. Individual tests anticipated at this time include the following:

1. pumping module performance and reliability testing to be carried out in various locations as required to meet the various safety requirements;

2. heat exchanger testing to demonstrate freedom from flow stratification and its ability to meet the design requirements of both cooldown and steady state conditions;

3. cryogenic testing of the cold source thimble assembly to ensure its reliability and thermal integrity;

4. testing of the transfer lines to ensure they meet specification with regard to thermal efficiency and pressure integrity; and

5. testing of the control system as a part of its development, including routine benchmarking of temperature and pressure sensors and development of new techniques that might be required for monitoring the cold source.

When the design of the cold source has been completed, a demonstration prototype will be constructed. A complete program to replicate as many operational conditions as practical will be prepared. Tests will cover long term operational running, cooldown operations both from an overall ambient temperature state and from a partially cold state, and simulated fault conditions such as vacuum failure, refrigerator failure or the failure of a circulation pump. Results of the prototype test will be analyzed and used to define the expected operating behavior of the ANS cold source. Startup test procedures for the ANS cold source will be prepared based on the results of these prototype tests.

3.8.6 Cold Source Material Selection

3.8.6.1 Justification for Cold Source Material Selection Task

Many of the cold source components will operate at very low temperatures and will be located in a high neutron flux field. Under these conditions, embrittlement of the material becomes of major concern and the issue of component lifetime becomes a very important question (e.g., if the cryostat has to be replaced every 3 months, it is an unworkable design). An effort is needed to examine candidate structural materials, identify reference material(s) for use in the cold source structures, and establish expected lifetimes of components. Most of this work will be performed under WBS 1.1.7, but the nature of the cold source components requires a close interface between the material evaluations and the development of the facility. The cold source material selection task was created to provide the appropriate interface.

3.8.6.2 Description of the Cold Source Materials Selection Task

Material selection for major components of the cold source will be made from normal cryogenic materials such as austenitic stainless steels and magnesium, Zircaloys, or aluminum alloys. Final selection of a material for the cold source vessel will be made only after a comprehensive study of candidate materials. Although this study will be performed under WBS 1.1.7, this task will support work that ensures that all aspects (structural, neutron physics, and thermal requirements) of material selection are considered.

A wire-wrapped sphere has been proposed as an alternative to a normal vessel. Although the qualification of a cryostat structure of this type would probably require more extensive R&D than the qualification of a normal vessel, its potential advantages merit further investigation. A study of the technique will be performed within this task in conjunction with normal vessel material selection and design. A change in reference vessel concepts could be made at a later date if the wire wrap approach is demonstrated as viable.

3.8.7 Instrumentation and Control

3.8.7.1 Justification for the Instrumentation and Control Task

Control of the cold source system must be maintained within set point bands with respect to both pressure and temperature. The instrumentation and control task is required to identify the operating constraints and to develop a control concept that ensures that these constraints are not violated. Without this task, the stability of the cold source system could not be demonstrated.

3.8.7.2 Description of the Instrumentation and Control Task

Evaluations will be performed to determine key operating parameters that must be controlled and the accuracy with which they must be maintained. Analyses then will be performed to determine the number, types, and locations of sensors necessary to allow constant surveillance of the important cold source parameters during operation. Methods of controlling parameters based on sensor readings will be established and analyzed. In addition, special control procedures for startup and shutdown will be identified and validated in the prototype tests (WBS 1.8.5). This task also will provide an input to the ANS cold source startup testing plan.

3.9 BEAM TUBE, GUIDE, AND INSTRUMENT DEVELOPMENT—WBS 1.1.9

Beam tube, guide, and instrument development includes the design, test, evaluation, and possible redesign of the candidate equipment and systems needed for neutron delivery to beam experiments and for performance of those experiments. The basic goal is to maximize the use of neutrons delivered by the reactor so that each improvement developed under this task is equivalent to increasing the flux of the reactor.

This WBS element contains one major project milestone:

> Complete beam tube assembly prototype tests by the end of March 1997. This task provides actual performance data to the designers at around the middle of the Title II design phase and allows some time for considering changes if design flaws are identified in the tests.

The beam tube, guide, and instrument development activity is divided into five WBS level four tasks that are summarized in Table 3.9. Most of this work would be performed at Brookhaven National Laboratory, the National Institute of Standards and Technology, ORNL, and other subcontractors, including several universities. The total estimated costs for this activity over the 9-year period covered by this R&D plan are given in Fig. 3.17, and the associated schedules are shown in Fig. 3.18. The initial development work would be performed using expense money, and the later work in direct support of Title I and Title II design as well as prototype tests would be performed using line-item funds. Capital equipment money would be used to construct test facilities. Subsections 3.9.1 through 3.9.5 provide more detailed information on the WBS level four tasks under this activity.

3.9.1 Beam Transport System Development

3.9.1.1 Justification for the Beam Transport System Development Task

A major feature of the ANS will be the provision of neutron beams to a large number of experimental stations via a judicious combination of direct beams and neutron guides. Because the size and shape of the beam tubes affect the core reactivity, the detailed core neutronics cannot be assessed until these parameters are known. In the case of the guides, the dimensions chosen also impinge critically on the cold source volume. There is an intrinsic conflict between reactor and cold source design requirements—which would seek to minimize the beam dimensions—and the experimental requirements—which demand the largest beams consistent with instrumental resolution. The beam transport system development task is to develop methods, codes, and models that will permit optimal choices of beam and guide sizes and shapes, and to determine optimal methods of delivering the neutrons to the different experimental stations. Tests will be performed on key components (e.g., guides, windows, shutters) to assess performance and the effects of exposure to radiation.

3.9.1.2 Description of the Beam Transport System Development Task

Six major subtasks have been identified and are described below:

1. Neutron beam tube dimensions will be evaluated using simulation of a wide class of experiments and an optimal size will be identified.

2. Neutron guide tube dimensions and optical coating specifications will be evaluated using simulation of a wide class of experiments, and an optimal design will be identified. This assessment will include the impact of the dimensions on the cold source design.

Table 3.9. Summary description of the beam tube, guide, and instrument development task work breakdown structure level four tasks

WBS	Task description
1.1.9.1	Beam transport system development—A major feature of the ANS[a] will be the provision of neutron beams to a large number of experimental stations via a judicious combination of direct beams and neutron guides. This task provides the methods, codes, models, and tests necessary to make optimal choices of beam and guide sizes and shapes.
1.1.9.2	Polarizer development—This task is to develop current polarizer technology and, if necessary, new technology to the point where it will ensure the production of ANS beams of sufficiently high intensity and polarization to revolutionize this important field of research.
1.1.9.3	Monochrometer development—This task is to select, evaluate, and test the best methods of monochromating for the different classes of instruments.
1.1.9.4	Detector development—This task will endeavor to develop methods to maximize detection area concommitantly with maintaining the highest possible signal-to-noise ratios.
1.1.9.5	Instrument systems development—This task will identify specific classes of instrument development and provide modular designs for each element so that the very broad range of instruments foreseen for the ANS may be constructed as much as possible from a minimum number of interchangeable, standard units.

[a]ANS=Advanced Neutron Source

3. The limits on the design and manufacture of the thin-film optical guide coatings, known as supermirrors, will be explored by theoretical simulation, followed by prototype fabrication and measurement by neutron reflectivity at HFIR, the National Institute of Standards and Technology, and elsewhere.

4. Once optimal supermirror design is established, a prototype thermal neutron guide section and a cold neutron beam bender using supermirror coatings will be fabricated and tested. Test specimens of the coatings and substrates will be evaluated for resistance to irradiation damage. Substrate evaluation will define whether the in-pile guide front-ends require metallic rather than glass substrates.

5. Shutter designs for beam closure and safety shutoff will be assessed and the best designs will be incorporated into a beam-tube mockup and evaluated for mechanical performance and fail-safe design.

6. Prototype target transport mechanisms for the through tubes and, if necessary, the slant beam tubes will be developed.

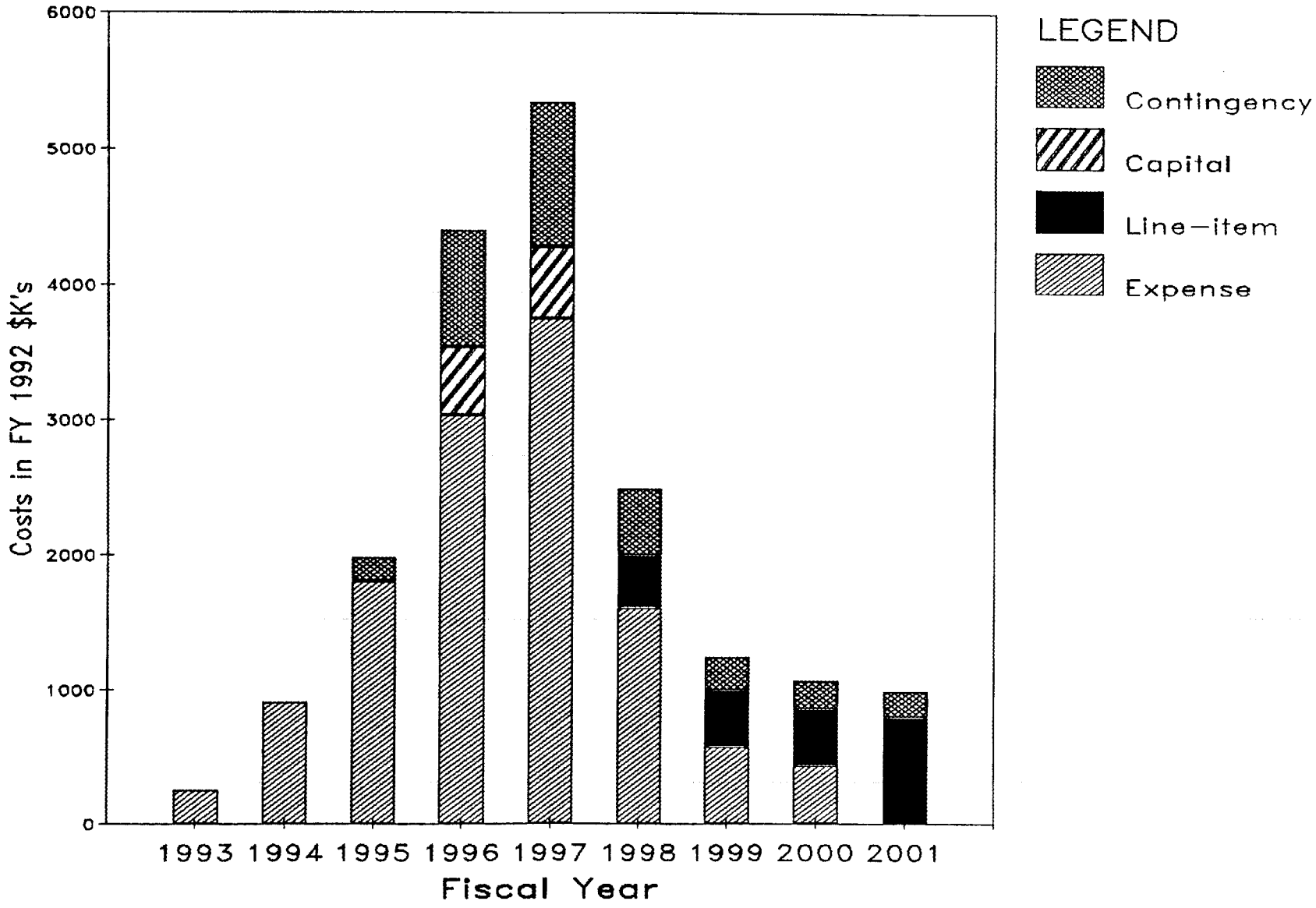

Fig. 3.17. Summary of cost estimates for work breakdown structure 1.1.9, beam tube, guide, and instrument development.

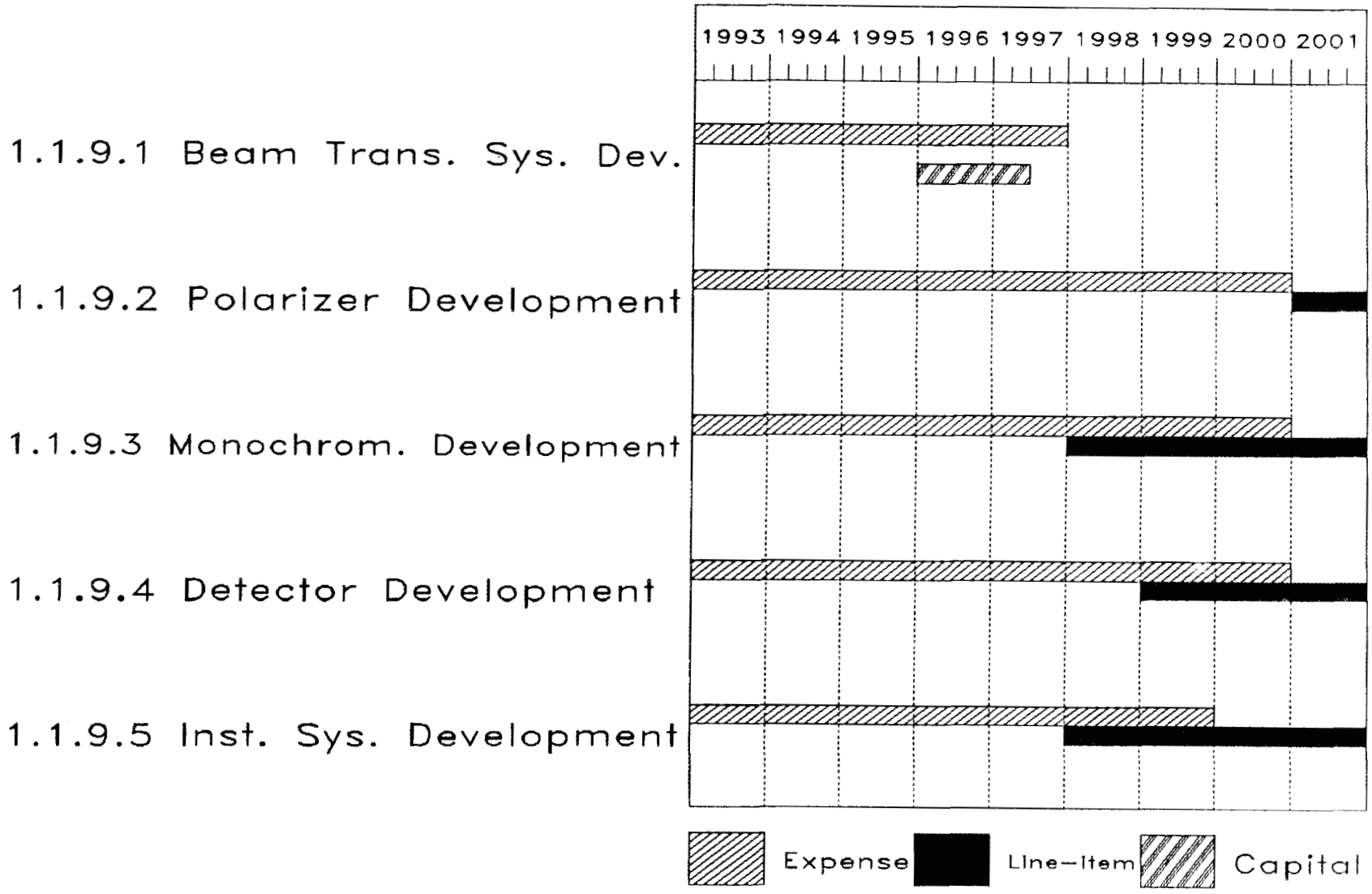

Fig. 3.18. Schedule for work breakdown structure 1.1.9, beam tube, guide, and instrument development.

3.9.2 Polarizer Development

3.9.2.1 Justification for the Polarizer Development Task

One of the most interesting properties of the neutron as a research tool is its magnetic moment, which permits a wide variety of unique experiments using appropriately polarized neutron beams. This technique has never been fully exploited, however, because of intensity limitations due to inefficient polarizing and analyzing methods. Current high flux reactors are able to provide polarized beams of sufficient intensity to allow specialized, but not routine, experiments of this type. This task is to develop current polarizer technology and, if necessary, new technology to the point where it will ensure that the production ANS beams have sufficiently high intensity and polarization to revolutionize this important field of research.

3.9.2.2 Description of the Polarizer Development Task

This task has been broken down into seven subtasks:

1. The design and fabrication of magnetic supermirrors for neutron polarization will be assessed by simulation followed by single mirror fabrication and testing.

2. The optics of transmission polarizers will be evaluated by ray-tracing techniques for different magnetic supermirrors, and a practical design that can polarize a divergent beam will evolve.

