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OAK RIDGE NATIONAL LABORATORY

MARTIN MARIETTA

The Advanced Neutron Source Research and Development Plan

Principal Author
D. L. Selby

August 1995

Advanced Neutron Source

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

ORNL/TM-12249/R1

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ACRONYMS

1-D	one-dimensional
2-D	two-dimensional
3-D	three-dimensional
AECL	Atomic Energy of Canada, Ltd.
ALARA	as low as reasonably achievable
AMIS	Advanced Neutron Source Materials Information System
ANS	Advanced Neutron Source
ANSR	ANS reactor
ASIC	application-specific integrated circuit
ASME	American Society of Mechanical Engineers
ATR	advanced test reactor
BFNA	break frequency noise analysis
BNL	Brookhaven National Laboratory
BOL	beginning of life
BPR	by-pass ratio
CBCF	carbon-bonded carbon-fiber
CECE	combined electrolysis and catalytic exchange
CETF	control element test facility
CFD	computational fluid dynamics
CHF	critical heat flux
CPBT	core pressure boundary tube
CSAR	conceptual safety analysis report
DAS	data acquisition system
DNB	departure from nucleate boiling
DOE	U.S. Department of Energy
EOL	end of life
ES&H	environmental, safety, and health
FBTF	flow blockage test facility
FE	flow excursion
FIV	flow-induced vibration
FSAR	final safety analysis report
HFBR	High Flux Beam Reactor
HFIR	High Flux Isotope Reactor
HWUDF	heavy water upgrade and detritiation facility
IKRD	inverse kinetics rod drop
ILL	Institut Laue-Langevin
INEL	Idaho National Engineering Laboratory
ISOL	isotope separation on line
LMES	Lockheed Martin Energy Systems, Inc.
LSBT	large slant beam tube
M&C	Metals and Ceramics
MIT	Massachusetts Institute of Technology
MSM	modified source multiplication
NECC	National Electrical Carbon Corporation
NIST	National Institute of Standards and Technology
NPSH	net pump suction head

Nu	Nusselt
OFI	onset of flow instability
ORNL	Oak Ridge National Laboratory
PCDAS	plant control and data acquisition system
PCSA	primary coolant supply adaptor
Pe	Peclet
PSAR	preliminary safety analysis report
QA	quality assurance
R&D	research and development
REDC	Radiochemical Engineering Development Center
RPS	reactor protection system
S&Z	Saha and Zuber
SANS	shallow angle neutron scattering
SAR	safety analysis report
SDD	system design description
St	Stanton
THTL	thermal-hydraulic test loop
V&V	verification and validation
W&F	Whittle and Forgan
WBS	work breakdown structure

FOREWORD

As the draft version of this report was completed, the U.S. Department of Energy (DOE) announced that the President's budget did not contain funding for continuation of the Advanced Neutron Source (ANS) Project in the next fiscal year, FY 1996. The primary reason for this was the high price tag of the project at a time when budgets were being reduced. The need for a new neutron source, however, is still high, and other options to the ANS are expected to be pursued. In the meantime, we have decided to publish this report without revisions, because it provides a summary of the issues that would have been addressed by the ANS research and development (R&D) program, and it provides documentation of the status of the individual R&D activities.

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 $>7 \cdot 10^{19} \cdot m^{-2} \cdot s^{-1}$. 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.

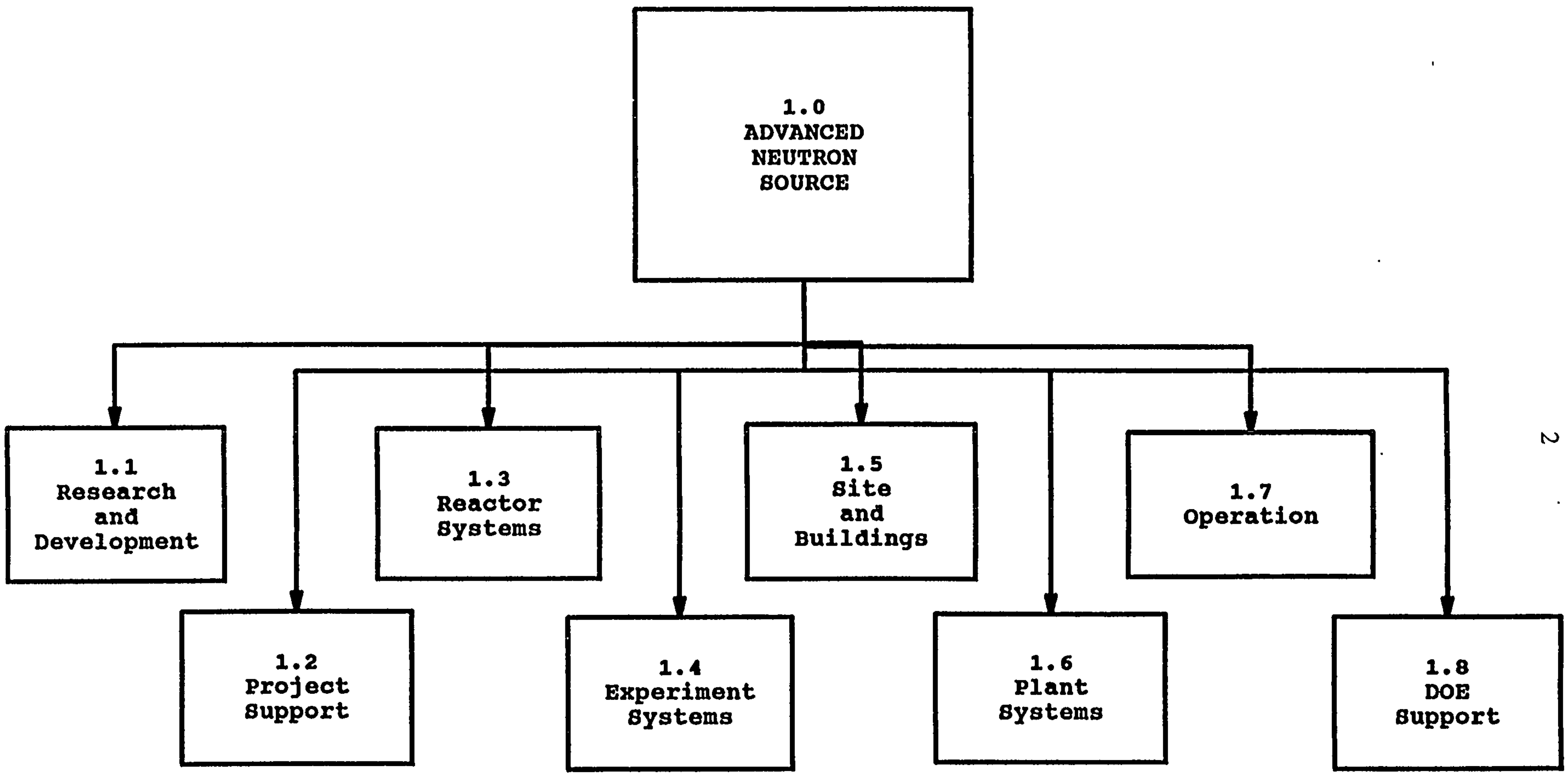

Fig. 1.1. Advanced Neutron Source level two work breakdown structure.

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

2. R&D 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 R&D PROGRAM WBS LEVEL THREE TASKS

The R&D tasks have been grouped into the 13 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 reactor facility concepts (WBS 1.1.13). A summary of the 13 tasks and their WBS level four subtasks is provided in Sect. 3 of this report.

2.2 R&D 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 (1) to design and fabricate test facilities at ORNL that are not considered to be direct prototype tests and (2) to 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 2003, 80% of the work identified within the R&D program is expected to be completed by the end of FY 1999, and more than 90% by the end of FY 2000. Only five activities have significant efforts beyond FY 2000:

1. The reactor core development activity extends into FY 2003 to provide the reactor physics and thermal-hydraulic analyses and documentation necessary to support the FSAR.

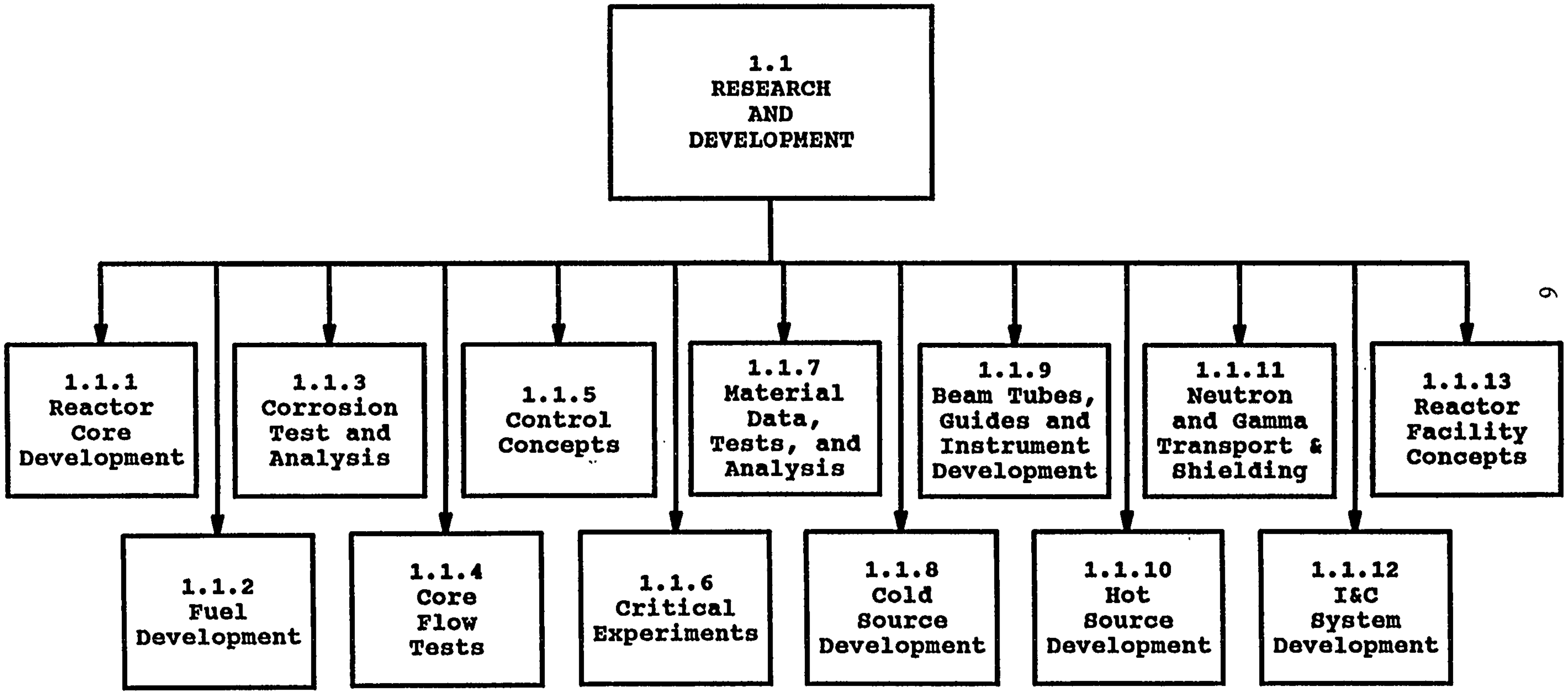

Fig. 2.1. Advanced Neutron Source research and development program level three work breakdown structure.

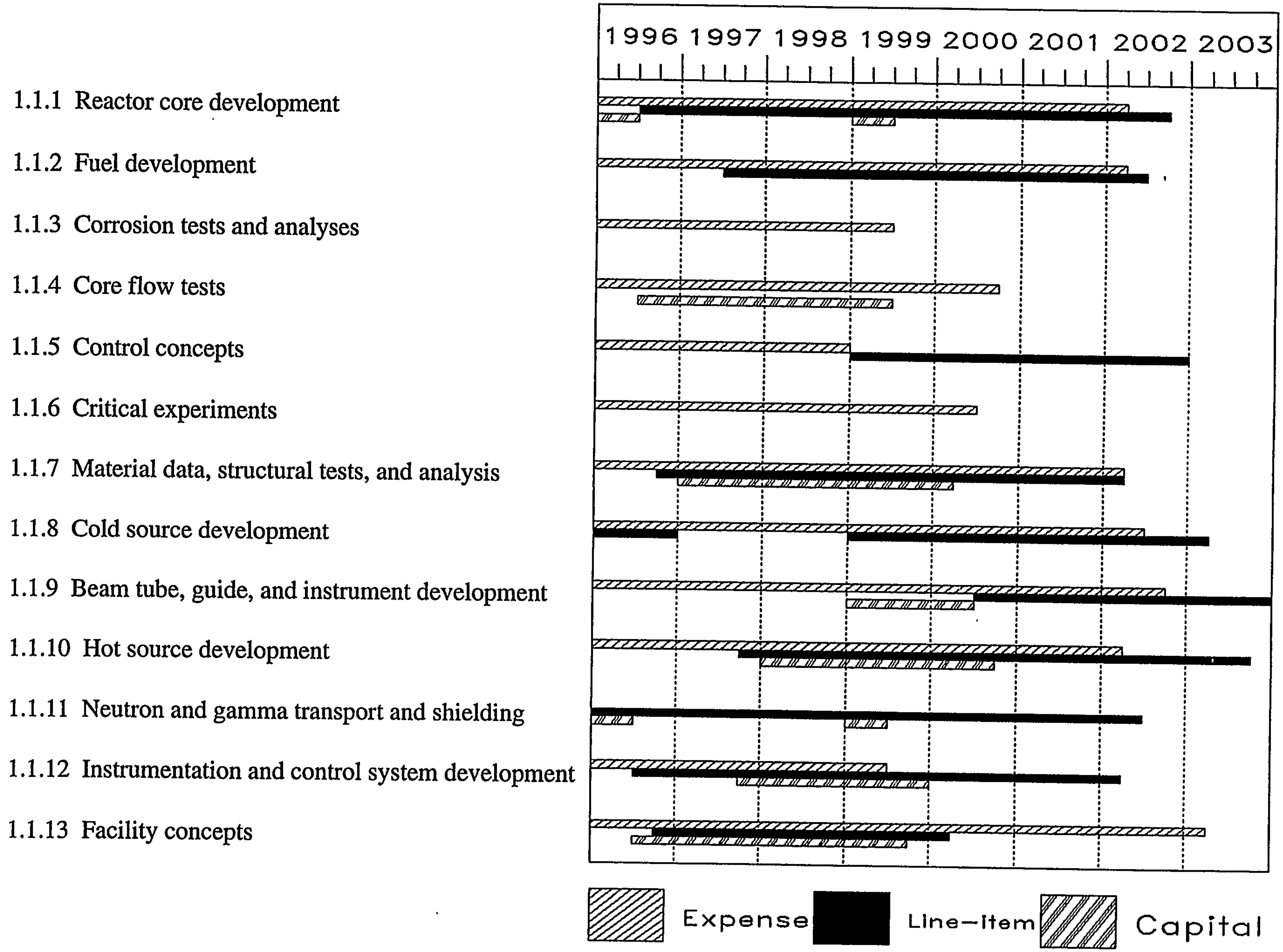

Fig. 2.2. Schedule for the ANS research and development program.

Table 2.1. ANS R&D program cost estimates in FY 1994 dollars

WBS Level 3	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.1	Reactor Core Development	Exp.	1400	1391	1328	1064	734	390	95	0	6402
		Line	1305	2401	2108	2324	2200	1542	1017	0	12897
		Cap.	180	0	0	250	0	0	0	0	430
1.1.2	Fuel Development	Exp.	2181	3576	2730	1213	147	106	53	0	10006
		Line	0	1010	2210	2235	660	660	660	0	7435
1.1.3	Corrosion Tests And Analyses	Exp.	669	594	355	120	0	0	0	0	1738
1.1.4	Core Flow Tests	Exp.	2416	2092	3647	5786	3390	0	0	0	17331
		Cap.	760	6859	6903	2080	0	0	0	0	16602
1.1.5	Control Concepts	Exp.	106	122	122	0	0	0	0	0	350
		Line	0	0	0	62	62	62	96	0	282
1.1.6	Critical Experiments	Exp.	260	2053	1770	2152	1276	0	0	0	7511
1.1.7	Material Data, Structural Test, And Analysis	Exp.	3495	6088	3490	1843	312	218	32	0	15478
		Line	165	481	1041	1041	714	210	50	0	3702
		Cap.	1000	923	799	574	153	0	0	0	3449
1.1.8	Cold Source Development	Exp.	843	1905	1540	907	255	165	94	0	5709
		Line	153	0	0	438	580	705	767	115	2758
1.1.9	Beam Tube, Guide, and Instrument Development	Exp.	862	1615	2891	3085	2190	821	399	0	11863
		Line	0	0	0	0	289	632	652	615	2188
		Cap.	0	0	0	617	271	0	0	0	888
1.1.10	Hot Source Development	Exp.	825	850	642	306	185	121	85	0	3014
		Line	0	113	520	623	318	178	239	176	2167
		Cap.	0	0	1072	1362	977	0	0	0	3411
1.1.11	Neutron and Gamma Transport and Shielding	Line	1000	1037	1027	1016	927	563	373	0	5943
		Cap.	46	0	0	46	0	0	0	0	92
1.1.12	Instrumentation and Control System Development	Exp.	539	620	279	228	0	0	0	0	1666
		Line	147	205	272	186	160	160	78	0	1208
		Cap.	0	81	325	271	0	0	0	0	677
1.1.13	Facility Concepts	Exp.	3710	2780	1613	1049	1574	1268	1159	253	13406
		Line	311	2035	2038	1747	270	0	0	0	6401
		Cap.	223	39	438	216	0	0	0	0	916
	Subtotals	Exp.	17306	23686	20407	17753	10063	3089	1917	253	94474
		Line	3081	7282	9216	9672	6180	4712	3932	906	44981
		Cap.	2209	7902	9537	5416	1401	0	0	0	26465
	Contingency	Exp.	1020	2369	2041	1776	1007	310	192	25	8740
		Line	154	1216	1843	1934	1236	942	786	181	8292
		Cap.	299	1581	1908	1032	280	0	0	0	5100
	Grand Totals	Exp.	18326	26055	22448	19529	11070	3399	2109	278	103214
		Line	3235	8498	11059	11606	7416	5654	4718	1087	53273
		Cap.	2508	9483	11445	6448	1681	0	0	0	31565

2. The fuel development activity extends into FY 2003 to provide the fuel element production mode fabrication methodology and procedures.
3. The cold source development activity extends into 2003 to complete testing and to prepare a cold source safety analysis report (SAR).
4. The beam tube, guide, and instrument development activity extends into FY 2003 to supply the final specifications and testing of the instruments before installation in the ANS facility.
5. The facility concepts activity extends into 2003 to finish testing of special components such as servo manipulators.