3. The use of supermirrors on silicon wafers to split a beam into its two component spin states and then transport each spin state in a different direction will be evaluated as a means of avoiding the factor-of-two "wastage" normally inherent to polarizing a beam.

4. New solid state magnetic materials, such as Nd-Fe-B, will be evaluated for use in unpowered magnetic circuits to guide polarized beams.

5. Different methods for making spin-turn devices, such as polarization flippers, will be assessed. These include Larmor precession coils tuned at a given wavelength, and white beam devices such as "figure-8" current sheets or cryogenic Meissner-effect methods; the latter may be highly influenced by progress in high-temperature superconductors.

6. Field-integral correction methods will be evaluated, using high-precision magnet-design programs, to choose between Fresnel-type corrections (using material in the beam) and optimal-field-winding techniques.

7. The results of subtasks 4–6 above will be used to fabricate a linear neutron spin-echo setup to test prototype devices and to assess the utility of different nonmagnetic materials (e.g., nonferrous bearings) in polarized-beam spectrometer construction.

3.9.3 Monochromator Development

3.9.3.1 Justification for the Monochromator Development Task

An important feature of the ANS is the wide spectral range offered to experimentalists via the beams looking at different neutron sources (cold, thermal, and hot). Further, it is intended to provide beams having characteristics that allow exploitation of the most modern focusing techniques. This feature will require a variety of state-of-the-art monochromating methods, using crystals, thin-films, mechanical choppers or selectors, and possibly Drabkin resonance techniques. The monachromator development task is to select and test the best methods of monochromating for the different classes of instruments.

3.9.3.2 Description of the Monochromator Development Task

This task has been divided into six subtasks:

1. The use of different crystalline materials for focusing crystal monochromators will be evaluated theoretically for both one- and two-dimensional focusing by simulating different classes of experiments. Likely candidates will be procured and tested by measuring mosaic spread and reflectivity.

2. Available (and possibly new) methods of fabricating velocity selectors will be examined and a prototype fabricated to test the best candidate. Ray-tracing codes will need to be developed for divergent-beam velocity selector design.

3. Disk chopper limits (imposed by magnetic bearing stability, etc.) will be assessed and used as input to an evaluation of the optimal number of rotating and counter-rotating choppers in a time-of-flight system.

4. Crystals for polarizing monochromators will be evaluated for use at thermal and hot wavelengths.

5. Spin-turn devices will be assessed for switching speed for use as magnetic choppers in polarized beams, particularly to produce pseudo-random sequences of pulses for correlation spectroscopy.

6. Doppler-drive techniques (e.g., piston, Ferris wheel) will be evaluated for use in backscattering spectrometers.

3.9.4 Detector Development

3.9.4.1 Justification for the Detector Development Task

The efficient exploitation of the ANS requires minimal waste in detecting scattered neutrons on the various instruments. The detector development task will endeavor to develop methods to maximize detection areas concommitantly with maintaining the highest possible signal-to-noise ratios and minimal cross-talk.

3.9.4.2 Description of Detector Development Task

This task has been divided into five subtasks:

1. Different methods of using gas detectors as position-sensitive devices will be evaluated. In particular, time-encoding vs multiwire encoding will be examined, along with different digitizing methods.

2. Scintillator-based detectors will be assessed for neutron efficiency and gamma-rejection. Methods of reading scintillator output (e.g., Anger camera geometry, fiber-optic coupling) will be compared.

3. Solid state detectors that use charge conversion devices read directly by an on-chip amplifier and register will be examined and assessed for trade-off between efficiency vs large area capability.

4. Real-time imaging methods based on video technology will be tested for use in both alignment and measurement.

5. The practicality of using batch-mode devices, such as image-plates, for precision measurements will be assessed, especially in conjunction with the availability of real-time methods for alignment purposes.

3.9.5 Instrument Systems Development

3.9.5.1 Justification for the Instrument Systems Development Task

Most modern techniques in neutron instrumentation have been developed since the workhorse spectrometers in HFIR and the High Flux Beam Reactor were built. The instrument systems development task will identify specific classes of instrument development and provide modular designs for each element so that the very broad range of instruments foreseen for the ANS may be constructed as much as possible from a minimum number of interchangeable, standard units.

3.9.5.2 Description of the Instrument Systems Development Task

This task has been divided into four subtasks:

1. Available commercial mechanical modules for monochromator, sample, analyzer, and detector support and positioning will be procured and assessed for (a) methods of making the modularity more global to minimize the number of different components and (b) methods of improving the positioning speed and reliability to minimize the time spent scanning relative to counting.

2. A prototype 3-axis spectrometer will be used to measure the performance of the best module designs, safety interlocks, and modular computer hardware and software for spectrometer control and data-logging and analysis.

3. The results of WBS 1.1.9.2 (see Sect. 3.9.2) will be used to design a prototype spin-echo spectrometer to assess the effects (and correction) of stray fields at different scattering angles.

4. A prototype shielding drum will be built to test methods of wedge movement during rotation, etc. Particular attention will be paid to developing a fail-safe design.

3.10 HOT SOURCE DEVELOPMENT—WBS 1.1.10

Hot source development includes the reactor physics, thermal-hydraulics, and structural analyses necessary to support the development of an ANS hot source facility. In addition, tests will be performed to support the development of a prototype and later to demonstrate the prototype operation.

This WBS element contains one major project milestone:

Complete hot source prototype tests by the end of September 1998. The prototype tests will be performed with heavy involvement from the design team and with emphasis on identifying design flaws early in the testing process. Therefore, the end of the test program is considered a confirmation of the design and is scheduled to allow an additional 9 months before the completion of the Title II design phase.

The hot source development activity is divided into five WBS level four tasks summarized in Table 3.10. Most of this work would be performed at INEL, ORNL, and other subcontractors to be determined. The total estimated costs for this activity over the 9-year period covered by this R&D plan are given in Fig. 3.19, and the associated schedules are shown in Fig. 3.20. The initial development work would be performed using expense money, and the later work in direct support of Title I and Title II design as well as prototype tests would be performed using the line-item money. Capital equipment money would be used to construct test facilities. Subsections 3.10.1 through 3.10.5 provide more detailed information on the WBS level four tasks under this activity.

3.10.1 Neutronics Analyses

3.10.1.1 Justification for the Neutronics Analyses Task

The hot source location, heat loads, and materials are either determined by or greatly impacted by neutron physics considerations: (1) The hot source must be located in a high enough thermal flux field that the hot flux goals can be achieved; (2) the present plan is to use the reactor core neutron and gamma sources to supply the heat loads necessary to obtain desired hot source moderator temperatures; and (3) the efficiency of the moderator material depends on the material cross sections. Therefore, significant neutronics support is needed to develop a hot source concept.

Table 3.10. Summary description of the hot source development work breakdown structure level four tasks

WBS	
1.1.10.1	Neutronics analysis—This task will provide the development of models, data (e.g., cold cross sections, kerma factors) and analyses necessary to optimize and analyze the physics of the hot neutron source device. Geometry studies will be performed to support design optimization. In addition, models will be used to estimate the neutron and gamma heating sources in the various regions of the hot source device.
1.1.10.2	Thermal analysis—This task will provide the models and analyses necessary to simulate the thermal-hydraulic behavior of the hot source system. Both normal and abnormal conditions will be examined.
1.1.10.3	Stress analysis and structural design—This task will provide the models and analyses necessary to determine the structural behavior of the hot source components during normal and abnormal conditions.
1.1.10.4	Prototype development—This task will use data from subtasks 1.1.10.1, 1.1.10.2, and 1.1.10.3 to develop a prototype hot source system.
1.1.10.5	Prototype demonstration—This task will demonstrate the hot source prototype developed under work breakdown structure 1.1.10.4. A hot source test facility will be designed and fabricated, a prototype hot source will be fabricated, and tests will be performed. The results of these tests will be used to support final design and the Final Safety Analysis Report.

3.10.1.2 Description of the Neutronics Analyses Task

Parametric studies will be performed using 3D neutronics models to study material options, the impact of location, and special geometry considerations such as the potential for using a reentrent cavity for the hot source. Neutron spectra obtained from these studies then will be used to optimize the hot source geometry. Activation analyses will be performed to determine the long-term impact of the very high flux field on the hot source materials. Another subtask will be to examine the interfaces between the hot source and the reactor core to determine whether the hot source has a significant effect on power distribution or core reactivity. Additional detailed neutronics analyses will be performed as necessary to support the design team.

3.10.2 Thermal Analyses

3.10.2.1 Justification for the Thermal Analyses Task

The thermal behavior of the hot source is especially important because it determines the thermal equilibrium spectrum of the neutrons entering the hot beam tubes. Models and tests of thermal conductivity of the insulation material must be developed to

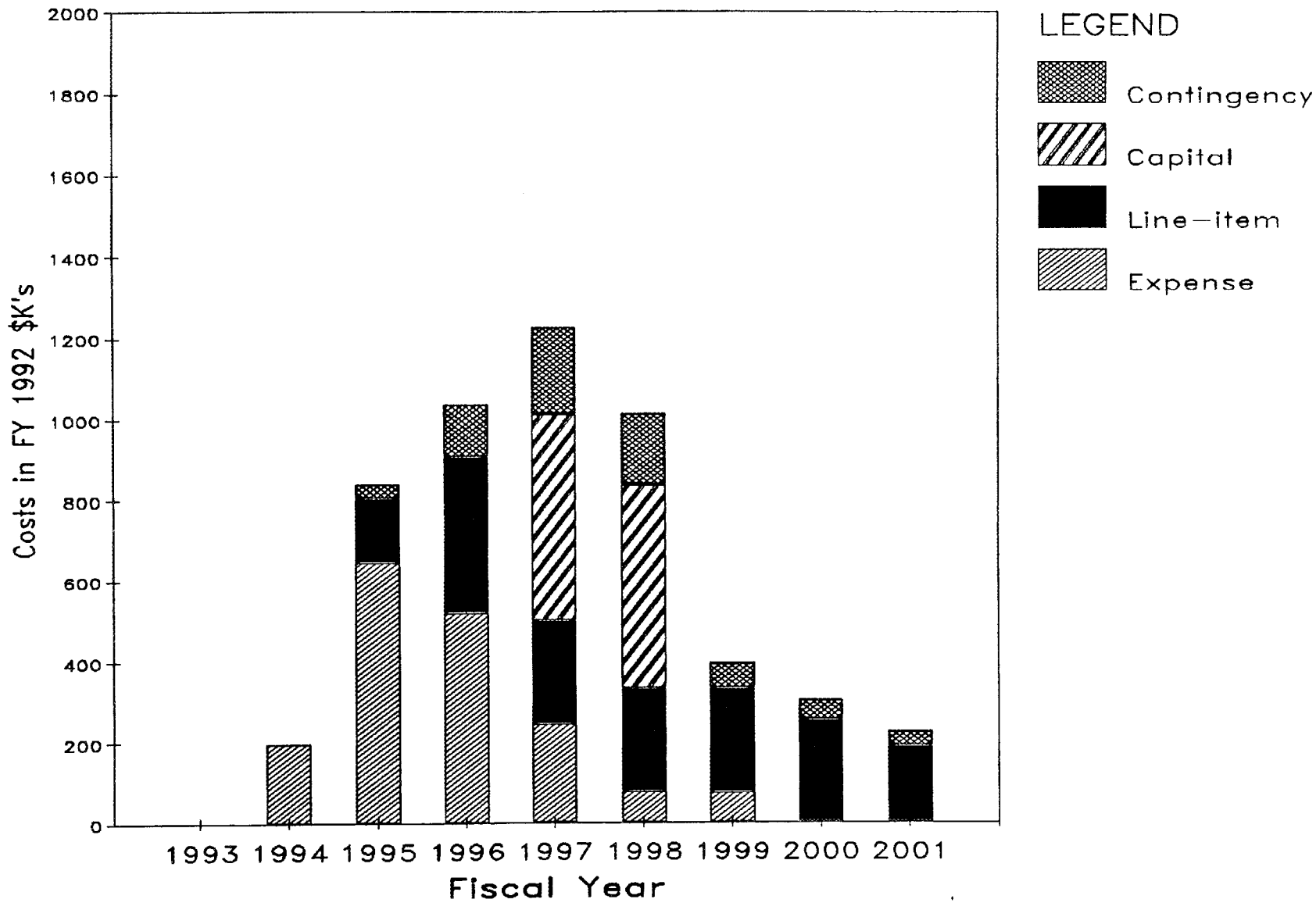

Fig. 3.19. Summary of cost estimates for work breakdown structure 1.1.10, hot source development.

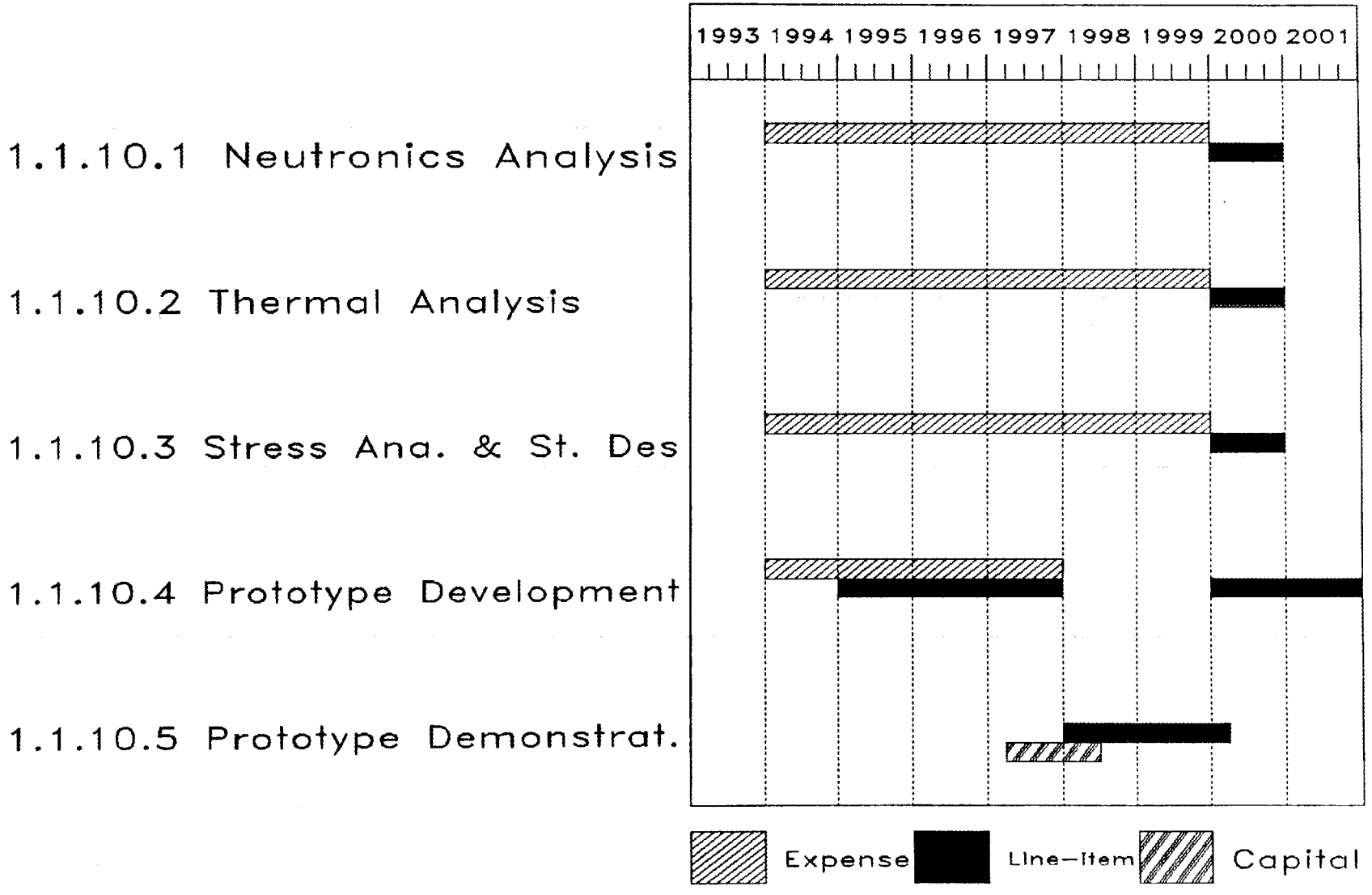

Fig. 3.20. Schedule for work breakdown structure 1.1.10, hot source development.

ensure that performance objectives are met. Analyses also will be necessary to support the safety case for the hot source.

3.10.2.2 Description of the Thermal Analyses Task

Thermal models of the hot source will be developed and used to determine the amount and type of insulation needed to achieve moderator temperature goals. Tests will be performed as necessary to confirm insulator thermal conductivity at design temperatures. Detailed analyses will be performed as necessary to support the design team, including analysis of the safety case.

3.10.3 Stress Analyses and Structural Design

3.10.3.1 Justification for the Stress Analyses and Structural Design Task

The design team will require various stress analyses to support the development of a basic hot source concept (i.e., geometry, material, etc.). Material expansion and other material properties must be examined for the proposed high temperature levels. In addition, irradiation damage must be evaluated to determine lifetime limits of the hot source structures.

3.10.3.2 Description of the Stress Analyses and Structural Design Task

The details of this task have not been fully defined. To estimate costs for this task, a level of effort has been assumed to support design. It is clear, however, that this task is not intended to provide the full detailed Title I and Title II structural analyses. But some money has been included in this task to help support Title I and Title II activities for aid in model development, for consideration of special issues, and for review.

3.10.4 Prototype Development

3.10.4.1 Justification for the Prototype Development Task

The uniqueness of the environment envisioned for the ANS hot source requires that the R&D team work very closely with the hot source designers in developing the concept. The prototype development task provides the interface with the hot source engineering design team and provides technical support in all areas not previously mentioned. One important subtask will be to help set the safety philosophy for the hot source and to perform analyses necessary to support the safety case.

3.10.4.2 Description of the Prototype Development Task

The hot source development R&D team will provide analysis support as necessary to complete the design of a prototype hot source system. This task also will support the safety analysis of the hot source and will obtain data as necessary to prepare the PSAR and FSAR sections on the hot source.

3.10.5 Prototype Demonstration

3.10.5.1 Justification for the Prototype Demonstration Task

The hot source system will be a unique design that should be tested in an environment much simpler than that of the completed ANS facility. These tests will focus on the thermal aspects of the design and will endeavor to confirm model estimates of the temperature profiles across the hot source moderator. The prototype demonstration task also will establish operating constraints and other information deemed necessary to perform startup testing of the hot source. If performance flaws are identified, these tests will provide the opportunity for design change prior to installation into the ANS.

3.10.5.2 Description of the Prototype Demonstration Task

Working with the engineering design team, a detailed test plan will be developed for the hot source prototype demonstration. A hot source test facility will be designed and fabricated. Tests will cover measurement of temperature profiles, material vaporization, and mass transfer of the moderator. Early tests are expected to focus on ensuring that there are no major hot source design flaws. Tests will be performed beyond the design confirmation milestone to provide additional data for the FSAR.

3.11 NEUTRON AND GAMMA TRANSPORT AND SHIELDING—WBS 1.1.11

The neutron and gamma transport and shielding activity contains the tasks that require detailed transport analyses of neutron and gamma sources.

This WBS element does not contain a major project milestone even though it does supply substantial support to various design activities.

The neutron and gamma transport and shielding activity is divided into four WBS level four tasks summarized in Table 3.11. Most of this work would be performed at INEL, ORNL, and other subcontractors to be determined. The total estimated costs for this activity over the 9-year period covered by this R&D plan are given in Fig. 3.21, and the associated schedules are shown in Fig. 3.22. The initial model development and parametric studies would be performed using expense money, and the later work in direct support of Title I and Title II design would be performed using the line-item money. A small amount of capital equipment money is provided early in the program to purchase a computer work station that would be dedicated to analyses associated with this task. Subsections 3.11.1 through 3.11.4 provide more detailed information on the WBS level four tasks under this activity.

3.11.1 Beam Transport Analyses

3.11.1.1 Justification for the Beam Transport Analyses Task

Experimental verification of the ANS beam tube performance is not practical. Therefore, design decisions concerning beam tube configurations and locations must rely upon analysis of beam tube performance, i.e., the predictions of the neutron and gamma fields in the beam tubes. In addition, the shielding design analysis must rely on the

Table 3.11. Summary description of the neutron and gamma transport and shielding level four tasks

WBS	
1.1.11.1	Beam transport analysis—This task provides for the development of physics models of the beam tubes and the resulting analyses of neutron and gamma flux fields in the beam tubes. In addition, source terms due to leakage from the beam tubes will be determined along the beam tube and supplied to the shielding support activity.
1.1.11.2	Impact of reflector components—The introduction of components and experiments into the reflector region will produce local perturbations that may affect other components including the reactor core and the reactor control components. This task will provide the development of three-dimensional models of the reflector region with appropriate components and the analyses to address this issue.
1.1.11.3	Analyses of heat loads and activation—This task will provide the analyses necessary to determine neutron and gamma heat loads for various components. Expected fluences for components will be determined and provided to the materials evaluation activities under work breakdown structure 1.1.7. In addition, the activation level of components will be evaluated under this task.
1.1.11.4	Support to design of shielding—A limited amount of shielding support is supplied by this task. Work will focus on evaluation of unique shielding problems. Methods and models to be used in design and evaluation of shields also will be developed under this task.

prediction of neutron and gamma leakages from the beam tubes as calculated in the analysis under the beam transport analyses task.

3.11.1.2 Description of the Beam Transport Analyses Task

This task has two subtasks: (1) the development of neutronics models and analysis to evaluate beam tube designs and (2) the calculation of neutron and gamma fields as sources for subsequent shielding analysis. A description of each subtask follows:

1. This subtask determines the calculated neutron and gamma fluxes within the reflector system and calculates the highly anisotropic flow of neutron and gamma rays through the beam tubes. For such difficult problems, modeling compromises between accuracy and calculational feasibility are necessary. Various modeling options will be evaluated, and modeling of existing systems (HFIR or ILL) will be performed to validate modeling assumptions. To date, the analyses have been carried out by coupling two 2-D discrete ordinate calculations. More detailed calculational models will be incorporated in the analysis as the core and beam tube design progress.

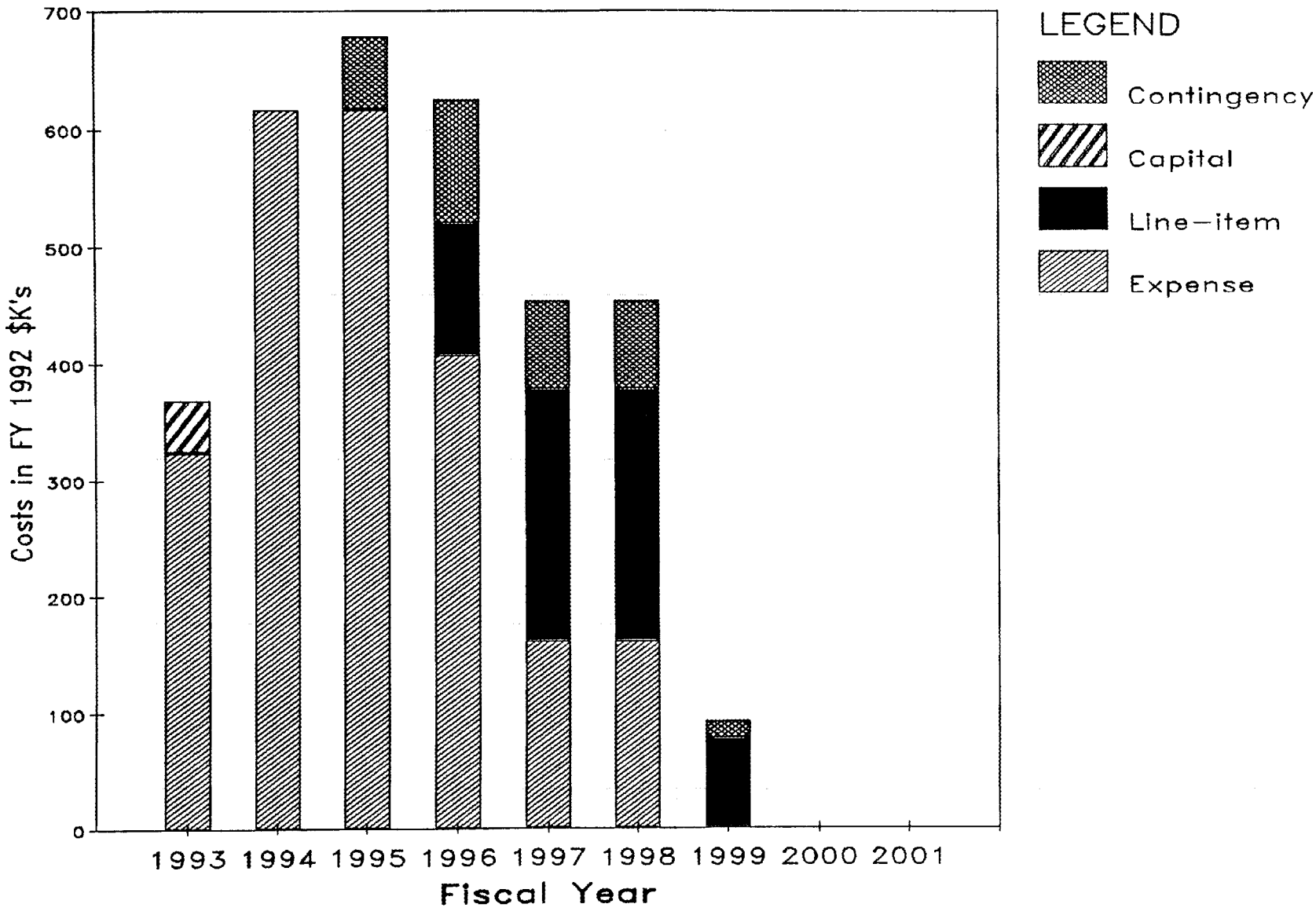

Fig. 3.21. Summary of cost estimates for work breakdown structure 1.1.11, neutron and gamma transport and shielding.

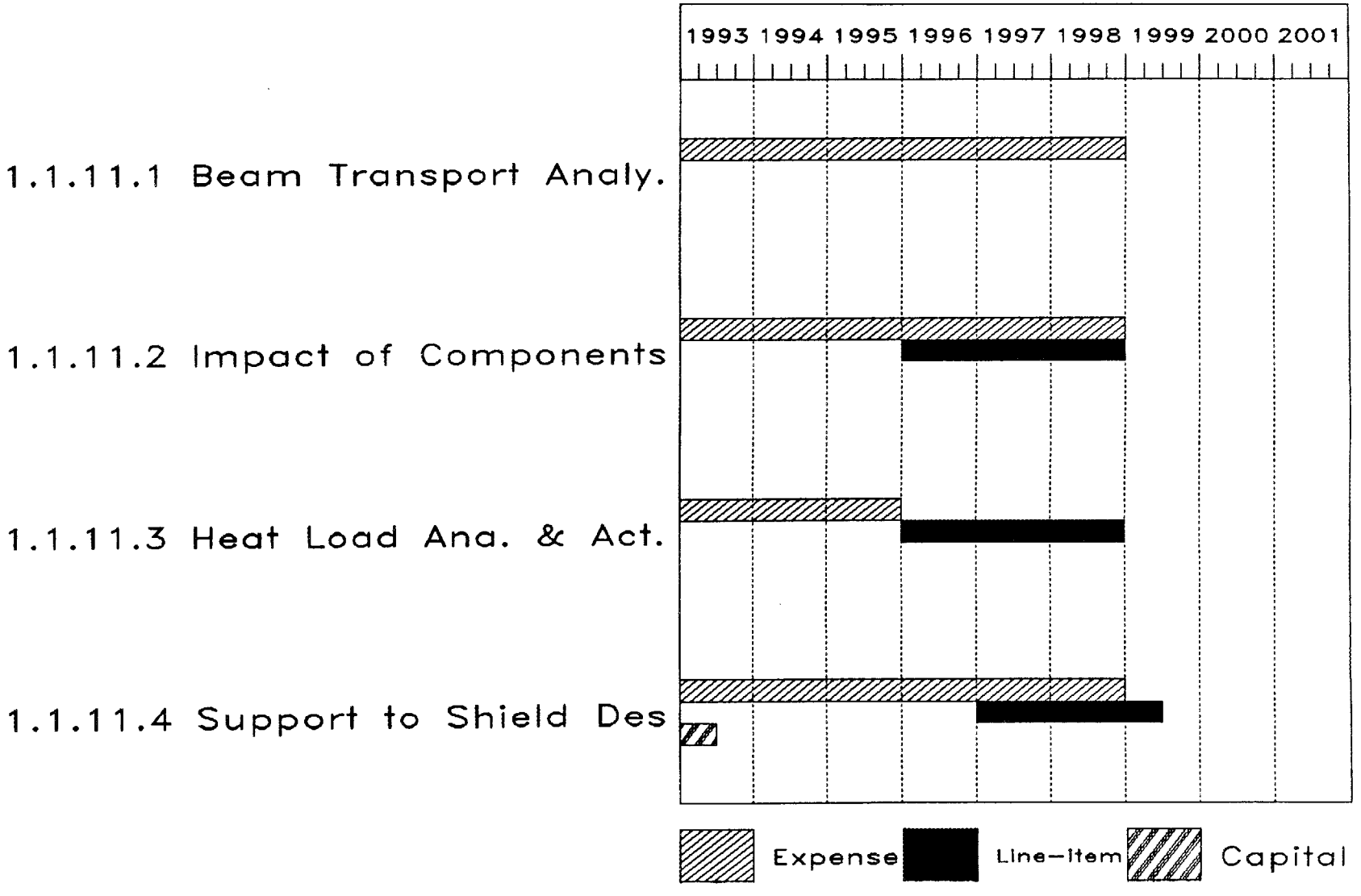

Fig. 3.22. Schedule for work breakdown structure 1.1.11, neutron and gamma transport and shielding.

2. This subtask supplies the neutron and gamma leakages from the beam tubes as a source term to support shielding analyses.

3.11.2 Impact of Reflector Components

3.11.2.1 Justification for the Impact of Reflector Components Task

An evaluation of design options for the various components in the reflector vessel must include analysis of the impact of these components on core reactivity and on neutron flux in regions of interest. Because these impacts cannot be determined experimentally during the design process, accurate neutronics analysis must be relied upon. This task is to develop accurate models for analysis and to calculate core reactivity and key flux densities for each of the reflector component design configurations.

3.11.2.2 Description of the Impact of Reflector Components Task

This task provides for the development of the computational models and for the analysis to determine how reactor performance is affected by the reflector components and experiments located in the reflector region. The subtasks are as follows:

1. The neutronic models used for this analysis must be 3-D in nature to represent the reflector components accurately. This subtask provides for the development of 3-D Monte Carlo models of the core/reflector system, with explicit modeling of each reflector component. Modeling is updated to follow the design evolution and to assess component design options.

2. Using the models developed above, analyses will be performed to determine the effect of reflector components, individually and collectively, upon core reactivity, neutron flux densities, and power density distribution within the core fuel region. The perturbed flux at the beam tube mouth will be calculated and provided to the beam tube analysis task (WBS 1.1.11.1).

3.11.3 Analyses of Heat Loads and Activation

3.11.3.1 Justification for the Analyses of Heat Loads and Activation Task

An evaluation of design options for the core and reflector components must include a determination of the heat generated in the components, the neutron and gamma fluences experienced in each component, and the radioactivity generated within each component (activation of each component). Knowledge of the heat generated within each component and the neutron and gamma fluences accumulated in each component is necessary data required for the studies carried out under WBS 1.1.7 to assess the material and structural integrity of each component. Activation of each component is also important data that must be obtained to evaluate handling needs and requirements during component replacement and/or maintenance. Because these impacts cannot be determined experimentally during the design process, accurate neutronics analysis must be relied upon.

3.11.3.2 Description of the Analyses of Heat Loads and Activation Task

The work under this task has been divided into three subtasks:

1. Neutron and gamma heat loads are analyzed. The heat loads are determined by coupled neutron/gamma flux calculations at various times in the fuel cycle. The calculated fluxes and kerma factors then are used to determine the component heat load. The heat load distribution within a component is calculated as required. Heat loads are calculated for all reactor components, including fuel elements, side plates, control rods, CPBT, reflector components, shielding, and reflector vessel. Methods for validating heat load calculation will be determined as part of this subtask and appropriate validation exercises will be performed.

2. Neutron fluences and gamma fluences are determined per fuel cycle from the coupled neutron/gamma flux calculations over the fuel cycle. The fluences are provided to the materials evaluation task (WBS 1.1.7).

3. The neutron and gamma flux data calculated under subtask 2 are input to a depletion code for evaluation of the activation within each component as a function of time after shutdown. The energies and magnitudes of radioactivity associated with the activation of components are determined and used in the component replacement and handling studies.