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 50% to as low as 5%; the average is about 13%. 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 R&D 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 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 R&D 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. Because of 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

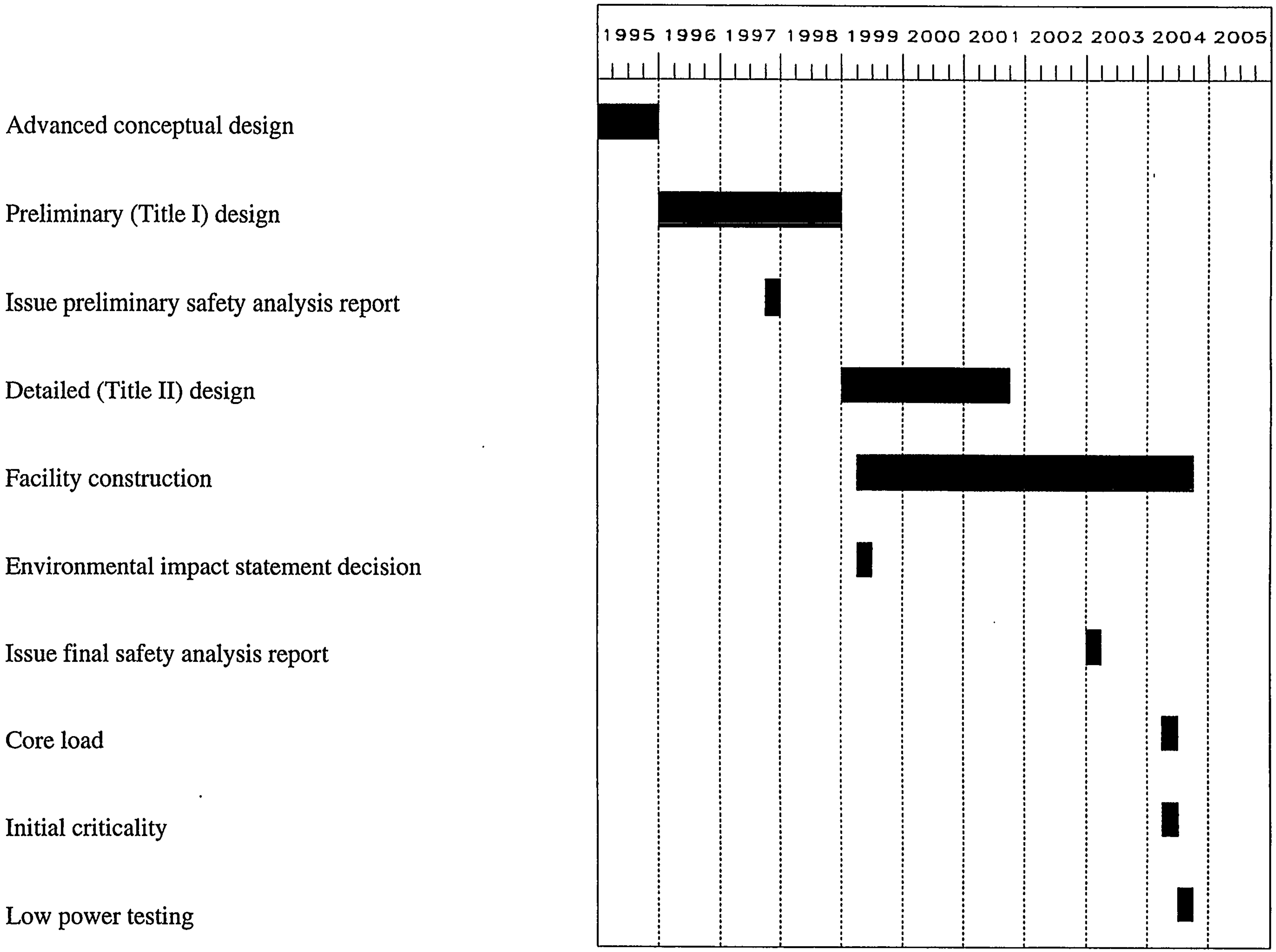

Fig. 2.3. Advanced Neutron Source Project planning schedule summary.

Table 2.2. R&D program test 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
1.1.4	Thermal-hydraulic tests	In progress
1.1.4	Natural circulation tests	In planning
1.1.4	Flow blockage tests	In progress
1.1.4	Full span tests	In planning
1.1.4	Low mass flux-DNB[a] tests	In planning
1.1.4	Hydraulic tests of nonfuel components	Future
1.1.4	Integral transient tests	In planning
1.1.4	Refueling thermal-hydraulic testing	Future
1.1.4	Full flow hydraulic 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	In planning
1.1.7	Fuel element and control element vibration tests	In planning
1.1.7	Material irradiation tests	In progress
1.1.8	Two-phase cryogenic thermal-hydraulic tests	Completed
1.1.8	LH2/LD2[b] pump tests	In progress
1.1.8	Thimble assembly test	Future
1.1.8	Cryogenic fill test	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

Table 2.2 (continued)

Work breakdown structure	Plan testing activity	Status
1.1.13	Vessel seal tests	In planning
1.1.13	Locking ring/bolt torque tests	In planning
1.1.13	Latch component wear 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
1.1.13	CECE[c] pilot plant test facility	Future
1.1.13	Latch release magnet tests	Completed

[a]DNB = Departure from nucleate boiling.
[b]LH2/LD2 =Liquid hydrogen/liquid deuterium.
[c]CECE = Combined electrolysis and catalytic exchange.

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). Therefore, it is important to monitor the schedule of these tasks carefully to minimize the potential for R&D task impact on the project's critical path.

2.5 R&D 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 emphasizes 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 R&D Activity

Thirteen 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.

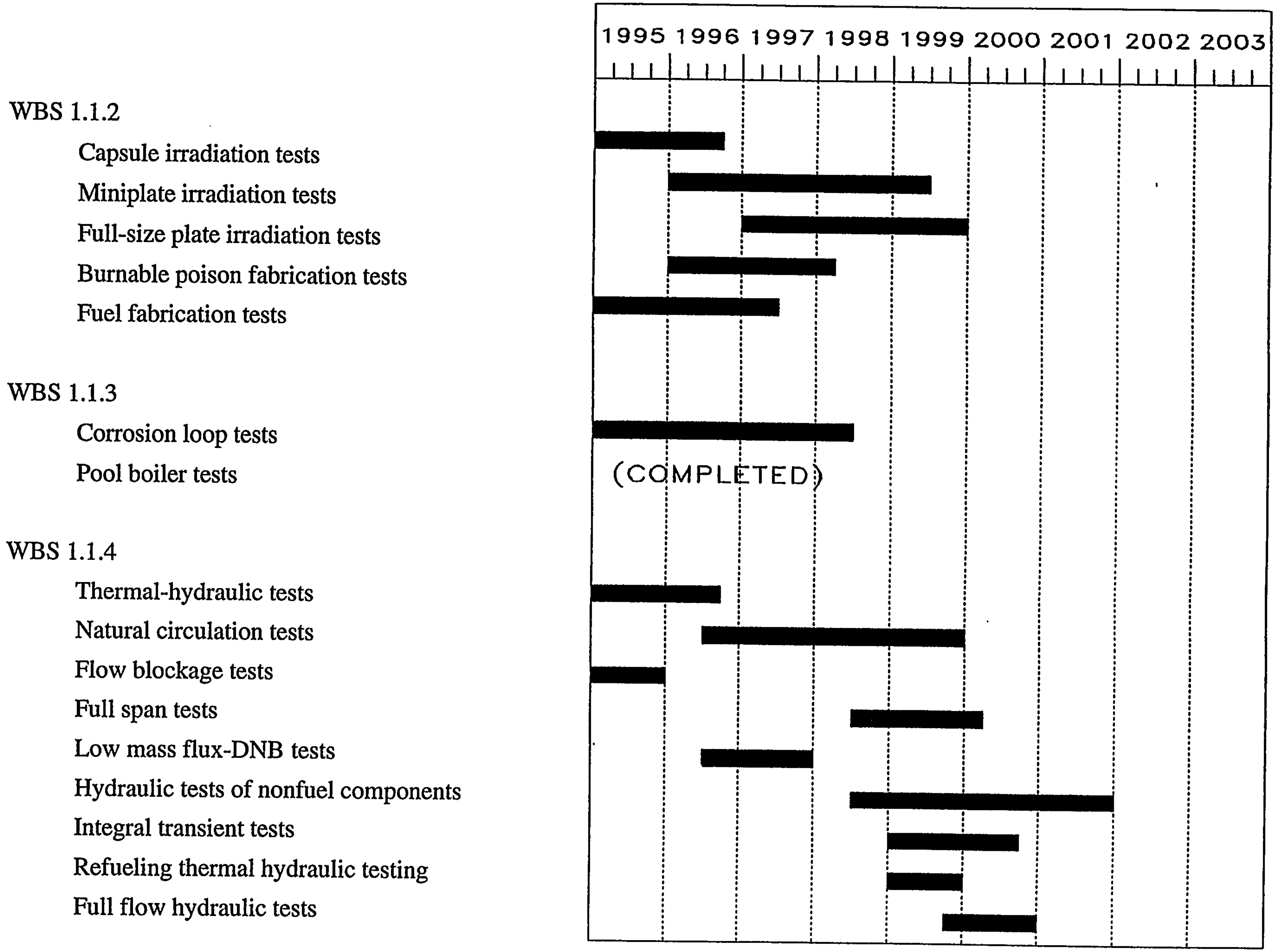

Fig. 2.4. Schedule for R&D testing activities.

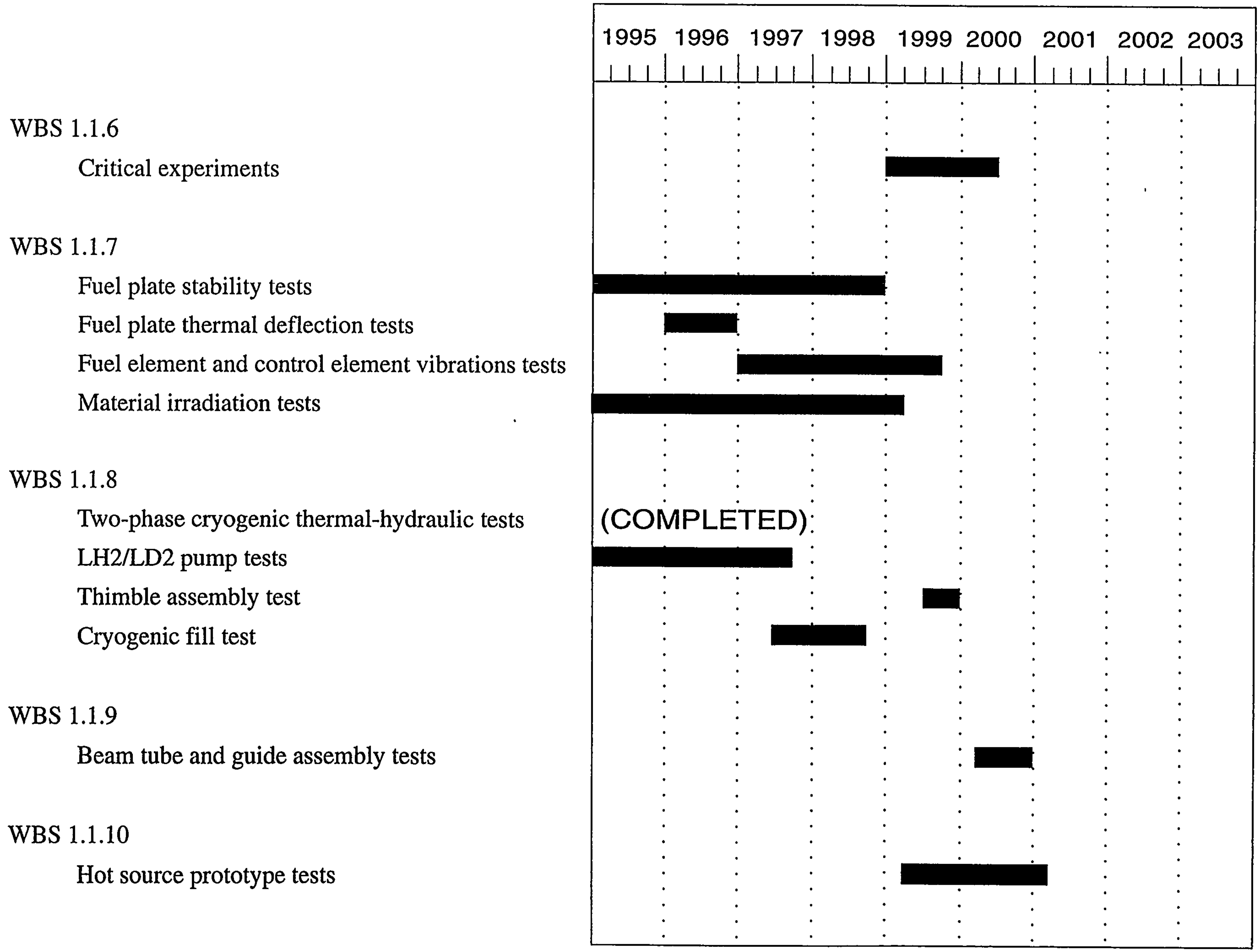

Fig. 2.4 (continued)

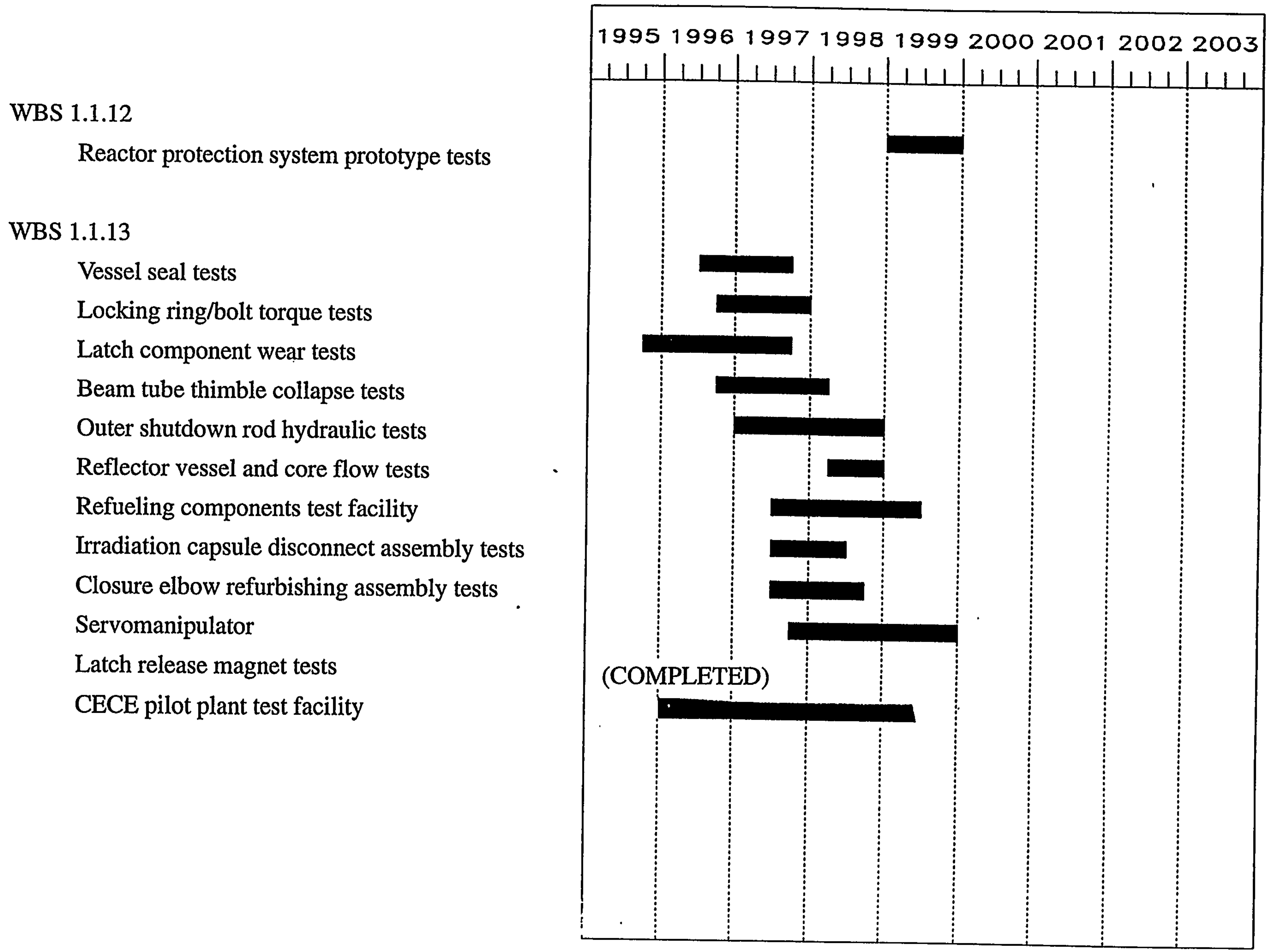

Fig. 2.4 (continued)