3.11.4 Support to the Design of Shielding

3.11.4.1 Justification for the Support to the Design of Shielding Task

The shielding of components and personnel from harmful neutron and gamma radiation resulting from reactor operation and out-of-core fuel and component transfers must be accomplished through the proper choice of shielding materials and configuration. The necessary shielding requirements must be determined during the design process. Because the shielding requirements cannot be determined experimentally during the design process, accurate analysis of gamma and neutron fluxes must be relied upon.

3.11.4.2 Description of the Support to the Design of Shielding Task

By nature, the shielding analysis relies upon sophisticated neutron/gamma transport methods and upon accurate representation of the configuration of the areas to be shielded. This task provides for the development of models and methods for evaluating shielding designs and for the supporting analysis to determine shielding requirements. The work is divided into three general subtasks:

1. The methods and models needed to support shielding design evaluations will be determined. A range of evaluations is anticipated, from scoping calculations to analysis of detailed design configurations. Models will be developed for use in coupled neutron/gamma discrete ordinates and Monte Carlo codes. Methods capable of calculating adjoint fluxes will be of particular interest.

2. Shielding requirements will be determined and design options for shielding components and personnel will be evaluated. Evaluations of special shielding issues, such as shielding designs for beam tubes and cold source guides, will be particularly emphasized.
3. The shielding of components and personnel during component replacement and maintenance, in addition to shielding from radiation from reactor operation, is a concern. The evaluations carried out under this subtask will support the determination of shielding requirements related to refueling and replacement of nonfuel components such as the CPBT or control rods.

3.12 INSTRUMENTATION AND CONTROL SYSTEMS DEVELOPMENT

The ANS operational objectives place unusual demands on the operating regimes which require unique control and plant protection system capabilities. Instrumentation and control systems development is provided to examine the capability of different controls and plant protection systems to meet operational and safety needs. The extent to which special control and plant protection system capability can be developed will affect the reactor's design concepts and its eventual operational capabilities.

This WBS element contains one major project milestone:

Complete prototype tests of the reactor protection system by the end of June 1997. This subtask provides the final confirmation of the performance of the reactor protection system in time for design adjustments, if necessary.

The instrumentation and control system development activity is divided into 11 WBS level four tasks summarized in Table 3.12. Most of the work would be performed at ORNL and at subcontractors to be determined later. The total estimated costs for this activity over the 9-year period covered by this R&D plan are given in Fig. 3.23, and the associated schedules are shown in Fig. 3.24. The initial development work would be performed using expense money, and the later work in direct support of Title I and Title II design, as well as prototype tests, would be performed using the line-item money. Capital equipment money would be used to construct test facilities. Subsections 3.12.1 through 3.12.11 provide more detailed information on the WBS level four tasks under this activity.

3.12.1 Dynamic Model

3.12.1.1 Justification for the Dynamic Model Task

The dynamic model has proved to be useful in the conceptual design of the reactor and cooling systems. Further development of the model will be important to advanced conceptual design as the reference design is modified. It also is important for studying control system concepts for startup, normal operation, shutdown, transients, and accidents.

Table 3.12. Summary description of the instrumentation and controls system development work breakdown structure level four tasks

WBS	Task description
1.1.12.1	Dynamic model—Develop, update, and employ a dynamic model of important characteristics of the ANS[a] systems. The primary purpose of the model is to aid the development of safety and control features, but it also will aid in making decisions on the design of thermal-hydraulic systems. Although the basic model has already been developed, this modeling is expected to continue throughout the project and to be modified as the reactor system design changes.
1.1.12.2	Safety analysis support—Support safety analyses by developing quick-look data with the dynamic model, establishing uncertainty bounds for control and safety parameters, and confirming the adequacy of protection system characteristics.
1.1.12.3	Reactor control—Develop the essential features and performance requirements for the reactor and heat removal control systems.
1.1.12.4	Reactor protection system—Develop the requirements and analyze the adequacy of ANS protection systems and develop requirements for components that are not commercially available, for example, release magnets, rod latches, control rod performance requirements.
1.1.12.5	Sensor development—Establish requirements for and initiate the development of nuclear and process sensors needed to meet the special needs of the Advanced Neutron Source where direct application of commercial products is not possible. Establish qualification requirements.
1.1.12.6	Instrumentation development—Define systems, establish requirements, and develop special instrumentation for the protection and control systems. Evaluate the suitability or adaptability of commercially available instrumentation.
1.1.12.7	Actuator development—Develop requirements for and details of special actuators that may be required for protection or control, for example, scram rod release magnets, position indicators, hydraulic actuators.
1.1.12.8	Diagnostics and surveillance—Develop requirements and methods for a reactor monitoring system to provide computational, diagnostic, and display capabilities for timely, concise, and directed information for operator aids, engineering diagnostics, maintenance, and administration.
1.1.12.9	Experiments systems—Develop requirements for the performance of experiment facilities, including the cold sources, and their interfaces with the reactor control and protection systems.
1.1.12.10	Control integration—Develop the philosophy, requirements, and techniques for the overall integration of plant control features, protection systems, and human factors of plant operation.
1.1.12.11	Subcriticality monitoring—Develop the requirements and identify and develop the techniques necessary for subcriticality monitoring of the reactor core, both installed and during refueling operations.

[a]ANS=Advanced Neutron Source

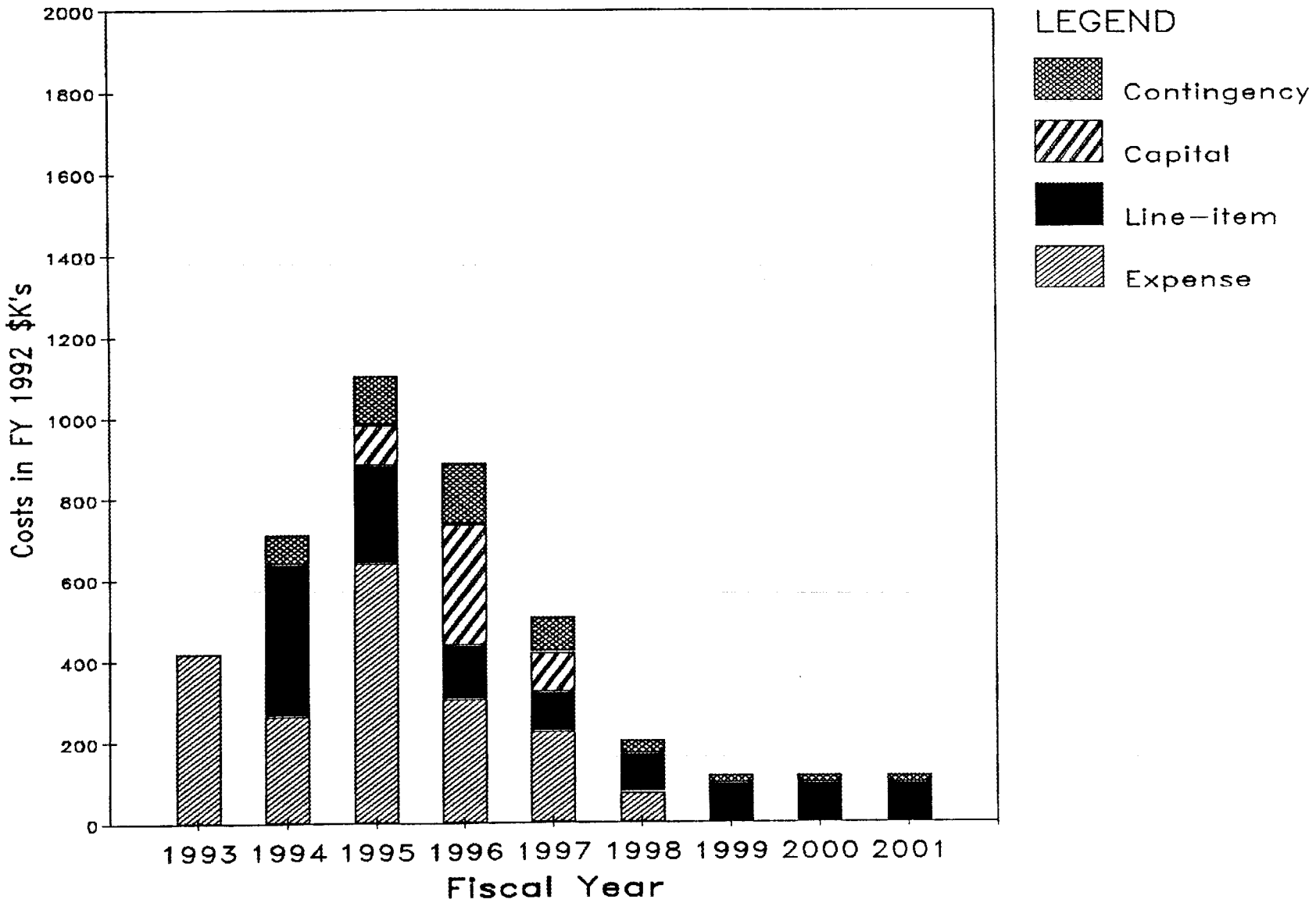

Fig. 3.23. Summary of cost estimates for work breakdown structure 1.1.12, instrumentation and controls system development.

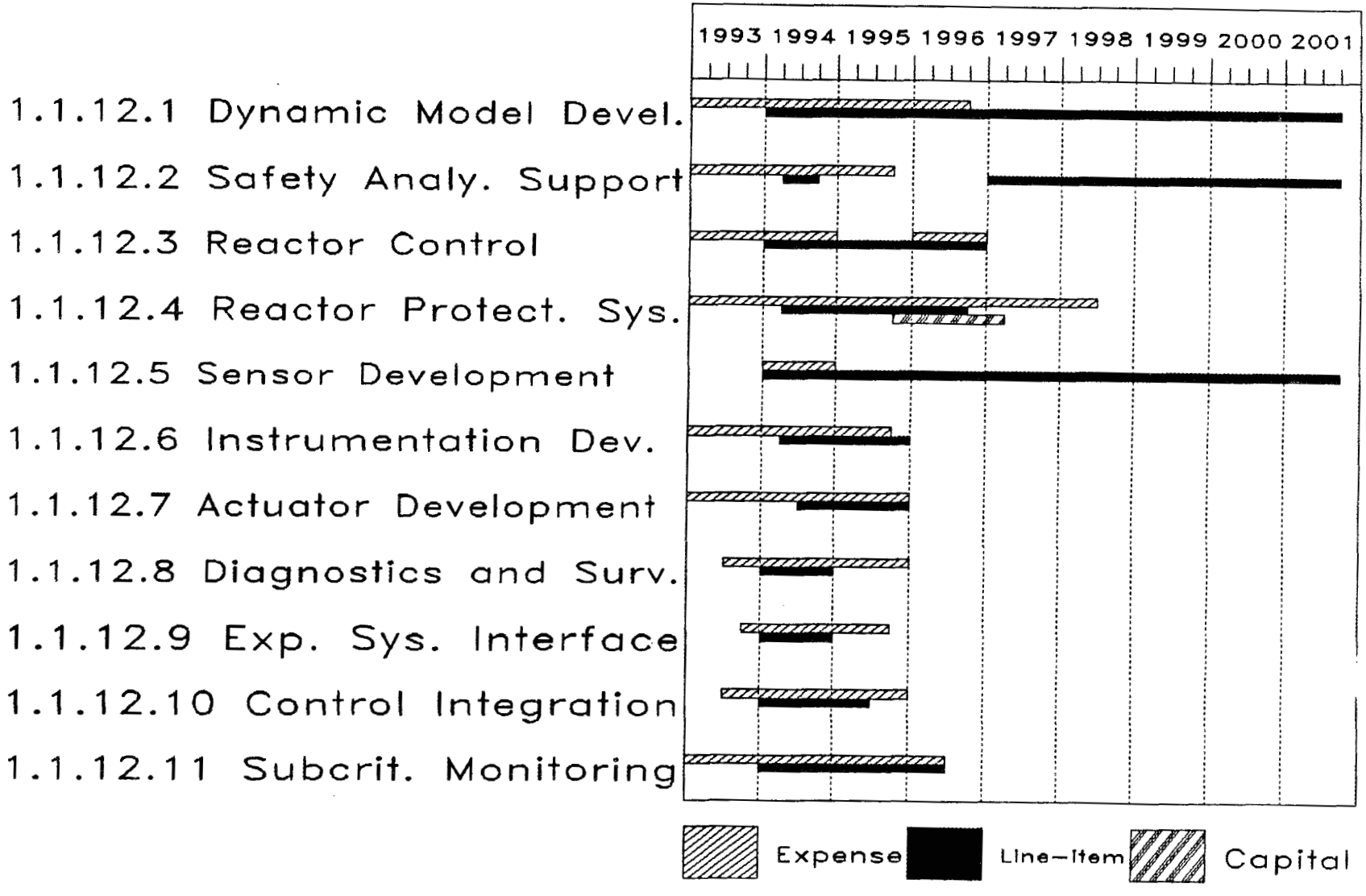

Fig. 3.24. Schedule for work breakdown structure 1.1.12, instrumentation and controls system development.

3.12.1.2 Description of the Dynamic Model Task

A dynamic model of important characteristics of the ANS systems will de designed, updated, and employed. The model is primarily to aid the development of safety and control features, but it also will aid in making design decisions for thermal-hydraulic systems. The model will be modified to evaluate possible design changes, to study the performance of control algorithms, and to support safety analyses. This modeling is expected to continue throughout the project. The next phase of model development will involve refining the secondary coolant system model so that the secondary system can be studied in detail. Documentation of the model will be improved and upgraded to a more useable manual. This task also will include a continuing effort to validate the dynamic model.

3.12.2 Safety Analysis Support

3.12.2.1 Justification for the Safety Analysis Support Task

Ensuring adequate instrumentation during and after accidents is critical to plant safety. The safety analysis support task provides information on plant response for transient conditions. The adequacy of plant instrumentation during accident conditions will be examined Documentation on instrumentation response also will be required to support the PSAR and the FSAR.

3.12.2.2 Description of the Safety Analysis Support Task

This task will support safety analyses by developing quick-look data with the dynamic model, establishing uncertainty bounds for control and safety parameters, and confirming the adequacy of protection system characteristics. Significant effort will be focused on examining the instrumentation required during and after accidents to ensure adequate monitoring during transient conditions. The effort will include assessing displays and instruments needed to meet the post-accident requirements. Documentation to support the PSAR and FSAR will be supplied as necessary.

3.12.3 Reactor Control

3.12.3.1 Justification for the Reactor Control Task

Control of the ANS requires strategies different from those for commercial plants and the HFIR. Analysis of control strategies identifies where some of the critical operational problems will occur. The design may need modification to meet the required steady state and transient performance. Part of the reactor control task will be analyzing the capability of the control system to avoid accidents or avoid the necessity to initiate safety systems. Because the ANS responds more quickly than commercial plants, the response times and data throughput requirements must be analyzed to determine what is adequate for the ANS control system.

3.12.3.2 Description of the Reactor Control Task

This task is primarily to develop the essential features and performance requirements for the reactor and heat removal control systems. It involves three subtasks:

1. A control strategy and algorithms will be developed to provide steady state and transient control of the plant. This subtask requires developing control strategies for different conditions and modes of operation and testing them on the dynamic model.

2. The information and controls operators' need for steady state and transient conditions will be determined. Although credit for the nonsafety control system cannot be used in a safety analysis, the Advanced Light Water Reactor passive plants are using nonsafety systems to avoid operation of the passive features. This approach may affect the number of times that the passive features must function.

3. The data throughput requirements for the distributed control system will be determined. The control study will determine the approximate sampling rates for different parameters and the response time or update rate for the control system. This information will be used to develop a model of the real time network for the distributed controller. Initially, this network will be developed in more detail as the control system develops.

3.12.4 Reactor Protection System

3.12.4.1 Justification for the Reactor Protection System Task

Special components in the reactor protection system must be developed and analyzed to ensure that they will function properly and that they can be qualified for Class 1E service. The interfaces between the mechanical and electrical components also must be developed and qualified. The reactor protection system task is included in the R&D effort because it requires development of nonstandard components with requirements that commercial reactors do not have.

3.12.4.2 Description of the Reactor Protection System Task

This task is to develop the requirements for and analyze the adequacy of ANS protection systems and develop requirements for components that are not commercially available, for example, release magnets, rod latches, and control rod performance requirements. The reactor protection system has unique requirements that commercial protection systems do not have. To meet these requirements, the actuators, latches, and electronics must be much faster than similar components in commercial plants. In addition, work will be performed to develop a model of the release magnets to evaluate coil requirements, holding forces, release time, and the magnetic properties of candidate materials. Support will be provided to the mechanical designers on the design of the mechanical components of the protection system. Design support also will be supplied to examine diversity requirements between the primary and secondary shutdown systems and

to evaluate designs that meet the time response requirements. As the likely candidates are identified, support will be supplied to develop prototype electronic models for more detailed analysis. One of the critical issues will be to develop a protection system that is acceptable to the regulating agency (as yet undefined); therefore, some emphasis will be on identifying those features most likely to cause problems because of their uniqueness. The performance of the reactor protection system will be demonstrated with a prototype to show that the unique problems have been solved. This information can be used to prepare functional specifications for the reactor protection system.