Table 2.3 R&D program critical milestone events

	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	4/98		7/98	7/98		
	Final safety analysis report support analyses complete	1/02	1/03				10/02
1.1.2	Fuel development						
	Fuel design specification complete	6/96		9/97			10/02
	Fuel performance report complete	6/00	9/02				
1.1.3	Corrosion loop						
	Water chemistry requirements defined	9/96				9/98	
1.1.4	Core flow tests						
	Transient thermal-hydraulic testing complete	9/00	9/02				
1.1.6	Critical experiments						
	Critical experiments complete	6/00	9/02				
1.1.7	Materials data, tests, and analyses						
	Core pressure boundary tube fracture tests complete	6/98		9/98			
	Materials properties data base complete	9/97		12/97	12/97		
	Component vibration tests complete	9/99		12/99			
1.1.8	Cold source development						
	Cold source prototype test complete	1/01			6/01		
1.1.9	Beam tube, guide, and instrument development						
	Beam tube assembly prototype test complete	3/99			9/99		
1.1.10	Hot source development						
	Hot source prototype test complete	9/00			6/01		
1.1.12	Instrumentation and controls system development						
	Reactor protection system prototype testing complete	6/99		1/00			
1.1.13	Reactor facility concepts						
	Reactor component evaluation tests completed	12/99		6/00			

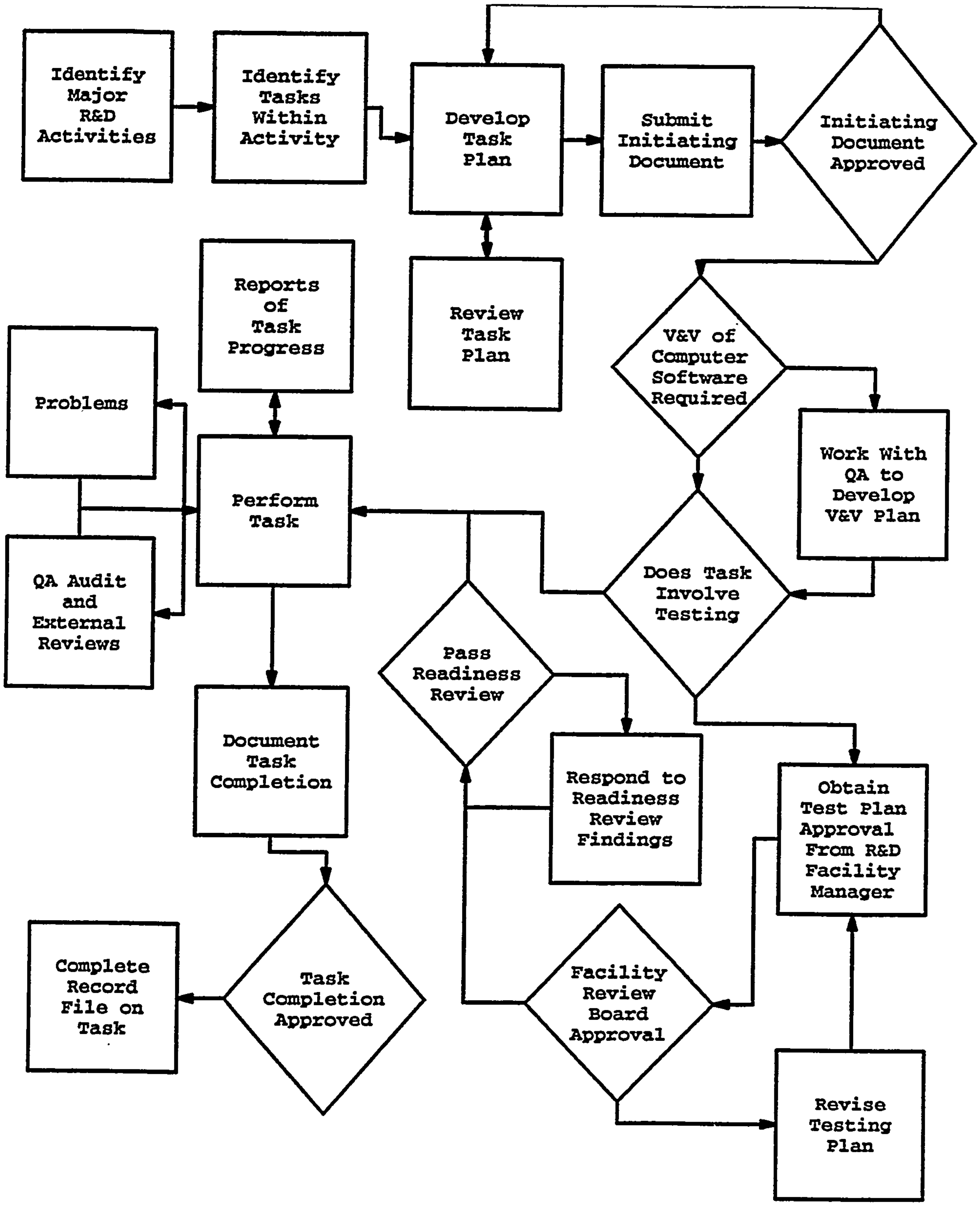

Fig. 2.5. R&D program task flow chart.

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.

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. 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 (Fig. 2.6) 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 including: 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 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 from 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.

Advanced Neutron Source
TASK INITIATING DOCUMENT

WBS Number: ______________ Task Title: ____

____

Task Leader: ________________________ Other Key People: ________________________

Estimated Task Schedule:

BEGINNING DATE: ____________________ TASK DURATION: __________________________

Task Description:

Deliverables (Provide specific items to be calculated or measured):

Interfaces with Other Tasks:

Potential Problems (Include factors that could impact the cost or schedule):

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Fig. 2.6. Task initiating document.

OVER

Reference Computer Codes, Procedures, Standards, Data Sets and/or Experimental Facilities and How They Are To Be Used Within the Task:

Key Decision Points Within the Task:

Comments:

Submitted by: ______________________________ Date: ______________

Approved by: ______________________________ Date: ______________
Task Leader

Project Approval: ______________________________ Date: ______________
Task Manager

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Fig. 2.6 (continued)

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 before 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) issues have been considered, 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, or 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's QA staff 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 R&D 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 assessing R&D activities.

2.6.1 Applicability of Quality Assurance to the R&D Program

R&D activities for the ANS Project are required to advance the current state of certain technical knowledge into the operating realm of the ANS. Since the output of this R&D will directly affect the design and safety of ANS structures, systems, components, and operating envelope, the results must be fully supported through accepted quality assurance practices.

The ANS Project is committed to meeting the requirements set forth by U.S. Department of Energy (DOE) Order 5700.6C.[1] As part of the implementation of these requirements, the project uses the national consensus standard of American Society of Mechanical Engineers (ASME) NQA-1, *Quality Assurance Program Requirements for Nuclear Facilities*,[2] to describe and define the management controls and practices 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*.[3]

2.6.2 Management of Quality Assurance Within the R&D Program

The implementation of the management control system 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 specific 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 foci of these training/indoctrination sessions are knowledge of working procedures and a working knowledge of NQA-1, project terminology, and organizational 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 available skill mix to arrive at the appropriate research team for a particular task.

Nonconformances and reportable occurrences are documented and handled by the originating ORNL R&D organizations, using their respective programs 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 R&D 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 in accordance with ANS Project procedures 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 on 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 the ORNL division where the work is performed and to 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 attributes required to ensure that the test apparatus and test specimen are in accordance with their designs for the performance of the test.

2.6.4 R&D Assessments

The R&D program's achievement of quality is measured through two types of assessment activities: (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 of 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 the customer. Independent assessments are, in part, organized external reviews/workshops and ORNL, LMES, and DOE audits and surveillances. The results of these assessments establish a baseline of acceptable performance. The corrective measures and actions contribute to the ANS Project's quality improvement program.

3. DESCRIPTIONS OF THE R&D 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,[4] 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 is required. This R&D activity provides for the development of a basic ANS reactor (ANSR) 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 1998. 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 January 2002. Under the present schedule, the FSAR is to be issued by January 2003. 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 summarized in Table 3.1. As indicated in that table, this activity includes the tasks associated with defining the reactor core concept, developing the process for treatment of uncertainties, and performing 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 8-year period covered by this R&D plan are given in Table 3.2, and the associated schedules are shown in Fig. 3.1. 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-hydraulic 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 provides the data necessary to define a reference core to start the conceptual core design. It includes 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.[5]
1.1.1.3	General design criteria development support—This task identifies and documents 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 provides 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 includes 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-hydraulic 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.

Table 3.2. WBS level four breakdown of costs for reactor core development activity

WBS Level 3	Level 4	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.1		Reactor Core Development										
	1.1.1.1	Methods Development	Exp.	1273	1198	1143	926	640	296	95		5571
			Line	144	153	180	443	790	320	250		2280
			Cap.									0
	1.1.1.2	Preconceptual core development	Exp.									0
			Line									0
			Cap.									0
	1.1.1.3	General design criteria development support	Exp.	127	193	185	138	94	94			831
			Line				46	101	95	95		337
			Cap.									0
	1.1.1.4	Reactor physics support to design	Exp.									0
			Line	459	973	944	925	630	485	330		4746
			Cap.	100			100					200
	1.1.1.5	Thermal-hydraulics support to design	Exp.									0
			Line	702	1275	984	910	679	642	342		5534
			Cap.	80			150					230
		Subtotals	Exp.	1400	1391	1328	1064	734	390	95	0	6402
			Line	1305	2401	2108	2324	2200	1542	1017	0	12897
			Cap.	180	0	0	250	0	0	0	0	430
		Contingency	Exp.	70	139	133	106	73	39	10		570
			Line	65	240	422	465	440	308	203		2143
			Cap.	0	0	0	0	0	0	0	0	0
		Totals	Exp.	1470	1530	1461	1170	807	429	105	0	6972
			Line	1370	2641	2530	2789	2640	1850	1220	0	15040
			Cap.	180	0	0	250	0	0	0	0	430

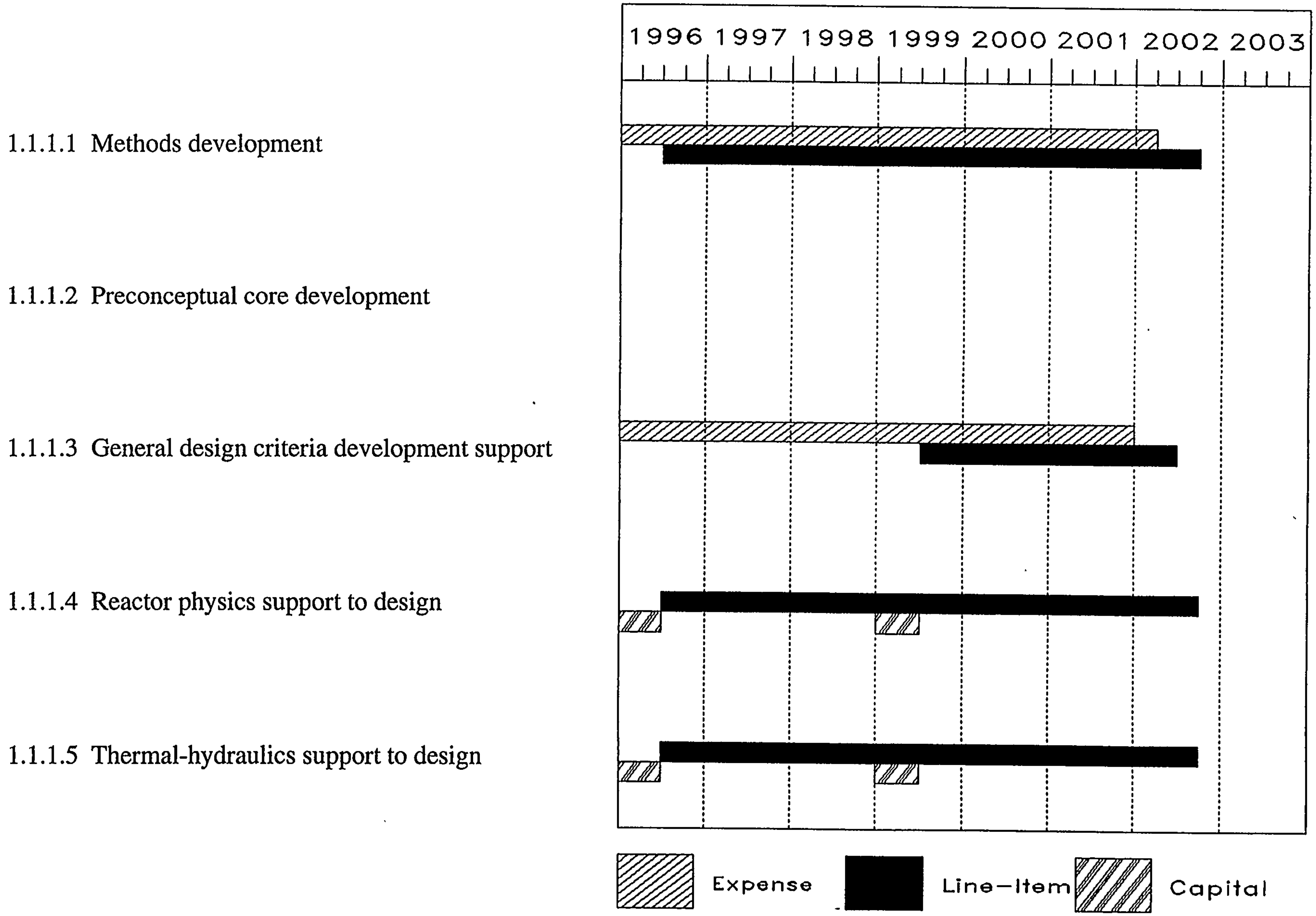


Fig. 3.1. Schedule for WBS 1.1.1 reactor core development.

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 start-up tests. Therefore, the development of the ANSR 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 selecting cross-sections (energy treatment, spatial dependence, and burn dependence); evaluating transport vs diffusion methods; and identifying the approach for treating fuel burnup and fission product buildup. For the thermal-hydraulic analyses, this subtask covers issues related to correlation and code selection and development, identification of the physical property data to be used in analyses, development of appropriate phenomenological data bases, determination of the method for treatment of uncertainties, identification of methodology for treatment of fuel defects, and examination of one-dimensional (1-D) vs two-dimensional (2-D) vs three-dimensional (3-D) heat transfer and fluid mechanics treatments. In addition, methods to interface with the structural analysis task will be developed within this task to evaluate both the fluid mechanics and thermal impact on the core structural design.
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 ANSR 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 ANSR 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 and computational models by comparisons with measured data. The data needed for validation are being identified, and experiments are being developed (WBS 1.1.4) 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 ANSR start-up test program that will provide a final fine tuning of the analytical models.

3.1.1.2.1 Reactor physics

Under this subtask, the determination, development, and improvement of the most appropriate methods necessary to meet calculational requirements and accuracy is determined. Currently, a few-group, 2-D, finite-difference diffusion theory model using the VENTURE[6] code system is used for the fuel cycle analysis. An MCNP[7] Monte Carlo model that includes 3-D representations of the reactor core and experimental systems is used for detailed beginning-of-cycle modeling. Development areas include the incorporation of a transport theory method (using the DORT[8] S_N transport code) into fuel cycle calculations and a Monte Carlo depletion strategy using MCNP, ORIGEN2[9] and MOCUP.[10] In addition, improvements to existing methods to treat photoneutron production, isotope production, and hafnium control depletion are being developed.

Methods have been defined for selecting and processing the nuclear cross-sections required as input for the nuclear analysis codes. For the MCNP analysis, continuous energy cross-sections are required and are obtained using the NJOY[11] cross-section processing system. Future MCNP cross-section development will include fission product and activation cross-sections to be used in the Monte Carlo depletion strategy. Codes that use an energy group approach (VENTURE and DORT) require the selection of the appropriate number of energy groups and the treatment of spatial and exposure-dependent effects. The cross-sections for these group codes are obtained from the ANSL-V master cross-section library[12] previously developed under this subtask. The AMPX[13] and SCALE[14] systems are used to process the master library cross-sections into working library form required by VENTURE and DORT. Future development in the group cross-section area is further determination of spatial and exposure effects on the cross-sections and the development of multidimensional cross-section collapsing methodologies.

Development of models begins with gathering and interpreting design drawings and sketches and determining the necessary level of detail required for the neutronics and thermal-hydraulics models. The MCNP modeling tasks have involved translating design drawing information into 3-D geometrical surface input parameters and selecting output tally specifications. The 2-D r-z analysis performed with VENTURE and DORT has required homogenizing 3-D structures and determining appropriate mesh and material zone structure. Future development areas include determining the impact of homogenous vs heterogeneous fuel plate models in MCNP; developing 2-D r-z representations of the central control rods, reflector components, and target/production rods for the VENTURE and DORT models; and further developing the fuel depletion and fission product buildup models.

Presently, calculational analysis of the FOEHN critical experiments have been completed[15] with the MCNP Monte Carlo code and are nearly completed for the VENTURE and DORT deterministic codes.[16] Additional validation of the different analysis methods and models has been obtained by intercomparisons such as the comparison of the 2-D few-group VENTURE and DORT calculations with 3-D continuous-energy MCNP calculations.

3.1.1.2.2 Thermal hydraulics

The thermal-hydraulic portions of this task are generally broken into major phases corresponding to Title I design, Title II design, and the three major project safety milestones: the conceptual SAR (CSAR), the preliminary SAR (PSAR), and the FSAR. The work in each phase is similar; however, more detailed methodology development and V&V efforts are necessary as the project progresses through the phases.

Thus far, thermal-hydraulic core design has used a steady-state thermal-hydraulic code originally developed to support the High Flux Isotope Reactor (HFIR). This code was modified to calculate the maximum core powers possible without exceeding user-selected thermal limits: $T_{wall} = T_{sat}$, incipient boiling, critical heat flux (CHF), flow excursion (FE), maximum fuel centerline temperature, and oxide spallation. The code has been extensively revised to incorporate ANS-specific design features. Thermal-hydraulic correlations within the code have been modified to incorporate the most recent and accurate prediction methodologies. This includes the incorporation of an ANS-specific oxide film growth correlation developed under WBS 1.1.3. In the future, an ANS-specific steady-state thermal-hydraulic code will be developed to take advantage of more recent computational capabilities and provide a more logical platform to perform design and safety calculations. A code requirements document has been developed to help guide this task.

Methodologies have been developed within this task to evaluate localized fuel defects. This has involved selecting appropriate codes, formulating models, and identifying critical parameters. Local heat flux peaking and fuel temperatures calculated using these codes are used as input to the steady-state thermal-hydraulic calculations and used to supplement those calculations by providing detailed temperature information near fuel defects.

A heavy water and light water properties package has been assembled and is being used throughout the project.[17] This package presents correlations for both physical and thermodynamic properties that were developed from the Atomic Energy of Canada, Ltd. (AECL) and ASME tables. An initial selection of thermal-hydraulic correlations has also been made and is being used project-wide.[18] This package includes correlations for single-phase friction factor, heat transfer coefficient, incipient boiling limit, CHF limit, and FE limits. The correlation package will be updated periodically as new data and correlations become available.

A thermal-hydraulic data base has been assembled both to select appropriate correlations and to identify the uncertainties associated with the correlations. This data base is a living document, and data are added as they become available. The data base contain data for all of the thermal-hydraulic phenomena identified in the previous paragraph. Uncertainties and uncertainty distributions have been determined for all of the above correlations. These are used as one portion of the input required to perform statistical uncertainty analysis.

Effort has also focused on developing both the methodologies and the uncertainty distributions to perform this analysis. Uncertainties are assumed to originate from manufacturing defects, instrumentation errors, and correlation and model uncertainties. Two methodologies for steady-state uncertainty analysis have been employed to this point. The first method uses Monte Carlo sampling of the input uncertainty distributions and propagates them through the steady-state thermal-hydraulic code to arrive at a distribution of maximum power level. This technique requires many runs with the steady-state thermal-hydraulic code to produce enough samples to develop the distribution of maximum power. However, once developed, the distribution can be examined to determine the appropriate maximum power for any desired nonexceedance probability level. The second method relies on statistically developed peaking factors and uses a combination of statistical and deterministic methods. Two peaking factors are calculated: one for the local heat flux and the other for the coolant temperature rise. The peaking factors are generated by sampling the appropriate input uncertainty distributions and developing a peaking factor distribution for each. The peaking factor value then is selected depending on the nonexceedance probability level desired. The steady-state code is then executed only one time to determine the maximum power level at this nonexceedance probability level.

Several additional models and codes have been developed to study other thermal-hydraulic aspects of the core region. Flow splits between various regions of the core [i.e., each fuel element, the core pressure boundary tube (CPBT), the central hole region, the control rods] have been calculated to determine appropriate orificing within the core region. Thermal and fluid models of the inner control rods have been developed to identify cooling deficiencies and determine the fluid lift forces acting over the rods. Analytical techniques have also been developed to perform preliminary analysis of beam tube and reflector tank cooling.

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) 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. 5.

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 acceptance criteria (see Table 3.3) could not be defined. These five criteria, discussed in detail in ref. 5, are expressed in terms of probability levels; thus, core performance must be converted to a probability function to test for compliance.

Table 3.3. Acceptance criteria

Condition	Acceptance criteria	Nonexceedance probability
Normal operation	No boiling	95%
	Fuel temperature < 400°C (area < 1 mm^2)	95%
	Temperature drop across $A1_20_3$ <119°C	95%
Anticipated event	No critical heat flux	99.9%
	Fuel temperature < 450°C (area < 1 mm^2, time < 8.5 days)	95%
	Temperature drop across $A1_20_3$ < 119°C	95%
Unlikely event	No critical heat flux	95%
	Fuel temperature < 525°C (area < 1 mm^2, time < 1800 s)	95%

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.

The decision has been made to use statistical techniques to perform uncertainty analysis in order to eliminate the excess conservatism inherent in "worst-case" uncertainty analysis methods. Uncertainty distributions must therefore be developed for each uncertainty parameter used. For thermal-hydraulic analysis, these include manufacturing, measurement, and correlation uncertainty distributions.

3.1.3.2.1 Status

An initial sensitivity analysis was performed using the steady-state thermal-hydraulics code to determine the most significant uncertainty parameters. The original list of 26 parameters was reduced to 15 in this preliminary evaluation. Three of the 15 remaining parameters were treated deterministically: the local fuel segregation plus nonbond, fuel extending beyond the radial boundary, and fuel extending beyond the axial boundary. Distributions were then assigned to the remainder. Uncertainty distributions for the thermal-hydraulic correlations (such as heat transfer coefficient, friction factor, FE) have generally been developed from the data base described in Sect. 3.1.1.2, while development of calculational uncertainties (such as the local core power level) has relied on engineering judgment.

Distributions for manufacturing uncertainties have been assumed using fuel defect detection limits as anchor points. A statistical evaluation of as-built coolant gap variations has been performed using HFIR core data. In addition, a study has begun to statistically evaluate sample fuel plates for fuel segregation defects. The initial evaluation has been performed on a HFIR-design fuel loading that uses uranium silicide fuel. A digital plate scanning technique has been developed for this purpose. Data gathered from this instrument are being analyzed statistically to assemble a data base that can be used to develop appropriate distributions. Other techniques will be used to determine distribution data for fuel nonbond defects and to determine if a correlation exists between fuel segregation and nonbond locations. Efforts will continue to develop dependable manufacturing uncertainty distributions.

This task also supports detailed thermal calculations of the fuel plate defects which have been used to evaluate the local heat flux peaking and fuel temperature profiles caused by fuel defects (segregation and nonbond). These calculations have also been used to determine the sensitivity of the fuel to improved detection limits and have led to the establishment of ANS-specific detection requirements. These tasks have laid the groundwork for integrating detailed local fuel calculations with the more global calculations performed by the steady-state thermal-hydraulics code.

Uncertainties in instrumentation response presently are based on vendors' quoted errors, which, up to this point, have been assumed to be two or three standard deviations. The distributions have been assumed to be normal.

A more detailed description of the uncertainty analysis has been presented in ref. 19. This task will continue to develop improved uncertainty distributions as additional information becomes available.

3.1.4 Reactor Physics Support to Design

3.1.4.1 Justification for the Reactor Physics Support to Design Task

This task provides the reactor physics support to the development of the reactor core. Without reactor physics support for design, there would be little understanding of the nuclear performance characteristics of the ANSR. 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 considered important enough to be treated as a separate R&D activity (WBS 1.1.5). One of the central activities of this task is to interface with the reactor systems task (WBS 1.3) and the experimental systems task (WBS 1.4) to provide feedback on important physics parameters that have been identified as design requirements. The interface with the engineering staff defines the core geometry for the physics analysis. The development activities from the physics standpoint are those that support design development and safety analysis. Other key tasks to be carried out besides the basic physics analyses include the following.

3.1.4.2.1 Development of the fuel and burnable poison distribution

To maximize thermal-hydraulic safety margins, the fuel distribution within the involute fuel plates must be graded to optimize the power distribution. The fuel distribution is graded in both a radial and an axial direction in the conceptual core design. Continued studies must be carried out to develop a grading for the core design and enrichment to be carried forward into Title I design. The grading must be continually refined to account for design refinements and improvements in the fuel cycle modeling methods. In addition, work will be carried out to explore the feasibility of using a grading with variation of the fuel distribution in the radial direction only.

The boron burnable poison in the conceptual core design was located in the fuel end plates. Studies will be carried out to evaluate the use of additional or alternate burnable poison materials, as well as alternate locations such as the fuel meat or side plates of the elements. The purpose would be to improve the safety margins and reduce the risk of criticality events during refueling.

3.1.4.2.2 Calculation of reactivity worths and reactivity coefficients

The safety analysis carried out to support issuance of the PSAR and FSAR requires a determination of reactivity worths associated with core component and moderator temperature and density changes. As the design evolves and as the analysis methods are refined, these parameters must be recalculated and input to the safety analysis tasks. Specific reactivity worths of interest are the fuel temperature coefficient from Doppler broadening, which varies with fuel enrichment; moderator temperature and void reactivity worths at various locations inside the CPBT; and the reactivity worths associated with burnable poison and control rod temperature changes.

3.1.4.2.3 Perturbation analysis

The effects upon reactivity and power density peaking resulting from fuel plate movement, fuel expansion, and fuel plate loss must be determined to support safety analysis and the evaluation of uncertainty estimates in safety margins. Light water ingress effect upon reactivity, both inside the CPBT and in the reflector tank; reactivity effects from failed experimental components; and other material effects from irradiation targets and isotope production facilities are evaluated under this task.

3.1.4.2.4 Isotope depletion and buildup over the fuel cycle

The fuel cycle analysis must include an accurate accounting of fuel depletion, burnable poison depletion, fission product buildup, control material transmutations, silicon buildup in the aluminum components, and various other isotopic transmutations that might affect core lifetime, safety parameters, and material properties. The fuel cycle analysis will be predominantly carried out using 2-D diffusion or transport theory models that will be validated under WBS 1.1.1.1 and WBS 1.6. Some fuel cycle analysis will be performed using MCNP and MOCUP to provide additional confirmation of key fuel cycle parameters, particularly the heat loads (see WBS 1.1.11).

3.1.4.2.5 Refueling and start-up analysis

Refueling analysis interfaces with the mechanical design development of the core refueling system. Evaluation of core and fuel element criticality during the entire loading and unloading process must be carried out for each new refueling concept at each location in the loading path. New analyses must be performed whenever the core geometry, fuel loading, or fuel enrichment is changed.

3.1.4.2.6 Decay heat analysis during refueling

It will be necessary to determine the decay heat in the fuel elements from the fission product decay photons and beta particles, and from the die-away of the fission process just after shutdown, to evaluate cooling requirements after shutdown, during refueling, and during element storage. Decay heat loads on the components will be evaluated.

3.1.4.2.7 Source terms for shielding analysis

The reactor physics support to design must also provide fission source distributions over the fuel cycle as input to the reactor shielding task (WBS 1.1.11).

3.1.4.2.8 Target isotope production

The production of californium-252 and other isotopes must be determined for evaluation of target design options. Target optimization studies will be carried out.

3.1.4.2.9 Dosimetry analysis of proposed flux measurements

This task will support evaluation of flux regimes within the reflector tank.

3.1.4.2.10 Quality assurance and documentation

The analysis to be carried out will be fully documented, with cross references to the data, codes, and models used to produce reported results.

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 [e.g., coolant pressure, temperature, flow rate, cooling state (i.e., forced or natural flow)] 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., $T_w = T_{sat}$, incipient boiling, FE, CHF, maximum centerline temperature, and oxide spallation) as the basis for accessing core performance and will include appropriate uncertainty treatment. The thermal-hydraulic support to design task is also broken into phases that correspond to the major safety and design milestones. In addition to the SARs, there are two other subtasks that support Title I and Title II design.

3.1.5.2.1 Status

Core analysis has focused upon improving fuel loading design and operating margins. This work has included identification of a series of "ideal" core power profiles that create a uniform margin along the length of the core. These profiles have been used to guide the fuel loading and the neutronic calculations. "Local" analysis has also been used to link more intimately the neutronic and thermal-hydraulic design. The steady-state thermal-hydraulic code has been used iteratively with the neutronics codes to define margins on a local basis. A weighting scheme was then used to alter local fuel loadings based on these margins. This scheme has improved maximum limiting operating powers (based on oxide limitations) by approximately 30% over previous fuel designs. The resulting fuel design tends to skew the power distributions to the core inlet (i.e., higher power densities at the core inlet than at the exit) where coolant pressures are highest and coolant temperatures are lowest.

Thermal margin analysis has been performed for various nonexceedance probability levels, corresponding to various design basis scenarios. These have included both 95% and 99.9% nonexceedance probability levels, corresponding to unanticipated and anticipated events, respectively. Results have indicated that the core power is presently limited by oxide spallation at the operating margin, and by the FE limit at the safety margin. Calculations have shown that the CHF limits and FE power limits are very close in magnitude, while the incipient boiling limit is approximately 30% below the FE limit, and $T_w = T_{sat}$ limit is approximately 7% below the incipient boiling limit for the same nonexceedance probability level. Efforts are continuing to extend the analysis to much lower nonexceedance probability levels in order to prove core melt frequencies of less than 10^{-4} under normal operating conditions.

Thermal-hydraulic analysis of nonfuel reactor components [e.g., the CPBT, control rods, reflector tank wall, beam tubes] is also performed under this task. Preliminary parametric and design calculations have been used to determine appropriate operating conditions for these components considering limiting conditions such as incipient boiling and maximum component temperature. In addition, calculations have been performed to evaluate the lift forces acting on the control rods both to determine the forces acting on the latches, etc., and to evaluate rod response during accident conditions (e.g., a control rod shearing off).

Flow splits (between the two cores, the CPBT annulus, the central hole region, and the control rod coolant channels) have been 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 have assumed isothermal conditions but will eventually be extended to nonisothermal conditions. These calculations have been 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 that may require thermal-hydraulic evaluations.

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 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 1996. This allows the Title II design of the reactor system to proceed on schedule.
2. Complete the fuel performance report by June 2000. This allows about 2 years of review before the publication of the FSAR.

The fuel development activity is divided into 12 WBS level four tasks summarized in Table 3.4. 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 8-year period covered by this R&D plan are given in Table 3.5, and the associated schedules are shown in Fig. 3.2. Note that the expense costs are associated with initial concept developments, fabrication and performance restrictions, the analyses and documentations that directly support preliminary (Title I) and detailed (Title II) design activities, 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. The line item costs include the various tasks needed to define the production mode fabrication process and purchase some tooling and required inspection equipment. Subsections 3.2.1 through 3.2.12 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 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.

Table 3.4. 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 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 have been identified. At present no backup cladding material is considered and no backup fuel materials are viable at the loading levels required. 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. 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 target and/or 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— Since no existing irradiation facility has the capability to irradiate prototype ANS fuel under prototypic ANS conditions, a fuel performance model must be used to predict fuel performance at or beyond ANS operating conditions. Although the average burnup rate can be approached in the irradiation tests discussed, the peak burnup rate in the ANS fuel cannot be duplicated in any reactor. The irradiation damage simulation by ion bombardment at very high dosage rates will allow a determination of whether the irradiation behavior will change drastically at these higher burnup rates. These data, along with data from the in-reactor irradiations, will be used to develop and validate the fuel performance model.
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.
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 modified (or new) fabrication techniques.

Table 3.4 (continued)

WBS	Task description
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.
1.1.2.10	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	Production mode fabrication process development—This task will provide the effort needed to establish the production mode fabrication process. Although some cores may be fabricated under this task, there are no plans to use these cores in the ANS reactor.
1.1.2.12	HFIR experimental silicide core—This task will support the safety analyses and documentation necessary to support the use of an experimental core in HFIR containing U_3Si_2 fuel, along with a limited postirradiation examination.

ANS = Advanced Neutron Source.
HFIR = High Flux Isotope Reactor.

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

3.2.1.2.1 Status

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, 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 retain the 6061 aluminum alloy because of the extensive experience with it in both fabrication and corrosion performance. Other alloys certainly have the potential for satisfactory performance [e.g., the alloys used by the French (AlMg2) and the German (AG 3 NE) companies]. Some alloys have the potential to improve fabricability (higher compressive strength at fabrication temperatures), and others have the potential to improve corrosion performance (such as 8001 and the AlFeNi). However, the 6061 alloy appeared to be a suitable cladding material for the ANS. It was not clear that significant improvements could be obtained (or would be needed) by pursuing a development program on cladding alloys.