3.12.5 Sensor Development

3.12.5.1 Justification for the Sensor Development Task

The flux detectors require special consideration because they are mounted underwater, they are in an extremely high neutron flux and high delayed gamma flux, and they require fast time response. This situation is a different application from that in commercial light water reactors and requires development and testing. The temperature sensors require further R&D because direct immersion sensors are undesirable, and standard resistive temperature detectors in thermowells do not meet all of the ANS requirements. The sensors must be developed, tested, and qualified for nuclear service. If the sensors and thermowells are modified, the system must be qualified to be tested using loop current step response methods.

3.12.5.2 Description of the Sensor Development Task

The sensor development task will establish requirements for and initiate development of nuclear and process sensors needed to meet the special needs of the ANS where direct application of commercial products is not possible. The nuclear instrumentation for the ANS has some very special requirements, for example, high neutron flux, high gamma background, and detectors located under water. For this reason the detectors are not standard commercial detectors and require development and qualification. Alternative designs will be evaluated to determine which detectors can best meet the ANS requirements. Tests will be identified as needed to determine that the detectors can be used for the ANS.

Dynamic model studies indicate that the pressure sensors must be faster than pressure sensors typically used in the nuclear industry. A survey of the industry will be conducted to determine whether there are commercial sensors that can meet the time response, anticipated radiation dose, and other requirements. Qualification requirements for the candidate sensors will be identified.

The temperature sensors in the ANS must respond faster than those normally used in commercial plants. There are alternative designs that can meet the ANS response time requirements, but tests must be conducted to provide verification.

Another subtask will be to evaluate other safety and nonsafety sensors to determine whether additional special designs or tests are needed. For example, the flow sensor is used to calculate the reactor heat power, which makes it critical for determining trip points and margins between safety limits and trip set points.

3.12.6 Instrumentation Development

3.12.6.1 Justification for the Instrumentation Development Task

The nuclear industry has very special protection and control requirements, but the nuclear industry is not large enough to have a significant effect on the standard products delivered by commercial vendors. Field bus is expected to be used widely in many industries to replace the 4-20 ma standard. This may have a significant impact on safety, licensing, and obsolescence—issues for the ANS and the rest of the nuclear industry. Because the ANS may be the next nuclear plant built, it may lead the way in the use of new standard instrumentation. This issue needs to be addressed now to consider the impacts on the ANS.

3.12.6.2 Description of the Instrumentation Development Task

This task will define systems, establish requirements, and develop special instrumentation for the protection and control systems. One subtask will be to evaluate the suitability or adaptability of commercially available instrumentation. A significant change in process instrumentation will occur soon when the field bus standard (ISA SP-50) is approved. Acceptance of this standard is expected to antiquate the 4-20 ma current loop standard. Although equipment using 4-20 ma will still be manufactured, significantly fewer vendors and instruments will be available than currently. If the ANS instrumentation uses 4-20 ma, there may be support problems. The safety instrumentation almost certainly will not be field bus, but the nonsafety instrumentation may be field bus. Thus, field bus capabilities must be evaluated and their advantages and disadvantages documented.

3.12.7 Actuator Development

3.12.7.1 Justification for the Actuator Development Task

Actuators for the reactor protection system must be developed and tested as prototypes that will be used to prepare specifications for construction.

3.12.7.2 Description of the Actuator Development Task

This task will be to develop the requirements and details of special actuators that may be required for protection or control, for example, scram rod release magnets, position indicators, and hydraulic actuators. As the control and safety rod issues are refined, the requirements for the primary and secondary shutdown system actuators will be affected. Support will be provided to the mechanical designers to develop actuators that will meet the mechanical and electrical requirements. Because of the speed with which the reactor must be shut down, the actuator will not be a standard device available from the commercial nuclear industry. The response time requirement is a critical factor driving selection of the mechanical, electronic, and electrical components. The performances of and interfaces between these components must be accurately characterized and understood for correct operation. A prototype system will be developed and tested.

3.12.8 Diagnostics and Surveillance

3.12.8.1 Justification for the Diagnostics and Surveillance Task

The reactor monitoring system will be based on the existing plant computers, but it will be advanced in implementation and different because of special ANS requirements. R&D is needed to differentiate its requirements from those of a commercial system.

3.12.8.2 Description of the Diagnostics and Surveillance Task

The diagnostics and surveillance task will develop requirements and methods for a reactor monitoring system to provide computational, diagnostic, and display capabilities for timely, concise, and directed information for operator aids, engineering diagnostics, maintenance, and administration. The capability of the plant computers at existing commercial plants is expanding. The ANS reactor monitoring system may be similar to the Advanced Light Water Reactor monitoring systems presently in conceptual design. This task is to work with the reactor and process system designers to determine how the reactor monitoring system can support operation, maintenance, licensing, and management. The parameters that must be measured to perform the diagnostics will be defined. Special requirements for the ANS reactor monitoring system will be identified and compared with the requirements of the Advanced Light Water Reactor monitoring system.

3.12.9 Experiment Systems

3.12.9.1 Justification for the Experiment Systems Task

Commercial reactors do not have experiment protection systems, and the HFIR is currently reevaluating experiment-related safety issues. R&D is needed to expand on the HFIR philosophy and requirements to establish the philosophy and requirements for the ANS.

3.12.9.2 Description of the Experiment Systems Task

This task provides the requirements for the performance of the experiment facilities, including the cold sources, and their interface with the reactor control and protection systems. One issue to be addressed is the relationship between the reactor and the cold source operation. This issue requires coordination between the cryogenics designers and operators; the cold source designers; and the reactor, process, and instruments and controls designers. The reactor and process control strategy for recovery will be tested using the dynamic model. Those protection system parameters that are safety related and those that are nonsafety related must be identified. This information will be used to make requirements and specifications for instruments that connect to the experiment protection system and those that connect to the experiment control system.

3.12.10 Control Integration

3.12.10.1 Justification for the Control Integration Task

The control integration task provides bases for the specifications for the plant control systems. Safety and nonsafety issues will be examined to provide information for the Conceptual Safety Analysis Report and the PSAR. This information is important for meeting anticipated regulatory requirements.

3.12.10.2 Description of the Control Integration Task

The philosophy, requirements, and techniques for overall integration of plant control features, protection systems, and human factors of plant operation will be developed under this task. The conceptual design of the ANS uses integrated, distributed control because of the potential advantages of integrated control. However, such a system has never been used in the nuclear industry in the United States, and there are questions about its implications for reactor safety. The philosophy of integrated control will include summarizing the advantages and disadvantages or problems that many people in the process industries have experienced. The safety issues concerning the potential failure modes will be evaluated. Also, the issues being addressed by the Advanced Light Water Reactor project will be followed and examined. The philosophy will be used to expand the requirements and techniques of control further. The use of networks in protection systems will be examined further to determine the benefits of networks and distributed computers to the ANS. There may be significant design advantages in some cases, and less advantage in others. Issues such as the number and locations of distributed controllers will be considered.

3.12.11 Subcriticality Monitoring

3.12.11.1 Justification for the Subcriticality Monitoring Task

Subcriticality monitoring may affect the method and speed of refueling. Methods for monitoring subcriticality and the uses of such monitoring will be determined. This is an important issue that must be addressed by the R&D program because of its implications for design and operation of the plant.

3.12.11.2 Description of the Subcriticality Monitoring Task

This task will develop the requirements for and identify and develop the techniques necessary for subcriticality monitoring of the reactor core, both while installed and during refueling operations. R&D staff will work closely with the engineering design team to develop an acceptable process for monitoring fuel element subcriticality during all phases of the refueling process.

3.13 FACILITY CONCEPTS

The facility concepts activity includes all development work and tests not previous covered in the R&D program that are necessary to support the development of the reactor system.

This WBS element contains one major project milestone:

Complete component evaluation tests by the end of September 1995. This task provides important test data needed to start the Title II design effort for several major components.

The facility concepts activity is divided into four WBS level four tasks summarized in Table 3.13. Most of this work would be performed at ORNL. The total estimated costs for this activity over the 9-year period covered by this R&D plan are given in Fig. 3.25, and the associated schedules are shown in Fig. 3.26. The initial development work and the parametric testing will be performed using expense money, and the later work in direct support of Title I and Title II design, as well as prototype tests, would be performed using the line-item money. Capital equipment money would be used to construct test facilities. Subsections 3.13.1 through 3.13.4 provide more detailed information on the WBS level four tasks under this activity.

3.13.1 Reactor Components Test Facility

3.13.1.1 Justification for the Reactor Components Test Facility Task

The function of a number of components in the reactor system will have safety and availability impacts. Therefore, it is important that they be tested early in the design phase to provide performance data to the design team. Failure to test early will have certain cost and schedule penalties if later tests indicate design flaws or performances that do not meet design standards.

The operation of the seals of the reactor components is critical to the successful operation of the reactor system. The reactor components tests will thoroughly characterize the seals, the sealing interfaces, and the leak detection system that will verify the seal integrity during operation.

Locking ring/bolt torque tests are needed to ensure that the sealing concepts developed in the seal tests can be successfully installed using remote tooling.

Outer shutdown rod tests are required to characterize the rod system fully and verify performance.

Tests of the inner control rod latch are necessary to confirm the performance of the latch. Some of the parameters that must be examined are (1) the basic design of the latch, (2) wear on materials, (3) the size of an acceptable magnet, (4) latch (magnet) release time, and (5) the latch force transferred from the rod and springs to the magnet.

3.13.1.2 Description of the Reactor Components Test Facility Task

This task provides those tests that are not covered by other R&D tasks but which are necessary to develop or confirm the operational performance of reactor components.

Table 3.13. Summary description of the reactor concepts work breakdown structure level four tasks

WBS	Task description
1.1.13.1	Reactor components test facility—This task provides the necessary developmental testing of proposed reactor component designs to evaluate their suitability for the ANS and to explore the operational limits of particular designs. Included are seal tests, locking ring/bolt torque tests, outer shutdown rod tests, latch tests, and beam tube thimble collapse tests.
1.1.13.2	Reflector vessel and core flow tests—This facility will provide flow distribution and velocity data for design configuration. It will also include the core pressure boundary tube. Of primary interest is the flow distribution through each fuel element, the control rod cavity, and the by-pass between the core pressure boundary tube and the fuel element side plates.
1.1.13.3	Refueling components test facility—This facility will be used to test the ability to install an absorber on the fuel element, the tooling interface required to remove the fuel elements, and the lifting mandrel for the reflector vessel head.
1.1.13.4	Special test facility—A test facility is required to test the irradiation capsule disconnect assembly, closure elbow refurbishing assembly, the flush/purge/lock of the refueling assembly, and the PAR/servomanipulator to ensure proper operation during the shutdown work phase of the refueling cycle.

They include tests of seals, locking ring/bolt torque, outer shutdown rods, and the inner control rod latches. Descriptions of these tests are provided in the following paragraphs.

Seal test facility

The seal test facility will be used for two different functions during the life of the facility. First, the facility will be used to evaluate different seal configurations and/or seals from different vendors. Second, it will be used to verify the seal design chosen for the final design. This facility must be able to perform screening tests under conditions that simulate the actual operation and maintenance conditions expected in the reactor. The test will evaluate leakage characteristics, durability, special assembly techniques required, compatibility with remote maintenance philosophies, and leak detection methods.

The preliminary evaluation test will be performed using full scale seals with components to simulate full size hardware. The diameters of all the joints will be full scale. The facility will be designed to take into account axial thermal expansions of the full scale hardware. Initial tests most likely will be to evaluate different seal and flange configurations; therefore, the facility must be designed to accommodate different test pieces without major modifications. The differences in the test pieces may include differences in fastening methods, for example, bolted versus lock ring.

The candidate seal will be installed according to the manufacturer's specification onto the test hardware. The test hardware will be installed into the test fixture, which

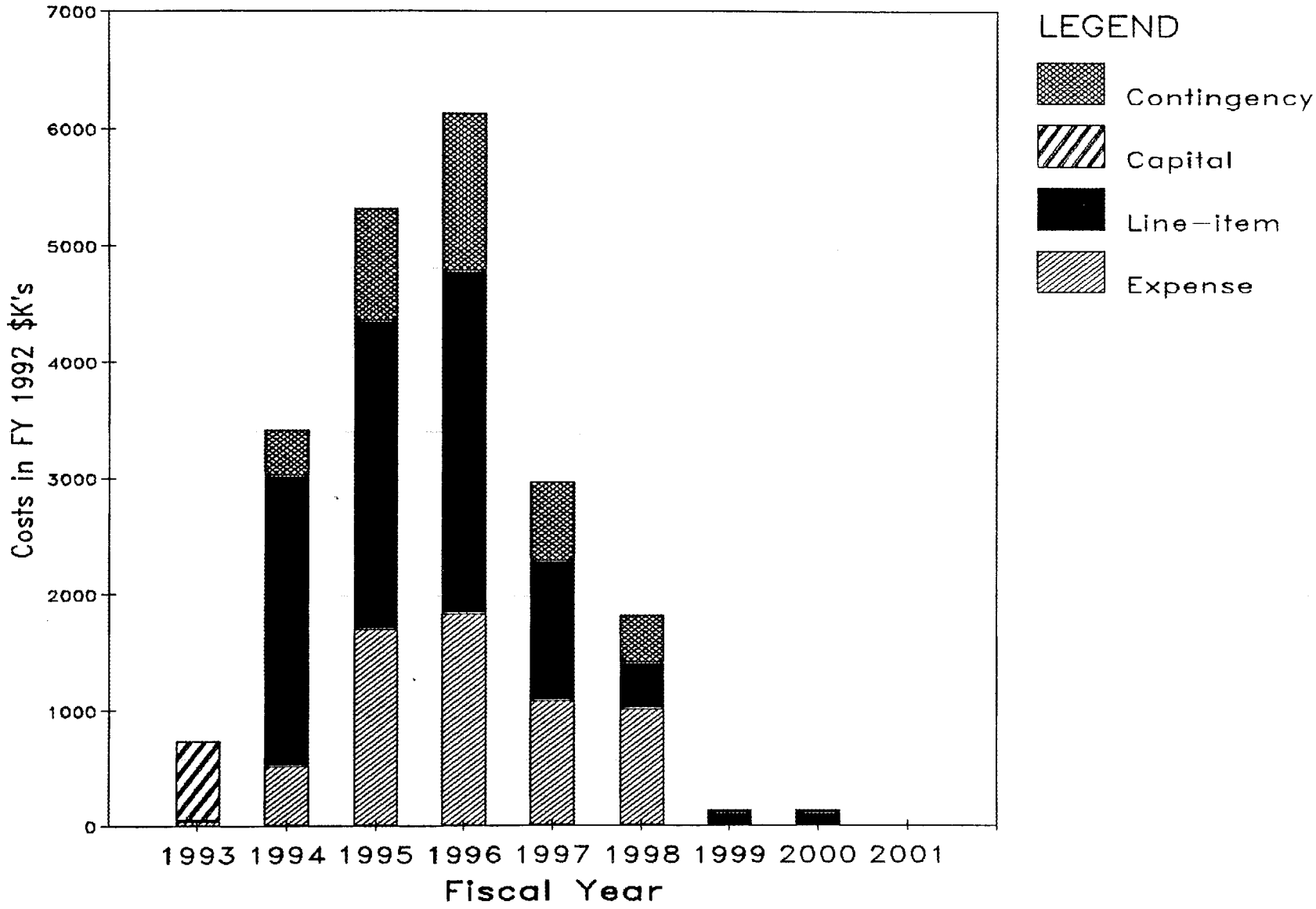

Fig. 3.25. Summary of cost estimates for work breakdown structure 1.1.13, reactor concepts.