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 either of the commonly used dispersoid fuel compounds for aluminum (the U_3O_8-Al dispersion used in the HFIR and others or the UAl_x used in the Advanced Test Reactor (ATR) and other reactors). The U_3Si_2-Al dispersion was chosen based on its higher density and proven good irradiation performance in tests

Table 3.5. WBS level four breakdown of costs for the fuel development activity

WBS Level 3	Level 4	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.2		Fuel Development										
	1.1.2.1	Selection and verification of fuel and cladding (Done)	Exp.									0
	1.1.2.2	Capsule irradiation test	Exp.	210								210
	1.1.2.3	Miniplate irradiation tests	Exp.	650	800	800	310					2560
	1.1.2.4	Irradiation damage simulation studies and fuel performance	Exp.	220	220	200	188	106	106	53		1093
	1.1.2.5	Final performance report	Exp.				41	41				82
	1.1.2.6	Plate and fuel element fabrication	Exp.	479	479							958
	1.1.2.7	Dummy plate and element fabrication	Exp.	200	700	330						1230
	1.1.2.8	Prototype fuel element fabrication for critical experiments	Exp.		431	854	200					1485
	1.1.2.9	Full-size plate irradiation	Exp.		502	268	204					974
	1.1.2.10	Burnable poison selection and testing	Exp.	50	52	52	52					206
	1.1.2.11	Production mode fabrication process development	Line		1010	2210	2235	660	660	660		7435
	1.1.2.12	HFIR Expermental Silicide Core	Exp.	372	392	226	218					1208
		Subtotals	Exp.	2181	3576	2730	1213	147	106	53	0	10006
			Line	0	1010	2210	2235	660	660	660	0	7435
		Contingency	Exp.	109	358	273	121	15	11	5		892
			Line		202	442	447	132	132	132		1487
		Total	Exp.	2290	3934	3003	1334	162	117	58	0	10898
			Line	0	1212	2652	2682	792	792	792	0	8922

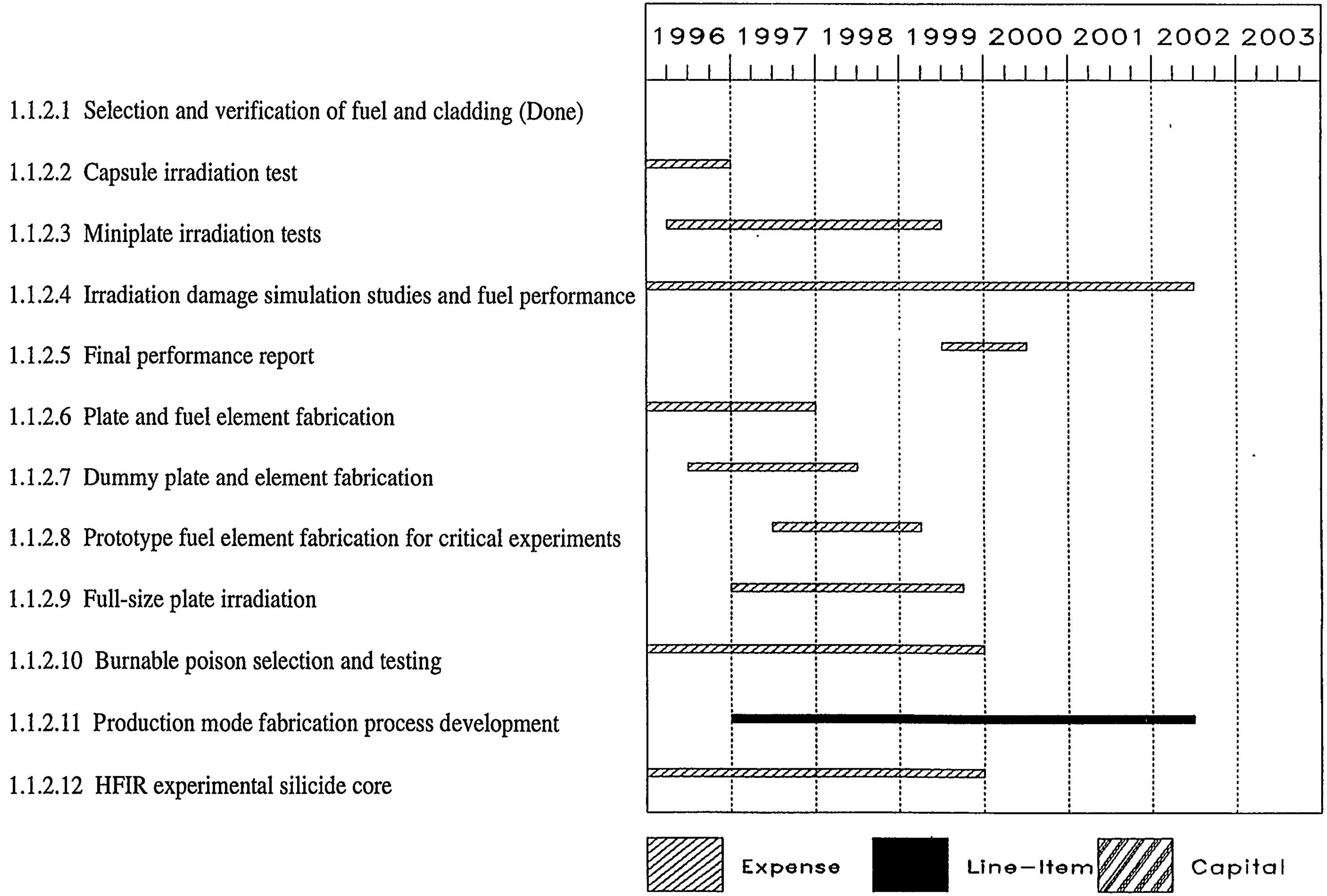


Fig. 3.2. Schedule for WBS 1.1.2 fuel development.

in lower power reactors. Since this selection, the specific uranium density in the ANS decreased to the level where U_3O_8 and UAl_x were also viable options, and they were placed in the program as backup fuel options. 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, lowering the uranium enrichment level to 50% ^{235}U has required increasing the fuel density to the point where neither U_3O_8 nor UAl_x ia a viable backup fuel material as a dispersoid 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 were 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 the maximums 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. Four tests are included in the program and are designated HANS-1 through HANS-4. All the tests contain passive temperature monitors, and the test matrices for the first three are in Table 3.6.

3.2.2.2.1 Status

The first two tests have been irradiated and postirradiation examination is complete. The third test has been irradiated and is awaiting disassembly for postirradiation examination. The test matrix for the fourth test is incomplete at this time, but plans are to irradiate it to lower burnup than the first three to examine the fuel behavior at lower fission densities more typical of the most of the fuel in the ANS. The capsule specimens for HANS-1 consisted primarily of the prime candidate fuel at various temperatures and burnup rates (32 of 36 specimens). HANS-2 consisted of the backup fuels U_3O_8 and UAl_x and the prime fuel U_3Si_2 under conditions as representative as possible of the extremes in ANS. The third test will contain a combination of the three fuels irradiated under various conditions with an important difference in specimen preparation. The specimens in the third test are small punchings of dispersion meats from hot-rolled plates, rather than lightly compacted powder blends. This will make the thermal and chemical conditions of the test fuel particles typical of those in the reactor fuel. The primary evaluation in all three tests is 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. The fission gas bubble size distributions will be used with the fuel performance model to predict particle swelling rates and overall fuel performance.

The results of the first two capsule tests show that the fuel particles exhibit predictable swelling under the peak conditions expected in the ANS, even though swelling is substantially higher than was experienced in the previous low-fission-rate, low-temperature irradiation tests. As shown in

Table 3.6. Test matrices for the High Flux Isotope Reactor target capsules

	HANS-1		HANS-2		HANS-3	
Fuel holder number	Fuel	Temp. (°C)	Fuel	Temp. (°C)	Fuel	Temp (°C)
1	U_3Si_2	425	U_3O_8	425	U_3Si_2	≤200
2	U_3Si_2	375	UAl_2	425	U_3Si	≤200
3	U_3Si_2	325	U_3O_8	250	U_3Si	250
4	U_3Si_2	250	UAl_2	250	U_3Si_2[a]	250
5	U_3Si	≤250	U_3O_8	325	U_3O_8	250
6	U_3Si_2	250	UAl_2	375	U_3Si	425
7	U_3Si_2	325	U_3O_8	375	U_3O_8	425
8	U_3Si_2	375	UAl_2	425	U_3Si_2	425
9	U_3Si_2	425	U_3O_8	425	U_3Si_2	250
10	U_3Si_2	425	UAl_x	425	U_3Si_2[b]	250
11	U_3Si_2	375	U_3Si_2	425	U_3Si_2[b]	425
12	U_3Si_2	325	UAl_x	375	UAl_2	425
13	U_3Si_2	250	U_3Si_2	375	UAl_2	250
14	U_3Si	375	U_3Si_2	325	U_3Si_2	250
15	U_3Si_2	425	UAl_x	250	U_3Si_2[a]	425
16	U_3Si_2	375	U_3Si_2	250	U_3Si_2	425
17	U_3Si_2	325	UAl_x	425	U_3Si_2[b]	425
18	U_3Si_2	250	U_3Si_2	425	U_3Si_2[b]	250

[a]These specimens contain uranium enriched to 40% in ^{235}U.
[b]These specimens contain uranium enriched to 20% in ^{235}U.

Fig. 3.3, the U_3Si_2 fuel particles exhibit three zones after essentially full burnup under these extreme conditions, which appear to be as follows: the outermost periphery is a reaction zone consisting of a $U(Si,Al)_3$ structure which converted to aluminide before recrystallization could occur and has only extremely small stable fission gas bubbles; the second zone has undergone recrystallization of the U_3Si_2 structure before converting to the aluminide structure and contains small stable gas bubbles; the innermost zone is an amorphous zone in which the uranium has been depleted to the point that a more unstable silicon compound is formed before aluminum diffusion and/or recrystallization could stabilize the bubble morphology and larger bubbles occur. The overall particle swelling of the high burnup particles is higher than previously experienced for U_3Si_2, but it does not appear to be a problem since the larger bubble region is contained by the outermost regions of the particle. The ANS will reach these extremes of burnup only in a small percentage of the fuel at the periphery of the fuel plates where the meat is thin. Particles of lower burnup typical of the bulk of the ANS fuel do not exhibit the inner core of larger bubbles. The switch to lower enrichment of the fuel will also limit the burnup and uranium depletion in the particles in the peak regions to the point where recrystallization may occur and limit the bubble size.

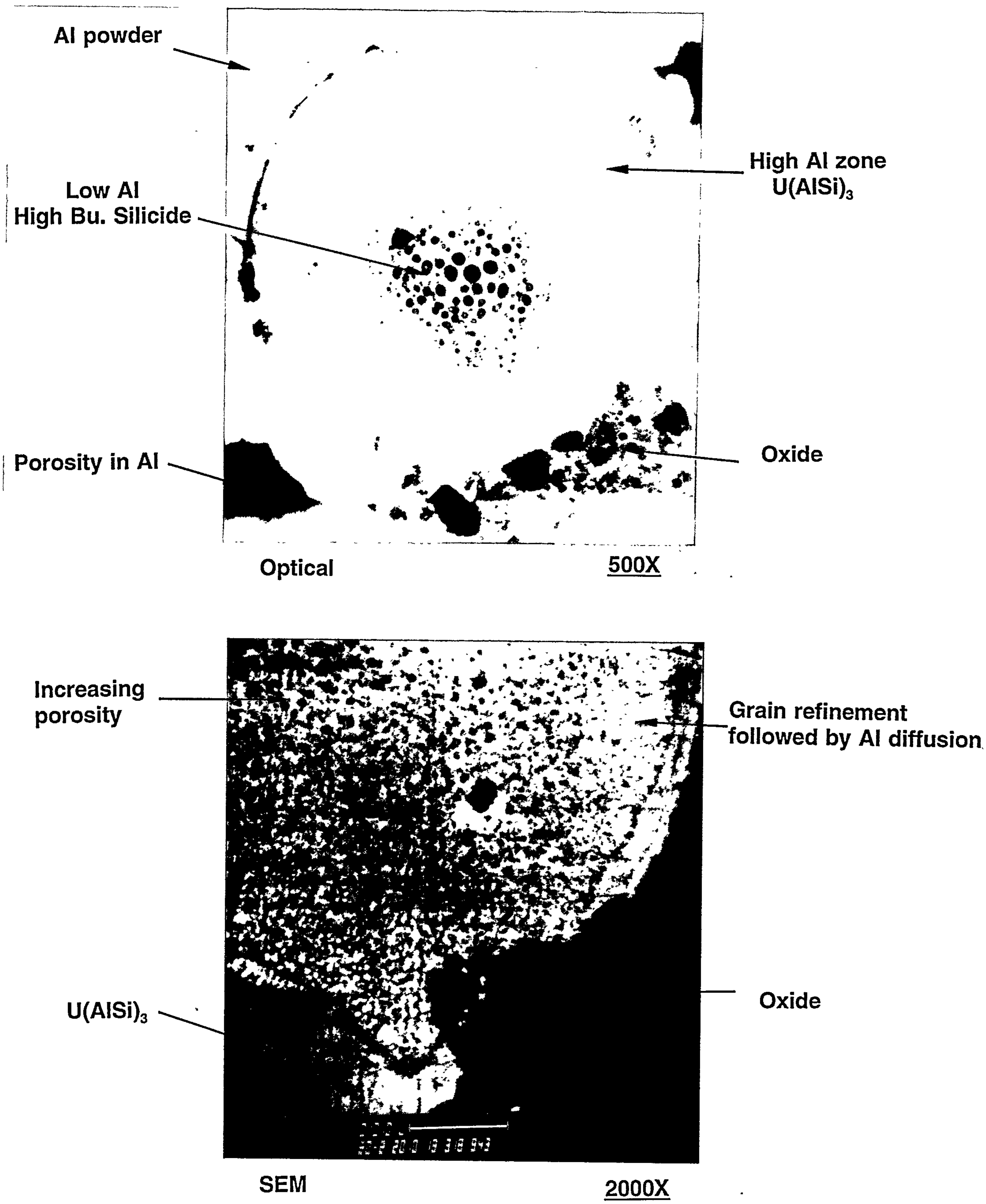


Fig. 3.3. Microstructure of U_3Si_2-Al dispersion fuel irradiated to 90% burnup in HFIR at ANS conditions.

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 post-irradiation 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. The test matrix has not yet been established. We are currently proposing a series of miniplate tests in the target region of HFIR where the flux conditions approximate those of the ANS fuel. The preliminary mechanical, thermal, and neutronic design must be completed before the test matrix is formalized. The preliminary thermal design indicates that in the HFIR target region dedicated to this test, the surface heat flux can approach the peak ANS conditions so that prototypic fuel loadings can be tested. Areal ^{235}U loadings will be varied by changing the meat thickness, the volume fraction, the enrichment level, or some combinations of the three to obtain the desired fission rates and burnups in the tests. Once the test limitations are established, a test matrix will be designed to validate the fuel performance model. When used in conjunction with the model, the matrix will qualify the fuel for ANS conditions. 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

It is impractical to test the ANS fuel under the full range of its expected operating (and anticipated operational transient) 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 test all anticipated conditions. 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 all anticipated regimes. 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.

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 U.S. fabricator of fuel for HFIR and eventually for ANS). 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. We plan to investigate whether the fuel particles can be passivated with a thin oxide film to ease these fabrication concerns. 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.

3.2.6.2.1 Status

Results of the fabrication development program to date are promising for the successful fabrication of fuel to ANS requirements. Modifications to the specifications for aluminum powder and to the particle size distribution of U_3Si_2 powder have resulted in the ability to produce HFIR plates containing U_3Si_2 with homogeneity as good as in the U_3O_8 HFIR plates. The feasibility of producing fuel plates with a dual fuel gradient has been demonstrated. The reproducibility and uncertainty levels associated with a given distribution appear to be very good in the development plate lots produced thus far. The goal homogeneity levels appear to be achievable and may even be tightened where desired for thermal performance margin. The conversion to digital data acquisition of the transmission X-ray scanning device at Babcock and Wilcox for determining fuel distribution will allow more sophisticated inspection criteria for homogeneity. Additional experience with different lots of fuel and aluminum powder is required before final practical homogeniety levels can be set.

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. Dummy elements are also required for the reactor mockup and the refueling machine. 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 four complete dummy elements (fuel elements composed of dummy plates) will be fabricated for use in other tasks and experiments. The dummy fabrication task will supply input to the fuel element fabrication development task 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 the physics performance of the ANSR core. This critical experiment, performed under WBS 1.1.6, requires prototype fuel elements, 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 set of fuel elements 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 and will consist of a module of several plates with prototypic spacing. Evaluations of the fuel capsule and fuel miniplate tests will be performed before final fabrication of the full-sized plate test.

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 has been shown to lower the resistance to blistering in postirradiation heating tests (the temperature at which postirradiation heating produces blistering is still above the maximum fuel temperature during operation). Poisons that have no gas generation or less gas release 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 produces fuel 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 safety margins 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.

3.2.12 Irradiation of Experimental U_3Si_2 Element in HFIR

3.2.12.1 Justification for the Irradiation of Experimental U_3Si_2 Element in HFIR Task

Fabrication of a HFIR element with U_3Si_2 plates will provide additional fabrication experience with the relatively new fuel, and the satisfactory operation of the element in a high-performance reactor such as HFIR will provide added confidence of satisfactory operation in the higher-performance ANS.

3.2.12.2 Description of the Irradiation of Experimental U_3Si_2 Element in HFIR Task

The experimental U_3Si_2 element to be operated in HFIR will operate at the standard HFIR power and flow conditions. The element will be loaded with additional uranium and burnable poison to extend the core life if it proves feasible to do so. The element would then serve a dual purpose as a demonstration for ANS and for an extended-life HFIR element. The ANS Project will pay for the necessary safety assessments and documentation changes to make it possible to operate the experimental core. The HFIR will pay for the actual fabrication of the element and its operation. The ANS Project will then perform the limited postirradiation examination to verify satisfactory performance.

3.3 CORROSION TESTS AND ANALYSES—WBS 1.1.3

Aluminum alloy fuel cladding for high-power-density research reactors (e.g., HFIR, 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 ANSR 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 the following:

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; and
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 September 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 3.7. With the exception of some ATR analysis work performed at INEL, and possibly some High Flux Beam Reactor (HFBR) analysis work to be performed at Brookhaven National Laboratory (BNL), all of this work will be performed at ORNL. The total estimated costs for this activity over the 8-year period covered by this R&D plan are given in Table 3.8, and the associated schedules are shown in Fig. 3.4. All tasks are funded by expense money. 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 new facility capable of testing aluminum alloy specimens over a wide range of heat flow, coolant flow rates, and coolant properties was assembled 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 start-up 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 1996: the corrosion test loop or the thermal-hydraulic test loop will be modified for heavy water use to allow confirmatory testing of corrosion in heavy water.

Table 3.7. 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 modifications 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. Staff, surveillance, loop operation and maintenance, and preparation of specimens and test sections are covered under this task.
1.1.3.3	Analysis of experimental data—This task includes the disassembly of the test sections; the nondestructive, metallographic, and specialty analyses; and the calculations and evaluations necessary to understand and use the data obtained from the out-of-pile corrosion tests.
1.1.3.4	Analysis of ATR oxide data—This task was established to examine exiting data on oxide film growth obtained in the ATR under ATR conditions, and the initial effort was completed in FY 1989. A reinvestigation of corrosion behavior of certain ATR fuel plates will be conducted in FY 1996—FY 1997 by INEL to provide support to task 1.1.3.5.
1.1.3.5	Comparison tests under irradiation conditions—The anticipated extension of task 1.1.3.4 to include correlation of ATR fuel plate corrosion behavior with the ANS out-of-pile data will provide a basis for assessing the effect of reactor radiation of oxide film growth. Follow-up tests in the ANS loop under the ATR thermal-hydraulic and water chemistry conditions may be required for direct contrast. In addition, an examination of HFBR clad corrosion behavior may be included in this task as an additional source of information.

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. An experimental basis is necessary to provide a suitable foundation for design accommodation of fuel clad corrosion phenomena for efficient and safe operation of the ANSR.

3.3.2.2 Description of the Performance of the Loop Experiments Task

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. In particular, the discovery that the film growth behavior is extremely sensitive to coolant water chemistry imposed strict requirements on its measurement and control. This task includes all the activities to procure and prepare test specimens, fabricate the test sections, and

Table 3.8. WBS level four breakdown of costs for the corrosion tests and analyses activity

WBS Level 3	Level 4	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.3		Corrosion Tests And Analyses										
	1.1.3.1	Design/procure and install test loop	Exp.	50	50							100
	1.1.3.2	Loop experiments	Exp.	274	249	100						623
	1.1.3.3	Analysis of experimental data	Exp.	260	225	245	120					850
	1.1.3.4	Analysis of Advanced Test Reactor oxide data	Exp.	75	50							125
	1.1.3.5	Comparision tests under irradiation conditions	Exp.	10	20	10						40
		Subtotals	Exp.	669	594	355	120	0	0	0	0	1738
		Contingency	Exp.	67	59	36	12					174
		Total	Exp.	736	653	391	132	0	0	0	0	1912

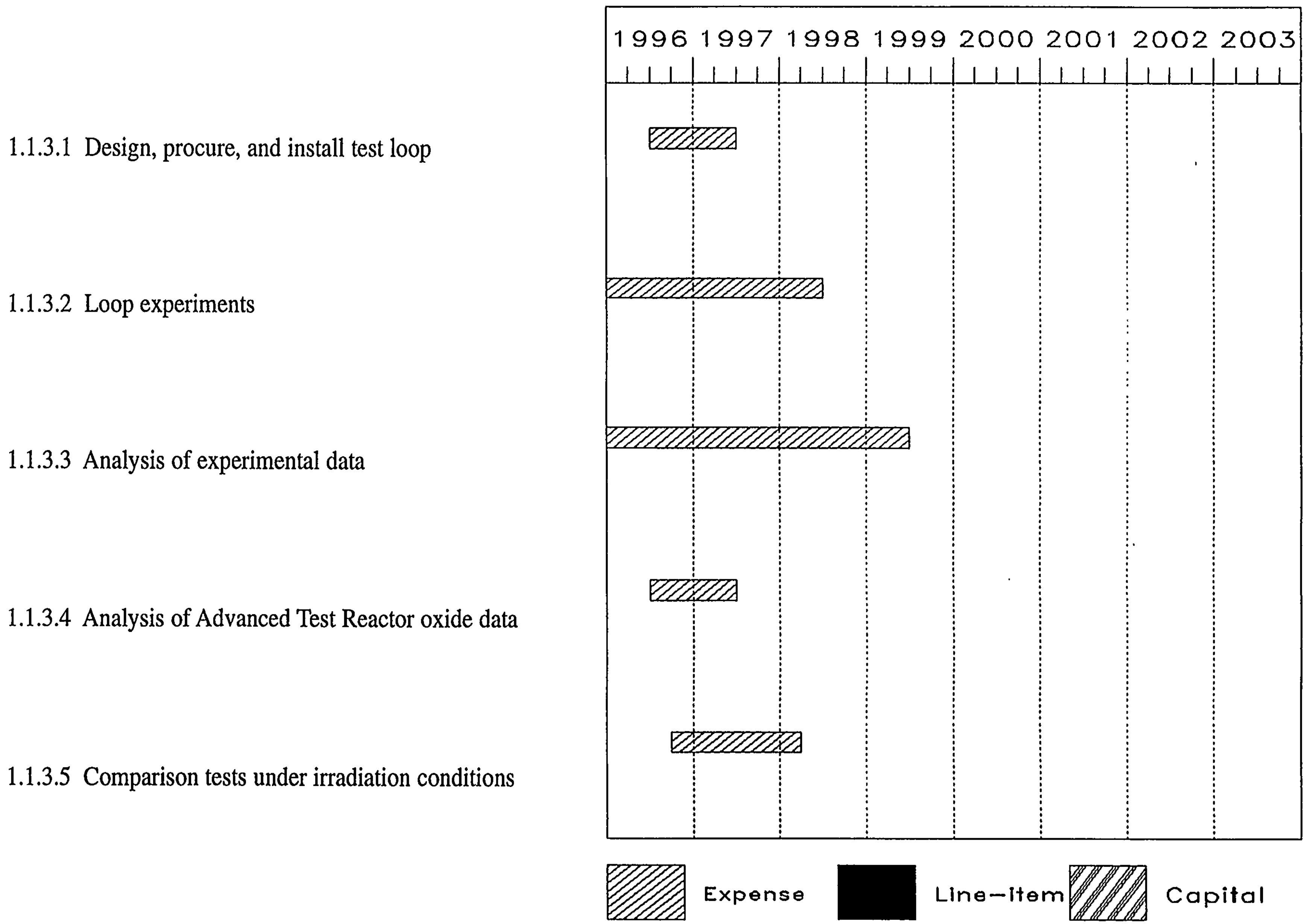


Fig. 3.4. Schedule for WBS 1.1.3 corrosion tests and analysis.

perform the tests. In addition to test surveillance and data accumulation, operation and maintenance of the test loop are included.

More than 60 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 500 h. 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 increasing corrosion product.

3.3.2.2.1 Status

Since the beginning of the corrosion tests, the procedures for specimen preparation and test section fabrication have been continuously improved. Similarly, the function and operation of the test loop, despite its many complexities, have been strengthened, and upgrades of several components have resulted in increased stability and measurement reliability. The data acquisition system (DAS) now includes several versatile features for monitoring the real-time progress of the experiment, allowing for continuous corrections and analyses.

More than 40 corrosion loop tests have now been completed for a wide range of heat fluxes and coolant velocities, temperatures, and chemistries. The experimental system parameters and their ranges for all experiments are given in Table 3.9. The whole range for some parameters was very wide, but the present focus is on the narrower maximum interest ranges also listed. It is anticipated that these interest ranges will be altered to accommodate upcoming changes in the ANS core design associated with the change to a three-element core. As is noted below in Sect. 3.3.3.2.1, these tests have contributed to the definition of the important variables controlling oxide film growth behavior on aluminum alloy cladding undergoing heat transfer and to the generation of a data base applicable over the range of thermal-hydraulic conditions appropriate to anticipated ANS operation. While most of these tests used 6061 Al specimens, a series of tests on "alternate alloys" is now being completed that includes examination of 8001 aluminum and the ILL cladding alloy. Further test series in this program are earmarked as "new core T/H conditions," "oxidation inhibition tests," and "heavy water tests." In particular, the latter two will generate information leading to a final specification of the optimum water chemistry conditions for the ANS primary coolant system.

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 from the experimental results is a necessary contribution to reactor design. Correlations to support analytical reactor performance models must be developed because all potential operating conditions cannot 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 also 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.

Table 3.9. Range of experimental parameters in ANS corrosion tests

Parameter	Expt. range (nominal)	Max. interest range
Coolant pH[a]	4.5–6.0	4.5–5.2
Coolant velocity, m/s	9–28	25–27.6
Coolant inlet temperature, °C	39–80	44–49
Local coolant temperature, °C	45–101	50–90
Local interface temperature,[b] °C	95–208	100–200
Local heat flux, MW/m^2	5–20	5–12
Experiment duration, d	to 35	to 21

[a]Mixed bed demineralizer plus HNO_3 additions in bypass stream.
[b]Calculated via forced convection heat transfer coefficient, [B.S. Petukhov, "Heat Transfer and Friction in Turbulent Pipe Flow with Variable Physical Properties," *Adv. Heat Transfer* 6, 504 (1970)].

3.3.3.2.1 Status

The accumulated test information has resulted in the building of a data base and data correlation that permit the prediction of the extent of corrosion product growth as a function of several important system parameters. This information has been reported in the open-literature and in ORNL technical reports.[20-24] For a typical experiment in which the power and coolant velocity are held constant, the temperature will vary as a function of position along the specimen in the direction of the coolant water flow, across the specimen in the direction of the heat flow, and over time as a low thermal conductivity corrosion product develops. In addition, the rate of growth of the corrosion product is itself a function of temperature so that the gradients in specimen temperature will also change over time. At the conclusion of each test, the specimen is examined extensively by several metallurgical techniques to characterize the corrosion products.

The measurements of specimen temperatures and other system parameters during the experiment are used to calculate the growth of the corrosion product oxide along its length. Computer programs that essentially model the coolant-specimen regime have been written and are used to calculate film growth and specify temperatures at inaccessible locations. The non-isothermal nature of the film growth process is responsible in part for the observed departure from simple rate laws, so we have used a semi-empirical rate equation to consolidate the data and establish the foundation for the data base and subsequent correlations.

A functional relationship, called Correlation II, for 6061 aluminum exposed to (nominally) pH 5 coolant and coolant inlet temperatures less than 50°C has been constructed in which the rate constants for oxide growth are given as a function of the oxide-metal interface temperature and the heat flux:

$$k = 6.388E7 \exp[-9154/(T_{x/c} + 1.056\grave{I})] \qquad (1)$$

where

k = film growth rate constant, $\hat{E}m^{1.351}/h$;
$T_{x/c}$ = oxide-coolant interface temperature, K; and
$\grave{I}$ = heat flux, MW/m^2.

The rate constants determined from 17 loop tests plotted in Arrhenius fashion according to the tenets of Correlation II are shown in Fig. 3.5, where the dashed line represents Eq. (1). This correlation is presently serving both as a predictor and as a basis for comparison for the various test results.

Along with the principal parameters of the specimen-coolant interface temperature and the heat flux, it has been established that the coolant water chemistry in the loop system is a critical variable. Oxide film growth has been observed to be affected by small changes in coolant pH in the range of pH 5, either as a primary or secondary influence, and is considered responsible for much of the data scatter in Fig. 3.5. Despite these issues, development of functional correlations of the data has progressed in a form useful for design calculations for the ANS core. These correlations have been used to calculate the oxide film thickness on the fuel cladding as a function of the thermal-hydraulic history, and have included a means to indicate the onset of film spallation, an unacceptable event for 6061 Al cladding.

The sensitivity of the corrosion process to water chemistry must also be considered for reactions in heavy water, the ANS primary coolant. The possibility that the observed pH effect may not carry over in a straightforward manner to pD values for heavy water operation will be assessed on an experimental basis in several planned tests to be conducted in either the corrosion test loop or the thermal-hydraulic test loop. We also intend to test the level of benefit achieved by lowering the coolant pH to 4.7 or the equivalent for heavy water operation. In addition, supplemental corrosion tests under thermal-hydraulic conditions germane to the latest ANS core design will be initiated in order to confirm the applicability of the current correlation and its contributory data base.

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 an additional basis for comparison for noting radiation effects on the clad corrosion process.

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

3.3.4.2.1 Status

The initial effort in this task was completed in 1989, when the film thickness data for a series of ATR fuel loadings were compiled. Preliminary 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. In the near future, we expect to reexamine certain of these data sets, enlarging the reactor operation information in order to compare these oxide measurements with predictions made from the ex-reactor ANS data base and data correlations.

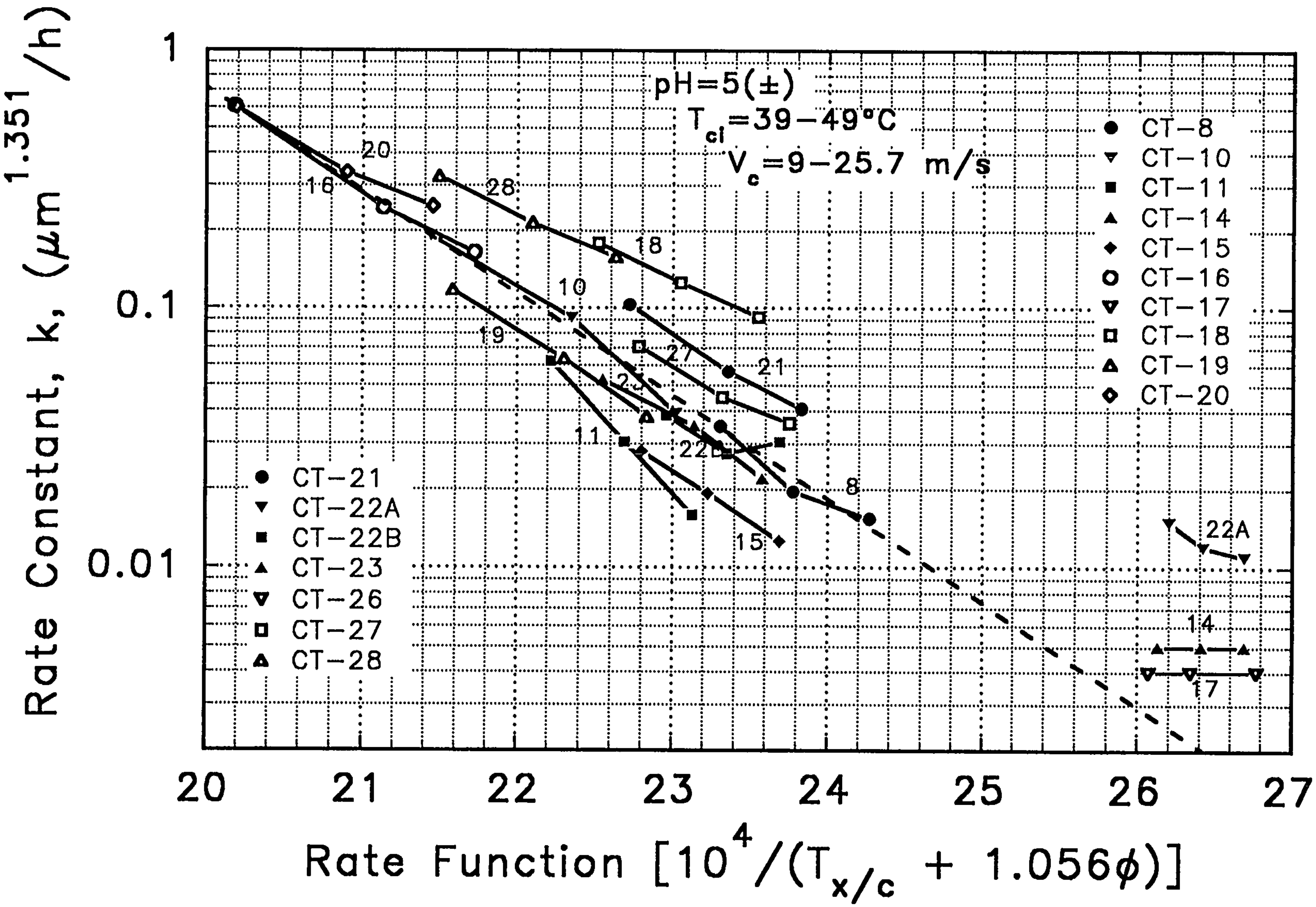


Fig. 3.5. ANS corrosion test loop results showing grouping of rate constants according to Eq. 1, "Correlation II"

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 result 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

Originally, this task included the design and preparation of a suitable comparison test vehicle and its exposure in the HFIR facility under conditions to simulate as closely as possible typical ANS operation for aluminum alloy components. Duplicate out-of-pile tests would have been 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.