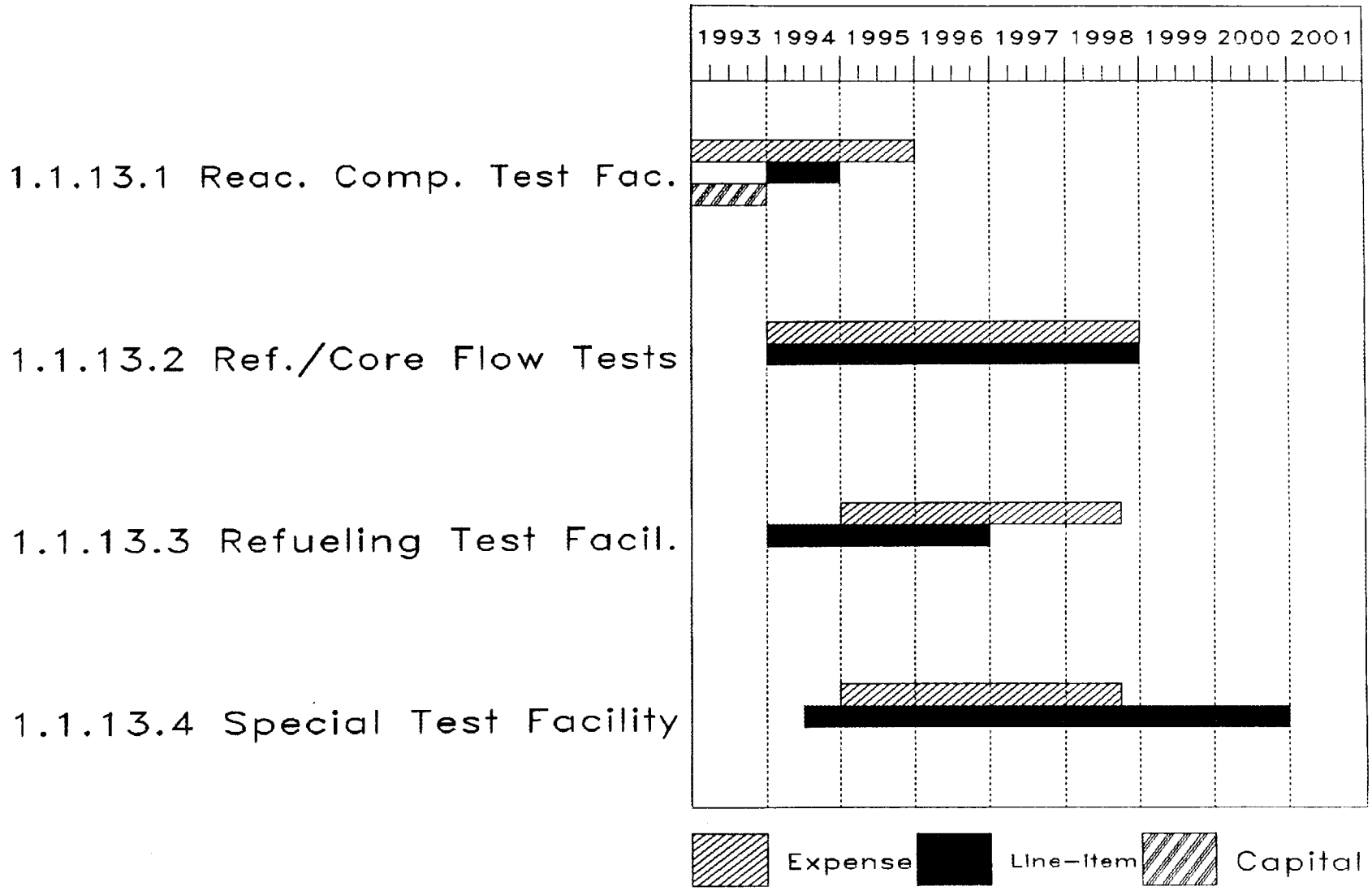

Fig. 3.26. Schedule for work breakdown structure 1.1.13, reactor concepts.

simulates the installation manipulator. The test hardware will be joined using forces and alignment conditions forecast for the reactor. The system will be pressurized with water to the operation pressure of the reactor, and the seals will be monitored for leaks. Initial tests may not include thermal cycling, but later tests will. Even though thermal cycling may not be a condition imposed on the early tests, any relative motion due to thermal expansion will be simulated and the effects monitored.

Types of joint configurations expected to be tested are bolted flange seals, radial seals, and labyrinth seals. Types of seals that could be evaluated for the bolted flange configuration are the Helicoflex 210 Series or the Helicoflex Double seal series similar to those used on the Orphee Reactor.

The design verification test will simulate as accurately as possible the actual operation conditions of the reactor for the design configuration selected as the final design. The parameters
monitored will be the leakage characteristics of new seals, seals that have been irradiated, seals that have been installed more than once, and irradiated seals that have been installed more than once. All of these tests will include pressure and temperature cycling, as well as relative motion cycling if that is expected to be part of the test.

Locking ring/bolt torque test

This test will simulate the installation of the CPBT using fastening techniques selected during advanced conceptual design. The CPBT will be installed into the test fixture, which simulates the installation manipulator. The test hardware will be joined using forces and alignment conditions forecast for the reactor.

Outer shutdown rod test

The development and testing program for the outer shutdown rod system will be performed in two phases. The first phase consists of testing and development of a single rod (full scale), and the second phase would use an eight-rod prototype system. Both phases would be performed with demineralized water substituted for D_2O and with the single rod tests preceding the eight rod tests.

Both phases of this program could be performed in the same vessel. However, the vessel that would be used for the eight-rod tests would be used for other test programs. Therefore, it may be desirable to use a separate vessel for the single-rod phase to save time.

Single-rod wet test stand. This facility would consist of one full-scale outer shutdown rod; a tank of sufficient size to accommodate the rod in the fully extended position, with connections for the water supply line and position indicator tubing; the necessary structure inside the tank to support the shutdown rod assembly; a water pumping system; minimal valving to operate the rod; a position indicator system; and appropriate instrumentation and controls.

As a minimum, the test facility would provide for significant development and testing of an outer shutdown rod and/or provide information on the following:

- the material suitable for the piston seal and lower guide;
- the scram spring;

- the shock absorber spring;
- the pressure required to withdraw the rod to its normal operating position as a function of the nominal diametral clearance at the piston seal and of the configuration of the lower guide;
- the flow rate to withdraw the rod with leakage only at the piston seal and lower guide;
- the configuration of the lower guide to provide adequate cooling water flow for the neutron absorber material;
- leakage past the piston seal and the lower guide as a function of pressure and the nominal diametral clearance with and without circumferential grooves in the piston seal;
- the practicality of using a hydraulic system for the rod position indicator, considering both flow and delta P; and
- the operability of the position indicator system as a function of the flow rate throughout the tubing.

Full scale eight-rod wet test stand. The facility would consist of a tank with a 1500 mm minimum diameter that will be used for the CPBT seal tests, a full-scale outer shutdown rod assembly, a support structure with holddown latches, plug-in inlet pipes, pilot-operated scram/pressure relief valves, a water supply pumping system, and valving and controls for both the rod operating and the position indicator system. The long pipe between the dry room and the reflector vessel should be simulated with coiled or folded pipe because the momentum of the water between the dry room and the relief valves is a factor in the scram time. The long lengths of position indicator tubing probably could be simulated with a properly designed restriction.

This would be a closed tank system, operating at the reflector tank's normal operating pressure and at any reasonably simulated upset pressure condition during which the shutdown system must function. If practical, it would be desirable to operate this facility at a temperature that would give the H_2O the same density as the D_2O at the normal operating temperature of the reflector vessel.

As a minimum, the test facility would provide for the development, testing, and demonstration of an outer shutdown rod system for the ANS by completing the following tasks:

1. provide for determining the time vs insertion distance as a function of the water pressure upstream of all leaks, but with a variable leak rate, and provide for selecting and testing a spring to accomplish the required insertion rate;

2. provide for determining the time to initiate movement of the rod after the scram signal occurs;

3. provide for determining the leak rate past the piston seal;

4. provide a means for determining the leakage at the seals of the manifold and delta P plug in units;

5. provide for determining the cooling water flow rate for the neutron absorber when in the withdrawn position;

6. provide for determining the suitability and modifications of the rod position indicator system, if required;

7. provide for determining the force exerted on the reflector vessel head and support structure during initiation of the scram;

8. provide for determining whether any unacceptable vibration occurs during normal operation (rods withdrawn) and with the rods inserted;

9. provide for selecting and testing a shock absorber spring or other means of slowing the rods near the end of travel during a scram;

10. provide for determining the suitability of the latches that retain the shutdown rod assembly on the support structure;

11. provide for assessing the scram impact effects of the rods on the manifold and hydraulic cylinders;

12. provide for investigating the effects of the varying flow that occurs inside the reflector tank on the fully extended rods; and

13. provide information on the behavior of the rods for the various scenarios of valve operating failures that could affect the shutdown capability of the system.

Inner control rod latch test

This test stand will be primarily for testing and developing the control latch and release mechanism, the accelerating springs, the magnets, and the switches that are required to inform the operators of the rod position. This will be a dry facility, whereas the rod, accelerating springs, latch, and shock absorber in the reactor will be submerged in water (D_2O). The behavior of these parts of the system will be somewhat different in the dry facility than in the reactor, but extensive useful information and experience can be gained with this relatively inexpensive facility. It will make the design of the wet test facility components very close to the design of those required in the final design, and it should result in a wet development and testing program that is considerably shorter and less expensive than would otherwise be required.

The test would include (1) the support stand structure, (2) mass to simulate control rod, (3) the control rod latch and scram mechanism, (4) a mock-up of the control rod lower end with accelerating spring, (5) the shock absorber, (6) the magnet with release and reset mechanism, (7) prototype seat and scram switches, (8) a mechanism for repeated cyclic operation, and (9) the instrumentation and controls necessary to operate the facility and collect data.

3.13.2 Reflector Vessel and Core Flow Test

3.13.2.1 Justification for the Reflector Vessel and Core Flow Test Task

Flow tests are needed to verify that the coolant flow behaves as was assumed in the thermal hydraulic analysis models for both the core and the reflector regions. The results of these tests could result in modifications to the designs of components in both the reflector and the fuel element regions.

3.13.2.2 Description of the Reflector Vessel and Core Flow Test Task

Tests under this task will be divided into two areas: those addressing flow issues in the reflector tank and those addressing flow issues inside the CPBT. The tests planned in each area are described in the following paragraphs.

The reflector vessel flow test subtask

This test is to determine flow patterns, flow velocities, and heat transfer characteristics within the reflector vessel. It will be run on a half-size scale model where the geometry and surface finish of the interior surfaces of the reflector vessel and the surface components inside the reflector vessel that are exposed to reflector vessel coolant will be accurately modeled. The surfaces of the components will be instrumented with temperature and pressure sensors to provide dynamic flow measurement of the cooling water. There will be provisions to allow flow mapping. The reflector vessel model will have view ports, and provision will be made for injecting dye or vapor bubbles for flow visualization testing.

Flow will be established in the reflector vessel that will be scalable to the operating conditions of the full-size reactor. The flow characteristic will be measured at locations of interest using the thermal and pressure sensors installed for this purpose. The temperature of the test fluid will be set to provide measures of heat transfer at the different locations. If necessary, dye or bubbles will be injected and high speed films made to determine flow patterns in critical areas.

The core pressure boundary tube region flow test subtask

This test is to determine the flow distribution of the cooling water through the core region of the reactor. Of primary interest is how much flow passes through each fuel element and how much is bypassed either through the core region around the control rods or through the region separating the upper fuel element from the outer CPBT wall. The wetted surfaces of the components in the interior of the CPBT will be modeled accurately for geometry and surface finish. The fuel elements will be modeled as a flow restriction. The exits from the fuel element region will be modeled to account for the flow characteristic around the irradiation sample assemblies, etc. The exit streams of coolant from the upper and the lower fuel element will be kept separate so that flow rate measurements can be made.

The wetted surfaces within the CPBT will be instrumented to provide data for determining the flow and heat transfer characteristics of the cooling water.

3.13.3 Refueling Components Test Facility

3.13.3.1 Justification for the Refueling Components Test Facility Task

The operation of the refueling components will have safety and potentially large availability impacts; therefore, they must be tested to provide performance confirmation. It is important that they be tested early in the design phase rather than late in the hardware fabrication phase. Failure to test early could have significant cost and schedule penalties.

3.13.3.2 Description of the Refueling Components Test Facility Task

This task will include an integrated facility to test the refueling system components that will operate in remote environments. The facility will be a dry facility with complete accessibility. The facility will be representative of actual equipment, but the test components may be fabricated without the extensive quality assurance and inspection that may be required for the actual components.

Three specific tests have been identified at this time: (1) the absorber/fuel test, (2) the tooling interface prototype test, and (3) the tool heat test. Separate descriptions of these tests follow.

The absorber/fuel test subtask

Testing of the absorber attachment to the fuel elements will be an important part of the test program. The interface between the fuel element and the absorber must be remotely connected, mechanically interlocked, and verifiable. The test will consist of four major tasks:

1. demonstrate the remote handling tooling that attaches the absorber to the tool interface,

2. demonstrate the function of the locking mechanism between the absorbers and the fuel elements,

3. demonstrate the integrity of the locking mechanism by subjecting the fuel element/absorber assembly to accident condition loading, and

4. demonstrate the verification process so that the computer system can identify that the fuel element is properly attached to the handling system.

The tooling interface prototype test subtask

The interface between the individual tool heads and the remote handling system must provide the required utilities and structure to allow the tool heads to function as intended. The interface will specifically provide the high pressure water, high pressure gas, electrical power, and instrumentation leads. This test will address the following aspects of the interface:

1. demonstrate that the interface has the proper number of utility resources available,
2. demonstrate that the interface has the structural integrity to carry the loads, and
3. demonstrate that the interface will operate in a remote D_2O environment.

The tool head test subtask

Typical tool heads will be tested in the refueling components test facility. Because there are several different tool heads, only the more complex tool heads will be tested in this series. The tool head test will include the following tasks:

1. demonstrate that the head will engage, lift, and disengage the components in a remote D_2O environment, and

2. test the structural integrity by loading the head with design basis loads and seismic loads.

3.13.4 Special Facilities

3.13.4.1 Justification for the Special Facilities Task

The special facilities task will provide early feedback to the design team on certain special facilities so that interferences and other problems can be eliminated before they cause delays in the project design and construction schedule.

3.13.4.2 Description of the Special Facilities Task

This task is not intended to be a single integrated test facility, but rather a series of related tests on some specific parts of the refueling process. These tests could be performed in an existing facility or in small special-purpose facilities. Four tests have been identified so far and are described in the following paragraphs:

The irradiation capsule disconnect test subtask

The functioning of the instrumented irradiation capsule in the core region depends on there being a reliable method to disconnect numerous capillary tubes and instrument leads. The disconnection must reduce the possibility that water might be introduced into the capillary tubes. The capsule must be capable of being remotely disconnected, and it will operate in a field of intense radiation. The irradiation capsule disconnect test will demonstrate that a gang of capillary tubes can be operated under those circumstances and that the component can be manipulated with remote tooling.

The closure elbow refurbish test subtask

The closure elbow must be removed and the seals refurbished during each fuel cycle. The closure elbow refurbish test will demonstrate that the process is reasonable in a conventional hot cell environment. The design of the seal retainer should be optimized with actual manipulator experience to reduce the time required for this process.

The flush/purge/lock test subtask

Transferring the spent fuel from the D_2O to the H_2O involves placing the fuel in a transfer lock and purging the lock with water. The flush/purge/lock test will measure the amount of D_2O/H_2O mix produced and determine the amount of tritium that will migrate across into the spent fuel pool. The analytic models that predict the amount of mixing will be verified.

The PAR/servo manipulator test subtask

The transfer cell in the refueling system will be equipped with several remotely operated manipulators, including a PAR 6000 and an advanced servo manipulator. This equipment will be checked for functionality in this test prior to its installation at the actual cell. It will be necessary to program the manipulators on site to optimize the process.

3.14 SAFETY RESEARCH AND DEVELOPMENT TESTS

The very high heat flux and coolant velocity of the ANS are beyond the range of available data for power reactors or even for other research reactors. The ANS core design is unique among reactors. Therefore, basic data are needed on thermal-hydraulic phenomena pertinent to design basis accident analysis and to severe accident analysis. The computer codes available for safety analysis, e.g. RELAP-5 for design basis accident analysis and MELCOR for severe accident analysis, were developed for power reactors. Test data are needed to develop input for these codes and to validate the performance of these codes for ANS conditions.

The tests described in Sect. 3.14.1 are designed to validate the design basis safety analysis of the reactor thermal margins during steady state and design basis accident conditions that do not involve fuel damage. The series of experiments described in Sect. 3.14.2 runs from power burst testing of ANS fuel mini-plates to define the important thresholds in fuel accident performance, to tests that examine specific severe accident phenomena necessary to predict severe accident source terms.

This WBS element contains two major project milestones:

1. Complete transient thermal-hydraulic testing by the end of September 1999. These tests play a key role in confirming the safety case and are to be completed at least a year before the completion of the FSAR.

2. Complete severe accident analyses testing by the end of September 1996. These tests play a key role in the development of ANS-specific computer models for severe accident analysis and should be completed early enough to be used in the code and model development effort.