We now feel that a more direct and efficient approach is to reexamine certain of the ATR oxide data, as discussed in Sect. 3.3.4, for comparison with the ex-reactor ANS corrosion test loop data base. Apart from the oxide measurements, a critical part of this work is to extract the thermal-hydraulic and water chemistry histories for the designated fuel loadings and fuel plates. In addition, an examination of HFBR clad corrosion behavior may be included in this task as an additional resource.

3.4 CORE FLOW TESTS—WBS 1.1.4

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 core flow tests activity includes the tests and analyses necessary to validate the thermal-hydraulic analytical models and to obtain basic data needed on thermal-hydraulic phenomena pertinent to design basis accident analysis. The computer codes available for safety analysis, for example, RELAP-5 for design basis 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. 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. In addition, facilities will be constructed and testing and analysis performed on thermal-hydraulic issues associated with natural circulation conditions in the reactor under both shutdown and during refueling, effects of partial inlet channel flow blockages, flow distributions in the core, and nonfuel components.

The work is divided into ten WBS level four tasks summarized in Table 3.10. Most of this work would be performed at ORNL with support from a number of subcontractors. The total estimated costs for this activity over the 8-year period covered by this R&D plan shown in Table 3.11, and the associated schedules are shown in Fig. 3.6. It should be noted that the capital equipment money is associated with the design and construction of test facilities. All of the analysis work and operation of test facilities planned for this task are supported by expense money. Subsections 3.4.1 to 3.4.10 provide more detailed information on the WBS level four tasks under this activity.

Table 3.10. Summary description of 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 ANS 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 other operating, two pumps off) to determine the worst case conditions for core and component cooling.
1.1.4.3	Natural circulation off-normal tests—The natural circulation loop developed under WBS 1.1.4.2 will be used with modifications made to incorporate specific requirements for two-phase testing. It will include an experimental design that accurately simulates the pressure drop and mass flux performance of the ANSR during natural circulation decay heat removal. Potential areas of study include single phase heat transfer coefficients, the point of incipient boiling, flow excursion, and critical heat flux behavior.
1.1.4.4	Low mass flux DNB tests—These tests will address the approach to thermal limits under pump coastdown-type and pony motor pumped core flow conditions. 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). These tests will be conducted in the THTL. Time-average pressure drop and mass flux information from the tests will be coupled with the CHF model to allow a lumped parameter simulation of the fuel assembly performance under the applied power and pressure drop situations.
1.1.4.5	Full 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. An experimental test section will be developed to determine the effect of increased span-to-gap ratio. In addition, a test section will also be designed to examine both hot spots and hot streaks that may exist in the core because of fuel manufacturing defects.

Table 3.10 (continued)

WBS	Task Description
1.1.4.6	Integral transient tests—These tests will be used to validate transient thermal-hydraulic codes for conditions (e.g., subcooling levels, coolant velocities) typical of the ANS reactor. An integral test platform will be developed to perform RELAP5 benchmark testing for models describing phenomena such as accumulator behavior, pressure wave propagation through the primary system, pump cavitation/air ingestion as well as overall system behavior.
1.1.4.7	Hydraulic tests of nonfuel components—These tests will be conducted, as necessary, to determine the adequacy of cooling for nonfuel components (e.g., control rods, beam tubes, CPBT, reflector tank walls) that are exposed to very high heating rates under both nominal and off-normal reactor operating conditions. The experimental design will be driven by thermal analysis of each component; experiments will be performed only as necessary and only on the phenomena or component area of interest.
1.1.4.8	Flow blockage tests—Spatial variations in the fuel assembly inlet flow velocity and temperature must be limited to assure safe operation of the reactor. Debris obstructing the flow channels or upstream structural components provide possible sources of these variations that must be addressed. A combined experimental and computational approach is being pursued within this task to define the range of inlet flow velocity and temperature allowable. Primary tests in the flow blockage test facility will characterize the flow behavior within the channel behind a variety of blockages under ANS hydraulic operational conditions and will also be used to benchmark CFD models.
1.1.4.9	Thermal-hydraulic testing for refueling—These tests will address natural convection cooling capability during movement of the fuel elements. Specific tests will be designed to develop an adequate thermal limit data base and allow model development for these conditions. This test series is expected to use the natural convection test facility instrumentation system, developed under WBS 1.1.4.2 and modified to address conditions expected during refueling.
1.1.4.10	Full flow hydraulic tests—These tests will examine issues associated with integrated operation of a complete core assembly under ANSR design flow conditions. Included will be measurement and evaluation of flow distribution between fuel elements and other core components, measurement of pressure distributions within the core region, and evaluation of component stability and vibration under full flow conditions.

Table 3.11. WBS level four breakdown of costs for the core flow tests activity

WBS Level 3	Level 4	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.4		Core Flow Tests										
	1.1.4.1	Thermal Hydraulic test facility	Exp.	994	100							1094
			Cap.									0
	1.1.4.2	Natural circulation tests	Exp.	107	189	1015	343					1654
			Cap.	323	1655	129						2107
	1.1.4.3	Natural Circulation Off-normal Tests	Exp.			647	1062	105				1814
			Cap.									0
	1.1.4.4	Low Mass Flux DNB Test	Exp.	701	1069							1770
			Cap.									0
	1.1.4.5	Full Span Test Facility	Exp.	408	275	1029	1271	665				3648
			Cap.	437	1824	605						2866
	1.1.4.6	Intergral Transient Tests	Exp.		90	179	890	869				2028
			Cap.		2047	1322	273					3642
	1.1.4.7	Hydraulic Tests of Non-fuel Components	Exp.	206	324	388	929	194				2041
			Cap.		1114	815	272					2201
	1.1.4.8	Flow Blockage Tests	Exp.									0
			Cap.									0
	1.1.4.9	Thermal Hydraulic Testing for Refueling	Exp.			289	1087	105				1481
			Cap.			602						602
	1.1.4.10	Full Flow Hydraulic Test Facility	Exp.		45	100	204	1452				1801
			Cap.	0	219	3430	1535					5184
		Subtotals	Exp.	2416	2092	3647	5786	3390	0	0	0	17331
			Cap.	760	6859	6903	2080	0	0	0	0	16602
		Contingency	Exp.	242	209	365	579	339				1734
			Cap.	152	1372	1381	416					3321
		Total	Exp.	2658	2301	4012	6365	3729	0	0	0	19065
			Cap.	912	8231	8284	2496	0	0	0	0	19923

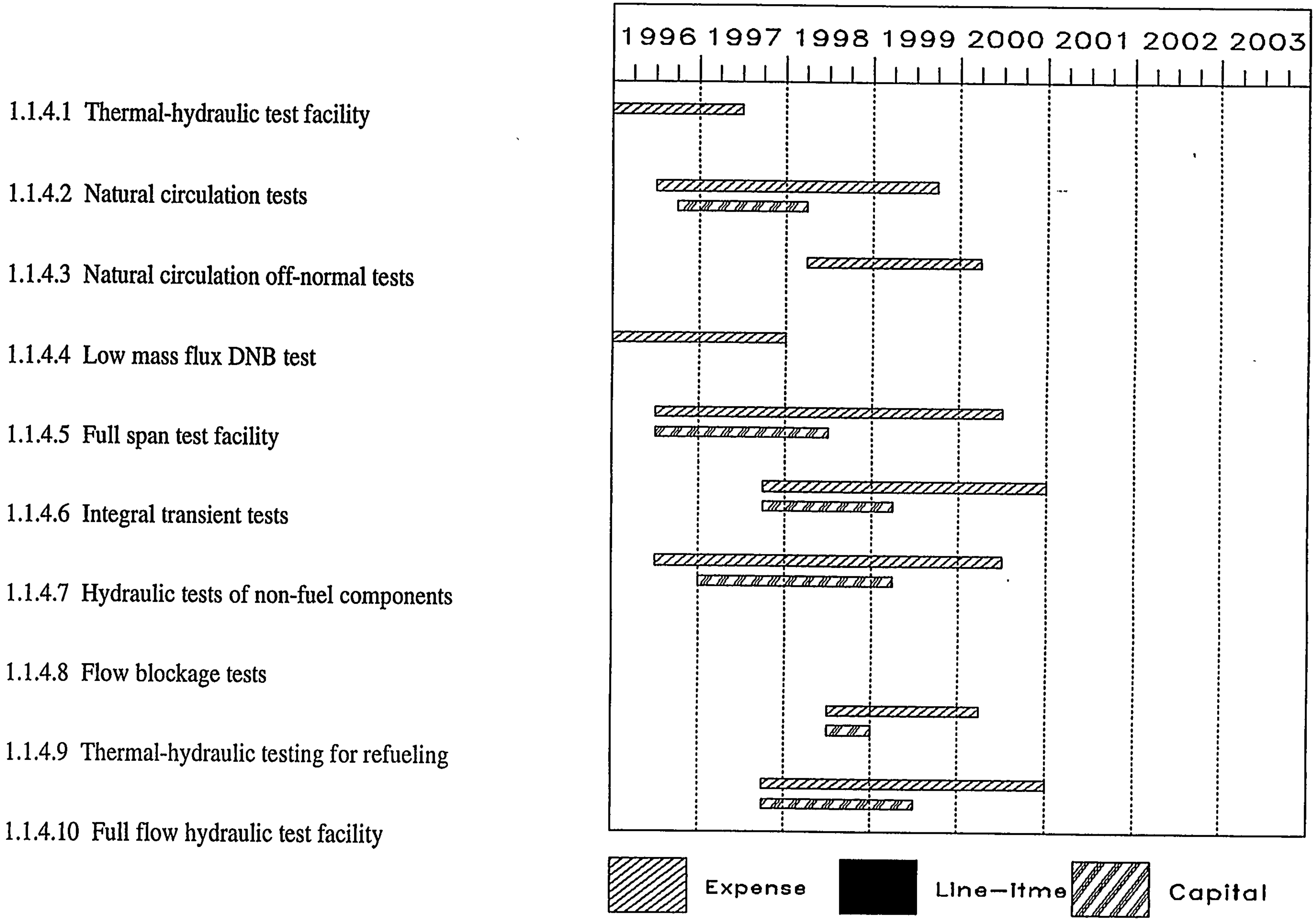


Fig. 3.6. Schedule for WBS 1.1.4 core flow tests.

This WBS element contains a major project milestone:

Complete transient thermal-hydraulic testing by the end of September 2000. 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.

3.4.1 Thermal-Hydraulic Single Channel Tests

The thermal-hydraulic single channel tests 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 Single Channel Tests

The availability of CHF 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 ANSR operating conditions. If these data are not obtained, larger uncertainties and additional margins would have to be included in the design that would most likely downgrade the performance of the facility.

3.4.1.2 Description of the Thermal-Hydraulic Single Channel Tests Task

The thermal-hydraulic performance of the ANSR will directly influence both its normal and off-normal operational limits and performance. To characterize the thermal-hydraulic performance of the ANSR, 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 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 1.2.3.) This plan incorporates experimentation to examine phenomena that are critical to characterizing the thermal-hydraulic behavior of the ANSR and to evaluating more integral phenomena that must ultimately be determined using codes and models. The plan attempts to isolate the important phenomena as well as, under some circumstances, collect data to support statistical analysis of the reactor performance.

Planned experiments will be performed in the thermal-hydraulic test loop (THTL). A photograph of this facility is shown in Fig. 3.7. A schematic diagram showing the key components of the facility is shown in Fig. 3.8. This facility was designed and built to provide known thermal-hydraulic conditions for a simulated full-length coolant subchannel of the ANSR core, thus facilitating experimental determination of FE and CHF thermal limits under expected ANSR conditions. Determination of these two thermal limits and the relationship between them is the main objective of the THTL facility. However, the facility is also designed to examine other thermal-hydraulic phenomena, including onset of incipient boiling, single-phase heat transfer coefficients and friction factors, and two-phase heat transfer and pressure-drop characteristics. Although the facility's primary aim is to develop the thermal-hydraulic correlations at the ANSR nominal conditions for normal operation and safety margin analysis, tests will also be conducted that are representative of decay heat levels at both high pressure (e.g., loss of off-site power) and low pressure (e.g., a loss of coolant accident), as well as other quasi-equilibrium conditions encountered during transient scenarios.

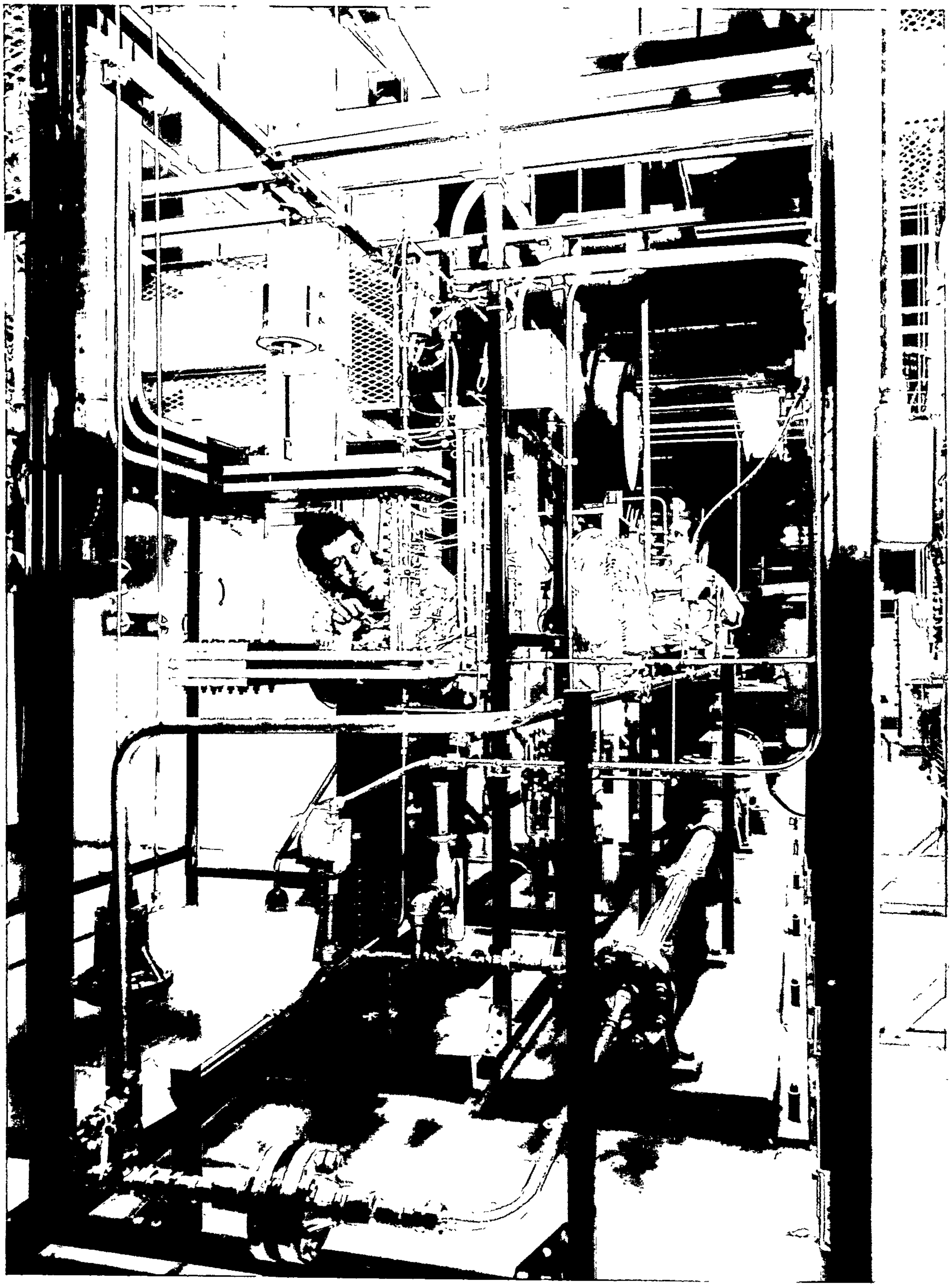


Fig. 3.7. Thermal-hydraulic test loop.