The safety R&D tests activity is divided into two WBS level four tasks summarized in Table 3.14. Most of this work would be performed at ORNL and a number of subcontractors. The total estimated costs for this activity over the 9-year period covered by this R&D plan are given in Fig. 3.27, and the associated schedules are shown in Fig. 3.28. All of the analysis work and testing planned for the task are supported by expense money,

Table 3.14. Summary description of the safety research and development work breakdown structure level four tasks

WBS	Task description
1.1.14.1	Accident-related fuel testing and severe accident testing—This task includes experimental tests in the areas of explosive interactions, fission product release, fuel transient tests, fuel/concrete interactions, aerosol chemistry tests, and containment tests related to fission product release.
1.1.14.2	Transient thermal-hydraulic experiments in support of design—The activities in this task include experimental testing to verify and/or validate the computer models being used. Tests to obtain data bases of measurements for critical heat flux, flow instability, and boiling natural circulation also will be performed within this task.

and capital equipment money is used to construct test facilities. Subsections 3.14.1 and 3.14.2 provide more detailed information on the WBS level four tasks under this activity.

3.14.1 Thermal-Hydraulic Experiments in Support of Design and Safety Analysis

3.14.1.1 Justification of Thermal-Hydraulic Tests in Support of Safety Analysis

Several tests are expected to be performed as separate subtasks under the thermal-hydraulic tests task. Descriptions of each subtask are provided in the following paragraphs.

Flow blockage tests

Partial flow blockage of the fuel element by debris in the primary loop is an important event that would significantly impact the likelihood of fuel damage. Experiments are planned to establish the type of blockage necessary to initiate fuel damage.

Transient two-sided channel tests

The availability of critical heat flux and flow instability data at off-normal and transient conditions in the ANS reactor is limited. Data are required to validate correlations and models used under those conditions for the specific thermal-hydraulic conditions (e.g., flow rate, pressure, geometry) imposed by the ANS reactor design.

Transient prototypic span tests

The proper use of subchannel analysis must be well defined because it is fundamental to both the steady state and accident thermal limit calculational procedures. The spanwise and axial flux profiles of the ANS reactor will influence the pressure drop characteristics of the channel, which will in turn influence the position where flow excursion is initiated. Hot stripe and hot spot conditions must be simulated adequately in core thermal hydraulic analysis to predict core thermal limits correctly.

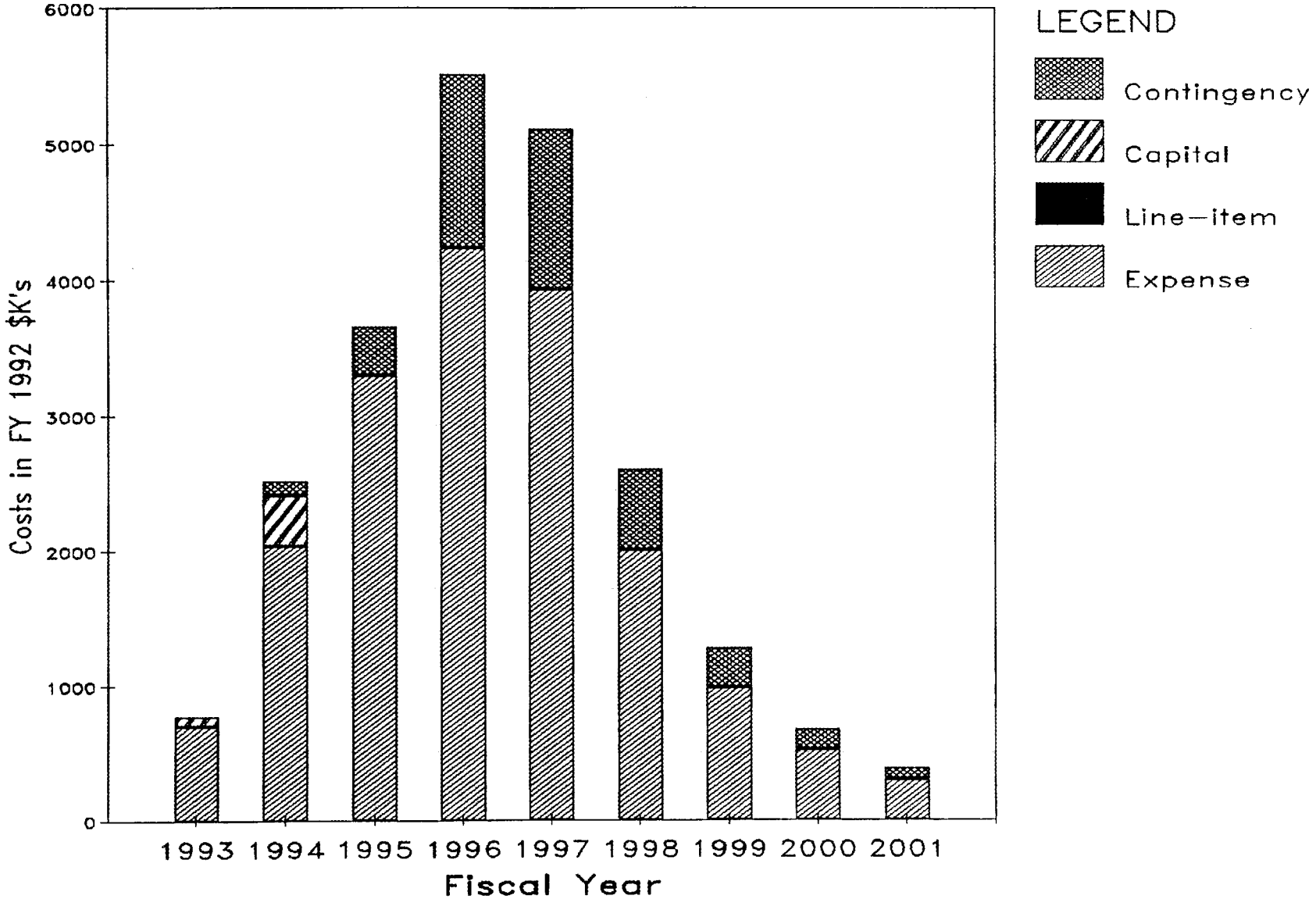

Fig. 3.27. Summary of cost estimates for work breakdown structure 1.1.14, safety research and development.

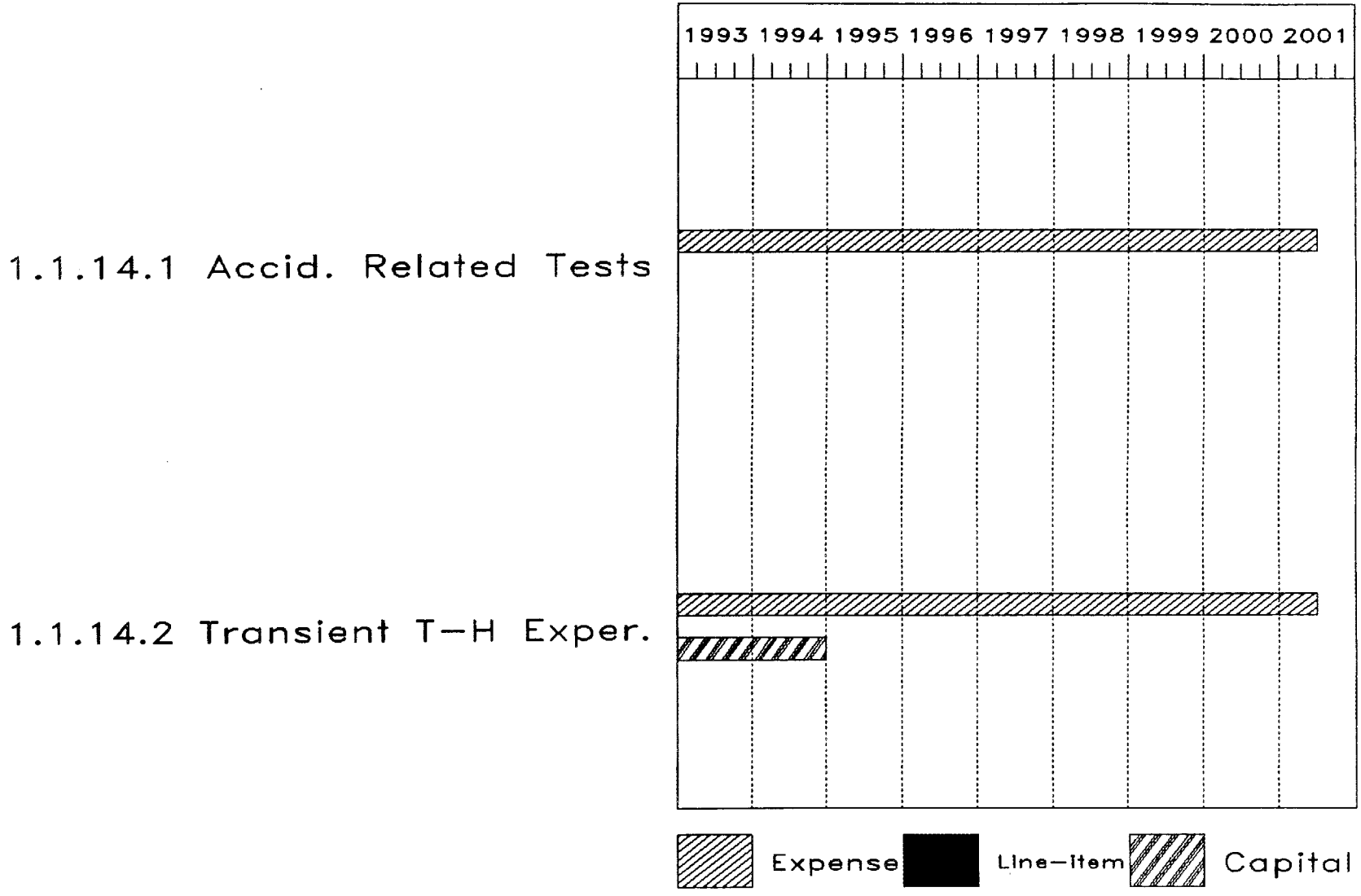

Fig. 3.28. Schedule for work breakdown structure 1.1.14, safety research and development.

Low mass flux—positive quality tests

Low pressure, low mass flux critical heat flux will occur at low heat flux values; the likely limiting thermal limit will be a local departure from nucleate boiling (DNB). Periodic DNB followed by rewetting is very likely to precede the prolonged DNB that would lead to overheating of the fuel. Boiling instabilities are expected to be affected by a number of parameters, including oxide on the aluminum cladding, thermal diffusivity and heat capacity of the fuel plate, channel length, and the heat flux and power-to-volume ratio in the channel. The flow conditions preceding critical heat flux in low mass flux, low pressure systems are rich in phenomena involving strong nonlinearities. Therefore, those flows are difficult to model computationally. No data prototypic of the ANS reactor exist for low mass flux critical heat flux where unsteady positive quality flows may be encountered. Prototypic data are needed to establish thermal limits in these situations.

Transient validation test loop

Transient thermal hydraulic codes that currently are being used to analyze the ANS reactor design have not been validated for conditions (e.g., subcooling levels, coolant velocities) typical of the ANS reactor. Experiments must be developed to validate those codes for safety applications. This experimentation must cover both core behavior and more global or system behavior.

Boiling natural circulation tests

One feature of the ANS reactor design is its capability to reject core heat under natural circulation conditions. Thermal limits such as critical heat flux, flow excursion, and incipient boiling will be used to determine the potential safe operation under off-normal natural circulation conditions. Very few data under these conditions are available, and experimentation will be necessary to develop an appropriate data base.

Hydraulic tests of nonfuel components

Many nonfuel components (e.g., control rods, beam tubes, CPBT, reflector tank walls) are exposed to very high heating rates and unknown flow conditions under both nominal and off-normal conditions. Experiments investigating the thermal and fluid behavior of these components will be necessary to characterize their performance and operating margins sufficiently.

Pipe break experiments

The consequences of a pipe break accident predicted using transient thermal hydraulic codes such as RELAP depend heavily upon the assumptions used in characterizing the break behavior (e.g., instantaneous opening vs timed opening, double-ended guillotine break vs leak before break arguments). Because the assumption of an instantaneous double-ended guillotine break in the piping system is unrealistic, and because this accident generally is predicted to cause fuel failure, it is necessary to characterize these breaks more realistically. Experimentation will be necessary to quantify break behavior sufficiently for application to safety analysis.

3.14.1.2 Description of Thermal-Hydraulic Experiments in Support of Design and Safety Analysis Task

Separate descriptions are provided in the following paragraphs for the various tests planned as separate subtasks.

Flow blockage tests

Experiments are planned to establish the type of blockage necessary to initiate fuel damage. A set of adiabatic experiments are planned to examine the flow field downstream of the blockage. These results will establish the point where net vapor would be generated in the heated channel. Heat transfer characteristics will be calculated from measured temperature distributions on a heater positioned downstream of the blockage. Data and corresponding analysis will be obtained over a range of blockage conditions to assess the minimum acceptable blockages.

Transient two-sided channel tests

Experiments are planned to measure critical heat flux and flow instability limits under off-normal and transient conditions for the ANS reactor. These will be performed in the thermal-hydraulics test loop and will include tests over a range of velocity, pressure, and subcooling levels anticipated for loss of cooling accidents.

Transient prototypic span tests. An experimental test section will be developed to determine the effect of increased span to gap ratio. In addition, this test section will be used to examine both hot spots and hot streaks that will exist in the ANS reactor core. These include both neutronic peaking and localized peaking due to fuel manufacturing defects. The test section design will use a heating element of nonprototypic material to simplify power requirements. Measurements will include test section pressure drop, local heater temperatures, inlet and exit fluid conditions, and power. Modeling of the test section will be done as necessary to ensure that local heat fluxes are accurately known. This test section likely will use the thermal-hydraulics test loop with modifications to increase flow capability and tailor the current voltage relationship to the new test section.

Low mass flux—positive quality tests. Experiments will be conducted to measure the instantaneous and time-averaged mass flux through the coolant channel under applied power and pressure drop conditions. These experiments will be conducted in the thermal-hydraulics test loop with a full-width test channel and will include simulation of the oxide layer on the aluminum wall. Time-average pressure drop and mass flux information from the tests will be coupled with the critical heat flux model to allow a lumped parameter simulation of the fuel assembly performance under the applied power and pressure drop situations.

Transient validation test loop

A scaled facility (1/5, 1/3, or other) will be used for integral testing to gather pressure and flow response (and other) data. These data will be used to validate codes such as RELAP and the Dynamic Model to ensure that those codes properly predict

transient system behavior. A test matrix will be developed based on examination of predicted ANS reactor transient response during accident conditions. A model of the experiment then will be developed and transient experiments simulated to validate each of the codes.

Additional transient testing will focus on the test section behavior during accident-like conditions. Those tests probably will be run in the thermal-hydraulics test loop, using test sections that are not intended to simulate all behavior of the ANS reactor fuel plates (conditions such as heat capacity and piping lengths will not be accurately simulated). The test sections will be modeled and the tests will be used only to benchmark the codes' predictive ability.

Boiling natural circulation tests

A scaled facility will be designed (or an existing one used) which incorporates prototypic heights and loss characteristics to simulate the reactor system properly under natural circulation conditions. This facility will be used to study the fuel thermal behavior under off-normal natural circulation conditions. Potential areas of study include single phase heat transfer coefficients, the point of incipient boiling, flow excursion, and critical heat flux behavior. This experiment will use the test apparatus developed under WBS 1.1.4.

Hydraulic tests of nonfuel components

The facility or facilities for hydraulic tests of nonfuel components will be designed to investigate the significant thermal aspects of each component. The experimental design will be driven by thermal analysis of each component; experiments will be performed only as necessary and only on the phenomenon or component area of interest. Since much of the analysis is in the infant stage, details of these experiments have yet to be formulated.

Pipe break experiments

Pipe break experiments will be used to examine the behavior of piping under realistic load conditions (e.g., earthquake, pressure). The experiments will use existing facilities where possible to characterize such phenomena as the type of loading required to run a preexisting crack, the crack loss characteristics, the correlation between the number of earthquake loads and the crack length, and the speed of crack propagation. Piping used in these experiments will be prototypic of ANS reactor piping, and experimental conditions (e.g., pressure, earthquake loads) will be typical of ANS reactor operation. These experiments will interface with the transient thermal hydraulic analysis task to provide useful data for use in thermal hydraulic predictions.

3.14.2 Accident-Related Fuel Testing and Severe Accident Tests

3.14.2.1 Justification for the Accident-Related Fuel Testing and Severe Accident Testing Task

The overall justification for severe-accident-related testing is improving the accuracy of calculations performed to predict the consequences of such accidents. The proposed

severe accident tests are divided into categories that address similar data needs. The justification of each of these categories of tests is given in the following paragraphs.

Justification for the core melt progression tests

The core melt progression tests are needed to benchmark the computer models that predict how the debris from a core melt accident might respond to the various gravity, flow, viscous, and other forces as it relocates from the immediate region of the core. The mode and timing of debris transport control risk-determining phenomena such as debris coolability, recriticality, and steam explosion (fuel-coolant interaction).

1. Transient fuel performance tests are needed to identify the amount of thermal energy the ANS fuel can absorb without damage and to identify the energy thresholds for the various stages of damage, including cracking, blistering, fragmentation, and metal-water reaction.

2. Fuel damage propagation tests are needed to benchmark computer models to predict the fraction of the fuel that would be involved in an accident initiated by local fuel damage. Such damage could be caused by, for example, a localized core inlet flow blockage. Preliminary ANS preliminary risk assessment results show that flow blockage will be the dominant severe accident contributor.

3. The debris dispersion tests are needed to determine the effect of fluid surface shear stress on the dispersion of the melting fuel or clad. The dispersion tendency of overheated fuel may be an important indicator of the likelihood that the fuel will reach a dispersed configuration suitable for recriticality of the fuel. Recriticality is a concern in severe accidents because it may initiate explosion of the debris.

4. The debris coolability tests are needed to benchmark the computer codes used to determine whether the debris reaches a thermally stable or near-stable configuration at any intermediate stage in its progression. This determination would be an important addition to the knowledge of accident sequence timing and would have implications for worker risk.

Justification for the fission product release and chemistry tests

The fission product release and chemistry tests are needed to benchmark the computer models that will be used to estimate the release and transport of fission products after an accident. The risk due to severe accidents is due largely to the release of fission products from damaged fuel, the escape of the fission products from the coolant system, and the transport of the fission products within the containment building. The chemical and physical properties of the fuel and the fission products are important determinants of the release and transport properties.

1. Tests to determine release of fission products from fuel in steam are needed because most of the reliable data currently available are for release in air. Due to the large amounts of water in the ANS design, it is highly likely that melting and fission product release would take place in a water or steam environment.

2. The fission product chemistry and mobility tests will provide information on the chemical species released from damaged fuel. The chemical state of a fission product can greatly affect its volatility and transport.

Justification for the debris-structure interaction testing

The debris-structure interaction tests are needed to determine the extent of debris transport and containment after a core melt accident, because these phenomena can have a significant effect on worker risk.

1. Debris-structure eutectic formation experiments are needed to identify the relevant solid solutions that can form between or among the debris and structural constituents. The formation of a eutectic can have a significant effect on core melt progression and structural retention because the melting point of a eutectic differs from that of either of the constituents. The melting point of the eutectic may be significantly lower than that of the pure form of the component with the highest melting point.

2. Debris-containment experiments will be needed if the ANS severe accident analysis work results in the adoption of a design feature to help contain the debris. In that case, it would be necessary to obtain the data necessary to benchmark the computer codes that are used to show that design feature will work as intended.

3. Molten core concrete interaction experiments will be needed to benchmark the computer codes used to estimate the generation of degradation products in severe accident sequences in which ANS fuel debris contacts concrete. Even though the ANS has a relatively small core, scoping calculations have shown that the debris from a severe accident can cause significant degradation of the concrete floor if it stays in a coherent melt and is transported downward into the subpile room. Concrete degradation can generate gases that would greatly affect the generation of fission-product-bearing aerosols.

Justification for the explosive phenomena tests

The explosion-related tests are needed to assess the propensity of the ANS fuel to explode after melting of the reactor core. Whether explosions occur is a significant index of severe accident risk.

1. Fuel coolant interaction triggering and energetics experiments are needed for estimating the propensity for ANS fuel to undergo explosive reactions following a rapid melting of the core and subsequent contact with subcooled water. A greater knowledge of the explosive tendency of molten ANS fuel will be necessary to quantify adequately the severe accident-related risk for the ANS.

2. Fuel coolant interaction prevention testing will be needed if it is determined that specific prophylactic design features should be incorporated to prevent explosive interaction between fuel debris and coolant.

Justification for the containment thermal-hydraulics testing

In accidents that involve partial degradation of the reactor building primary containment barrier, thermal-hydraulic phenomena in the annulus between the primary containment and the secondary containment may significantly affect the quantity of fission products released to the environment. These phenomena should therefore be studied experimentally.

3.14.2.2 Description of the Accident-Related Fuel Testing and Severe Accident Testing Task

The proposed severe accident tests are divided into categories that address similar data needs. Descriptions of these subtask categories of tests are provided separately in the following paragraphs. Note that most of these tests are still in the planning stage; therefore, detailed descriptions cannot be provided in every case.

The core melt progression tests

The core melt progression tests are needed to benchmark the computer models that predict how a core melt accident might proceed. Four individual tests have been identified.

1. Transient fuel performance tests are designed to identify the amount of thermal energy the ANS fuel can absorb without damage and to identify the energy thresholds for the various stages of damage, including cracking, blistering, fragmentation, and metal-water reaction. Fuel mini-plates of prototypic fuel and clad composition and thickness (except that the uranium is 20% instead of 93% enriched) will be enclosed in water-filled test capsules that will be exposed to power bursts of varying intensity in the Nuclear Safety Research Reactor in Japan. The power bursts are conducted to enable the researchers to know and control the number of fission reactions that occur in the mini-plates and hence the amount of energy imparted to the mini-plate within a short time. After a cooldown period, the experiment capsule is opened, and the miniplates are examined and photographed to assess the degree and quality of damage associated with the integrated energy input.

2. Flow blockage propagation tests are to benchmark computer models to predict the fraction of the fuel that would be involved in a fuel damage accident

initiated by core inlet flow blockage. These will be scaled experiments and probably will use simulant materials (i.e., materials with a lower melting temperature and probably without uranium) to minimize the expense associated with the test. The major phenomenon that the tests will be designed to explore is the propagation of fuel melting from a locally damaged spot into the adjacent undamaged parts of the core.

3. The debris dispersion tests will gather data on the effect of fluid surface shear stress on the dispersion of the melting fuel or clad. The major phenomenon of interest is how easily an overheated or melting fuel plate can be dispersed into droplets of fuel and/or cladding by a flowing medium. Because the latter is basically a separate effects test, the fuel material will be made using depleted uranium.

4. The debris coolability tests will provide benchmark data for computer codes used to determine whether the debris reaches a thermally stable or near stable configuration at any intermediate stage in its progression. The test will involve an inductively heated melt of debris on a structural material, with and without a water covering. Temperatures will be recorded, along with the degree of attack on the structural material. Depleted uranium will be used to manufacture the simulated fuel material.

Description of the fission product release and chemistry tests

The fission product release and chemistry tests will provide basic data on fission product behavior for use in the computer models used to estimate the release and transport of fission products after an accident.

1. The tests to measure the release of fission products from ANS fuel in a steam atmosphere will involve laboratory scale heating of irradiated uranium silicide fuel material in an apparatus equipped to measure the release of fission products as a function of the temperature of the sample. These experiments will be similar to experiments previously performed at ORNL on other types of fuel.

2. The fission product chemistry and mobility tests will provide information on the chemical species released from damaged fuel. These are laboratory scale experiments designed to be performed in conjunction with the fission product tests in steam.

Description of the debris-structure interaction tests

The debris-structure interaction tests will be designed to provide basic data on the debris transport and containment after a core melt accident.

1. Debris-structure eutectic formation experiments are designed to identify solid solutions that can form between molten ANS fuel and structural materials. They may be as simple as heating to a sufficiently high temperature some ANS depleted uranium silicide fuel that is mixed with aluminum and that is in

intimate contact with stainless steel or another appropriate structural material. Metallographic examinations after cooldown and the observation of properties at the elevated temperature will provide the necessary knowledge of whether and how much eutectic material forms during the period of heating.

2. Debris-containment experiments will provide the data necessary to benchmark the computer codes that calculate the effect of ANS fuel material on the structural features incorporated specifically to retard or eliminate melt progression through structural material. These tests, if performed, will use heated, depleted uranium silicide fuel material and will be of intermediate scale. They will be done as needed to ensure applicable data.

3. Molten core concrete interaction experiments will provide benchmark date for the computer codes used to estimate the generation of degradation products in severe accident sequences in which the ANS fuel debris contacts concrete. These tests will involve depleted uranium silicide/aluminum fuel material heated to the molten stage and brought into contact with the appropriate type of concrete. The tests will be of intermediate scale, which is fairly small because the ANS core is very small.

Description of the explosive phenomena tests

The explosion-related tests provide basic data on the propensity of the ANS fuel to explode following a large-scale melting of the reactor core.

1. Fuel coolant interaction triggering and energetics experiments are currently being conducted at the University of Wisconsin. The first series of experiments involves small samples of molten depleted uranium silicide/aluminum fuel material that contact water. Various initial debris temperatures are used to explore the effect of temperature on explosivity. An underwater electrical spark device is used to create a shock wave to trigger the explosions in cases when there is no spontaneous explosion. The control in this experiment is aluminum without fuel admixture; this control will be valuable because experiments have already been conducted (elsewhere) on the explosivity of much larger scale melts of aluminum metal. The first series of experiments provides comparative data on the explosivity of various fuel mixes at various initial melt temperatures. The second series will use a shock tube apparatus to examine the energy conversion efficiency of various fuel mixtures and initial temperatures.

2. Fuel coolant interaction prevention testing will be needed if it is determined that specific prophylactic design features should be incorporated to prevent explosive interaction between fuel debris and coolant. The scale will be intermediate, or as large as necessary to benchmark the computer codes that will be used to assess the efficacy of the prophylactic features.

Description of the containment thermal-hydraulics test

The containment thermal-hydraulics tests will involve a scale modeling of the annulus between the primary and secondary containment buildings. The test will examine the thermal stratification of hotter gases in the annulus.

REFERENCES

1. *Plant Design Reqirements*, ORNL/TM-11625/R4, Oak Ridge National Laboratory, July 1990.

2. G. L. Copeland, et al. *Advanced Neutron Source Final Preconceptual Reference Core Design*, ORNL/TM-11234, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, 1989.

3. R. M. Harrington, et al., *Conceptual Safety Analysis Report*, ORNL/ANS/INT-33, Martin Marietta Energy Systems, Oak Ridge National Laboratory, June 1992.

4. W. D. Rummel, et. al., *The Detection of Fatigue Cracks by Nondestructive Testing Methods*, National Aeronautics and Space Administration CR-2369, February 1974.

5. W. K. Stromquist and O. Sisman, *High Flux Reactor Fuel Assemblies Vibration and Water Flow*, ORNL-50, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, June 1948.

6. W. K. Sartory, *Analysis of Hydraulic Instability of ANS Involute Fuel Plates*, ORNL/TM-11580, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, November 1991.

7. D. R. Miller, "Critical Flow Velocities for Collapse of Reactor Parallel-Plate Fuel Assemblies," *J. of Eng. Power*, Trans ASME, 83-95, April 1960.

8. *ABAQUS User's Manual*, Version 4.8, Hibbitt, Karson, and Sorensen, Inc., Pantucket, RI, 1989.

9. *ADINA, A Finite Element Program for Automatic Dynamic Incremental Nonlinear Analysis*, ADINA Engineering, Inc., December 1984.

INTERNAL DISTRIBUTION

1. C. W. Alexander
2. D. J. Alexander
3. R. R. Allen
4. E. E. Alston
5. J. L. Anderson
6. B. R. Appleton
7. R. E. Battle
8. R. S. Booth
9. W. W. Bowman
10. R. A. Brown
11. G. J. Bunick
12-16. J. H. Campbell
17. P. F. Cento
18. N. C. J. Chen
19. K. K. Chipley
20. J. E. Cleaves
21. J. T. Cleveland
22. G. L. Copeland
23. J. R. Dixon
24. F. F. Dryer
25. D. K. Felde
26. J. C. Gehin
27. M. L. Gildner
28. H. A. Glovier
29. R. M. Harrington
30. J. B. Hayter
31. M. M. Houser
32. D. T. Ingersoll
33-36. R. L. Johnson
37. M. Kaminaga
38. R. A. Lillie
39. A. T. Lucas
40. J. A. March-Leuba
41. B. S. Maxon
42. G. T. Mays
43. S. V. McGrath
44. T. J. McManamy
45. G. R. McNutt
46. B. H. Montgomery
47. R. M. Moon
48. D. G. Morris
49. R. E. Pawel
50. H. R. Payne
51. F. J. Peretz
52. C. C. Queen
53. S. Raman
54. C. T. Ramsey
55. J. P. Renier
56. J. B. Roberto
57. D. L. Selby
58. H. B. Shapira
59. M. Siman-Tov
60. B. R. Smith
61. R. P. Taleyarkhan
62. D. W. Thiesen
63. S. R. Tompkins
64. P. B. Thompson
65. K R. Thoms
66. B. D. Warnick
67. C. D. West
68. J. L. Westbrook
69. D. M. Williams
70. B. A. Worley
71. G. T. Yahr
72. G. L. Yoder
73. ORNL Patent Office
74. Central Research Library
75. Y-12 Technical Library
76. Document Reference Section
77. Laboratory Records
78. Laboratory Records (RC)

EXTERNAL DISTRIBUTION

79. R. Awan, U.S. Department of Energy, NE-473, Washington, DC 20585.
80. C. L. Christen, DRS/Hundley Kling Gmitter, FEDC, MS-8218, P.O. Box 2009, Oak Ridge, TN 37831-8218.

81. R. A. Edlund, U.S. Department of Energy, Oak Ridge Field Office, FEDC, MS-8218, P.O. Box 2009, Oak Ridge, TN 37831-8218.
82. R. R. Fullwood, Brookhaven National Laboratory, Upton, NY 11973.
83. W. R. Gambill, Route 5, Box 220, Clinton, TN 37716.
84. B. Gupta, AECL Technologies, 1155 Metcalfe Street, 2nd Floor, Montreal, Quebec, Canada H3B 2V6.
85. A. F. Henry, Professor, Department of Nuclear Engineering, Massachusetts Institute of Technology, 77 Massachusetts Avenue, Cambridge, MA 02139.
86. R. A. Hunter, Director, Office of Facilities, Fuel Cycle, and Test programs, Nuclear Energy Division, U.S. Department of Energy, NE-47, Washington, DC 20585.
87. L. C. Ianniello, Acting Associate Director, Office of Basic Energy Sciences, Office of Energy Research, U.S. Department of Energy, ER-10, Washington, DC 20585.
88. T. L. Kerlin, University of Tennesseee, College of Engineering, 315 Pasqua Engineering Buklding, Knoxville, TN 37996-2300.
89. J. A. Lake, Manager, Nuclear Engineering and Reactor Design, Idaho National Engineering Laboratory, P.O. Box 1625, Idaho Falls, ID 83415.
90. W. F. Manning, U.S. Department of Energy, Oak Ridge Field Office, FEDC, MS-8218, P.O. Box 2009, Oak Ridge, TN 37831-8218.
91. W. E. Meek, Project Manager, Gilbert/Commonwealth, Inc., P.O. Box 1498, Reading, PA 19603.
92. W. T. Oosterhuis, Materials Sciences Division, Office of Basic Energy Sciences, Office of Energy Research, U.S. Department of Energy, ER-132, Washington, DC 20585.
93. H. Preble, Research and Test Reactor Fuel Elements, Babcock and Wilcox Co., P.O. Box 785, Lynchburg, VA 24505.
94. J. M. Ryskamp, Idaho National Engineering Laboratory, P.O. Box 1625, Idaho Falls, ID 83415.
95. J. L. Snelgrove, Coordinator, Engineering Applications, RERTR Program, Argonne National Laboratory, 9700 South Cass Avenue, Argonne, IL 60439.
96. I. Thomas, Director, Materials Science Division, Office of Energy Research, U.S. Department of Energy, ER-13, Washington, D.C. 20585.
97. J. M. Warren, Gilbert/Commonwealth, Inc., 1055 Commerce Park Derive, Suite 200, Oak Ridge, TN 37380.
98. D. K. Wilfert, Energy Programs, U.S. Department of Energy, Oak Ridge Field Office, FEDC, MS-8218, P.O. Box 2009, Oak Ridge, TN 37831-8218.
99. Products and Chemicals, Inc., 7201 Hamilton Boulevard, Allentown, PA 18195-1501.
100. Office of Assistant Manager, Energy Research and Development, P.O. Box 2001, Oak Ridge, TN 37831-8600.
101-112. Office of Scientific and Technical Information, P.O. Box 62, Oak Ridge, TN 37831.