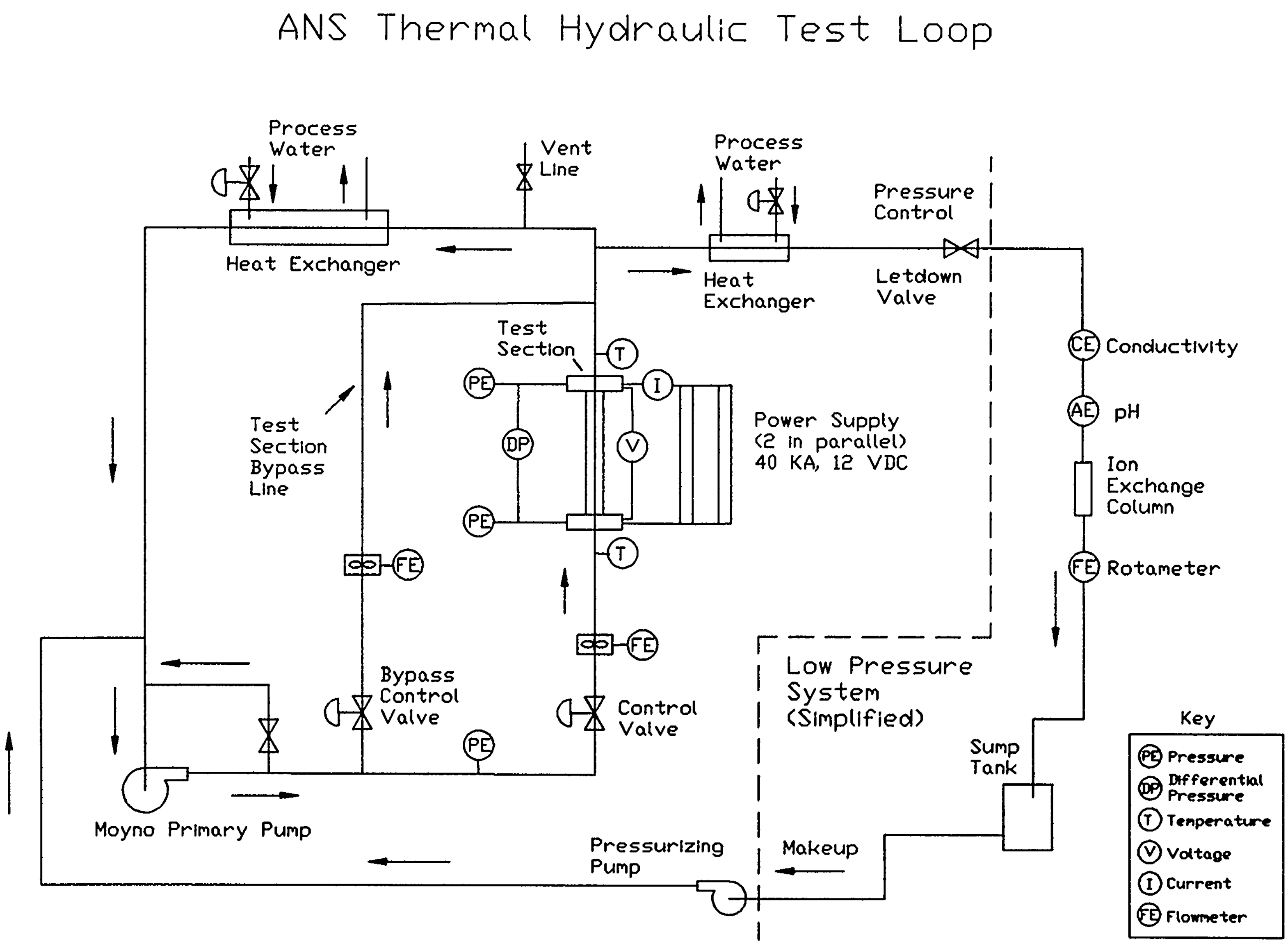


Fig. 3.8. Schematic diagram of the THTL primary components and instrumentation.

3.4.1.2.1 Status

The initial experiments have focused on the FE phenomena at ANS nominal conditions as well as a few true CHF experiments for comparison. The cooling channels in the ANSR fuel assembly are all parallel and share common inlet and outlet plenums, effectively imposing a common pressure drop across all the channels. Such a configuration is subject to FE and/or flow instability[25,26] that may occur once boiling is initiated in any one of the channels. Figure 3.9 presents a typical plot of the pressure drop vs flow rate relationship under various boundary conditions. In the case of many parallel channels between large common headers, as is the case in the ANSR, the slope of the external supply system is practically zero and is represented in Fig. 3.9 by horizontal lines (A and B). Based on this observation, FE or onset of flow instability (OFI) conditions were determined in most of the THTL FE experiments by detection of the test section pressure-drop minimum as the flow to the test section was reduced under a constant heat flux. This method allowed for repetition of many nondestructive FE tests without experiencing an actual FE that normally causes test section failure. For confirmation and comparison, limited experiments were performed with an actual FE burnout, and some experiments were run with true CHF burnout under constant flow. One of the main goals of these tests is to determine the relationship between CHF and FE under ANSR conditions. To accommodate all these experimental needs, the design of the THTL system had to respond to three separate modes of operation as enumerated below.

1. A **"soft"** system was used to perform actual FE tests with burnout. In this mode, a large bypass around the test section is fully open to maintain an almost constant common pressure drop across both the test section and bypass flow paths.
2. A **"stiff"** system was used to perform true CHF tests with actual burnout at constant and known flow rates. In this mode, the bypass around the test section was completely closed to maintain a constant flow through the test section. In addition, a near positive displacement pump was used in the primary loop.
3. A **modified "stiff"** system was used to perform simulated FE tests without experiencing actual FE. In this mode, a closed or minimal bypass configuration—along with a significant pressure drop across the flow control valve upstream of the test section—was used to prevent actual FE or other flow instability. In this case, the potential for FE was determined by detecting the minimum pressure drop in a plot of pressure drop vs flow rate (which coincides with the OFI point).

The test section simulated a single subchannel in the ANSR core with a rectangular geometry that had a full prototypic length (507 mm), the same flow-channel gap (1.27 mm) but with a reduced span (12.7 and 25.4 mm), and the same material (aluminum 6061-T6) with a surface roughness (~0.5 μm) reasonably close to that expected in the ANSR fuel plates. A detailed description of the test facility and the test section design is given by Felde et al.[27] and Siman-Tov et al.[28]

3.4.1.2.2 Data reduction, results, and analysis

An experimental data reduction model was developed for single-phase forced-convection flow, focusing on the flat portion of the test channel. A cross-section of the 12.7-mm-span test channel is shown in Fig. 3.10. Local heat flux on the flats was calculated based on temperature-dependent resistivity of the aluminum test section and the average heat flux on the flat. The experimental data reduction model also includes a single-phase pressure drop calculation model. Internal lateral and axial heat redistribution by thermal conduction were considered in a preliminary way in determining the local heat flux, but are being reconsidered in more detail through the use of a 3-D conduction model.

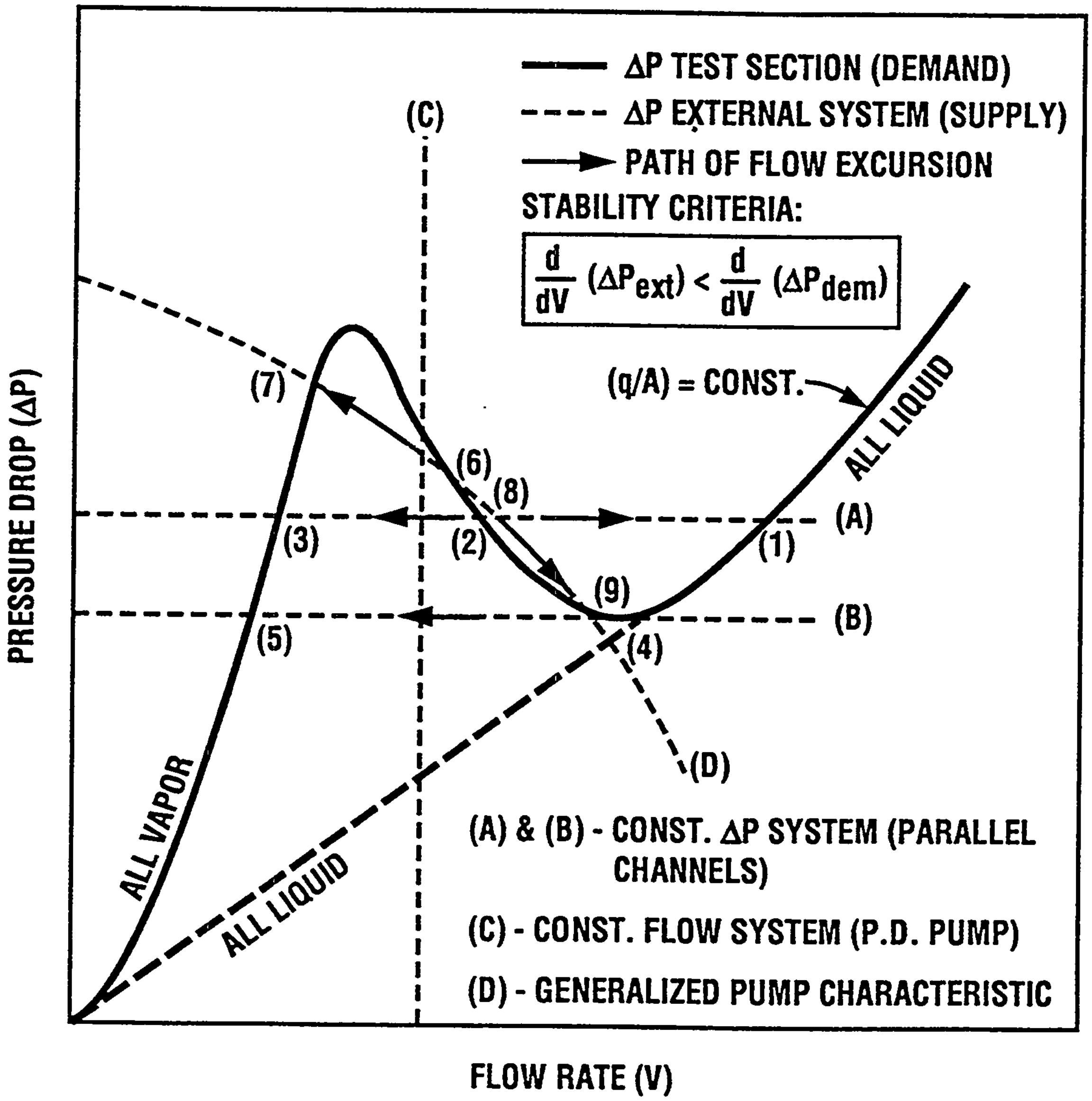


Fig. 3.9. Schematic interpretation of excursive instability.

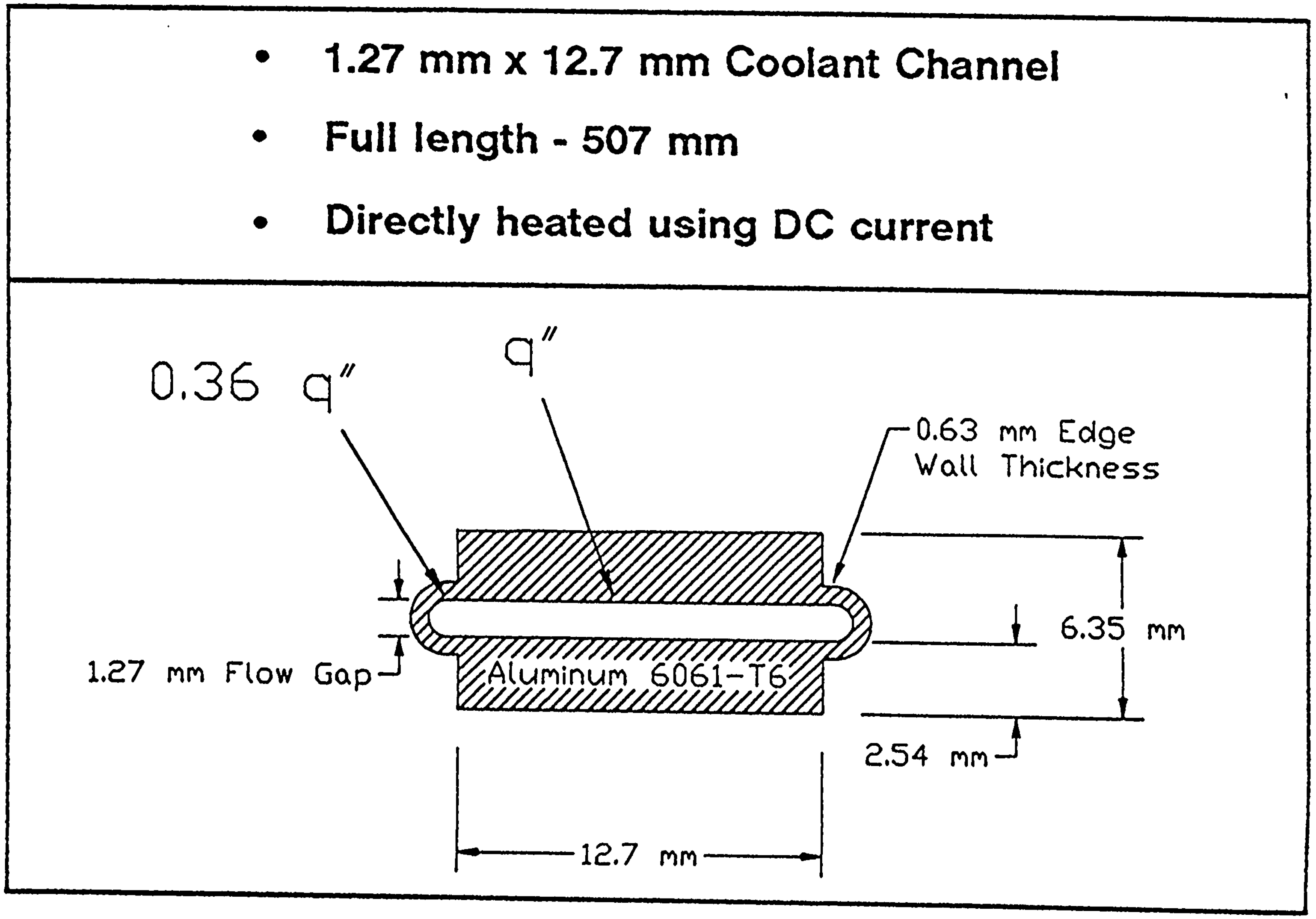


Fig. 3.10. Cross-section of the test channel in the THTL.

The THTL experimentation is focused initially on FE and CHF phenomena as reported by Siman-Tov et al.[28] Since then, additional experiments were performed that expand the data range to much higher heat fluxes (18 MW/m^2) and velocities (28.4 m/s). The major experiments completed so far include the following thermal-hydraulic conditions:

- coolant: water,
- inlet coolant temperature: 45°C,
- exit coolant pressure: 0.4–1.7 MPa,
- local (exit) heat flux range: 2–18 MW/m^2,
- corresponding velocity range: 2.5–28.4 m/s, and
- channel configuration: 1.27 × 12.7 × 507 mm (rectangular), and 1.27 × 25.4 × 507 mm (rectangular).

A summary of the data in the form of plots of pressure-drop vs mass flux for both the FE and CHF tests is shown in Fig. 3.11. The heat fluxes indicated for each test correspond to the channel average values at the critical point (minimum pressure-drop or actual burnout) of that specific test.

Acquiring FE data at this level of heat fluxes and velocities is significant for two reasons. First, very few data are available for FE at velocities higher than 10 m/s. In the initial part of this activity, the range was extended to 21 m/s as reported earlier by Siman-Tov et al.[28] and has now been extended further to 28.4 m/s (at the exit), which is above the maximum velocity expected in the ANSR, including uncertainties. Second, the high heat flux achieved in this study is well above both the ANSR nominal average heat flux of 5.9 MW/m^2 and the nominal peak heat flux of 12 MW/m^2, and is almost as high as the "hot channel peaking factor" heat flux (e.g., peak heat flux with all uncertainties) of 18 MW/m^2. The limiting velocity corresponding to the ANSR nominal peak of 12 MW/m^2 is about 20 m/s (at a subcooling of 23°C), well below the ANSR nominal velocity of 25 m/s.

Preliminary results from tests with a test section of wider span (25.4 mm) indicate no major effect on critical velocity due to the increased span dimension. Final conclusions on the subject, however, await additional experiments with larger span test sections.

The collected data were compared with correlations by Costa,[29] Whittle and Forgan (W&F),[30] and Saha and Zuber (S&Z)[31] and are shown in Fig. 3.12. In the Costa correlation, the FE heat flux is proportional to the square-root of velocity, whereas both W&F and S&Z (as well as most other FE correlations) show a linear dependence. The S&Z correlation shows the best agreement with the data; the Costa correlation becomes increasingly conservative with increases in the mass flux. These and other considerations support a shift from the Costa correlation to the S&Z correlation for the ANSR, especially since for most of the ANSR scenarios analyzed (including nominal conditions), the mass fluxes are quite high. Plotting both the THTL data and the rest of the ANS database for FE (Siman-Tov et al.[32]) in terms of Stanton (St) number [the selected dimensionless group for Peclet number (Pe) >70,000 in the original S&Z correlation] against subcooling reveals a clear trend for St [and to a lesser degree, Nusselt (Nu)] to increase with decreasing subcooling, a trend that becomes stronger at lower subcoolings. In addition, both St and Nu **must by definition** become infinite for zero subcooling, which contradicts constant values for these parameters, as suggested by the original S&Z correlation. This observation, in combination with the trend observed in the data, provides a strong case for modifying the S&Z constant St (0.0065) and Nu (455) criteria, especially in the low subcooling range. Based on best-fit with the ANS data base, the following modifications to the original S&Z correlations were developed and proposed for ANSR applications on a preliminary basis:

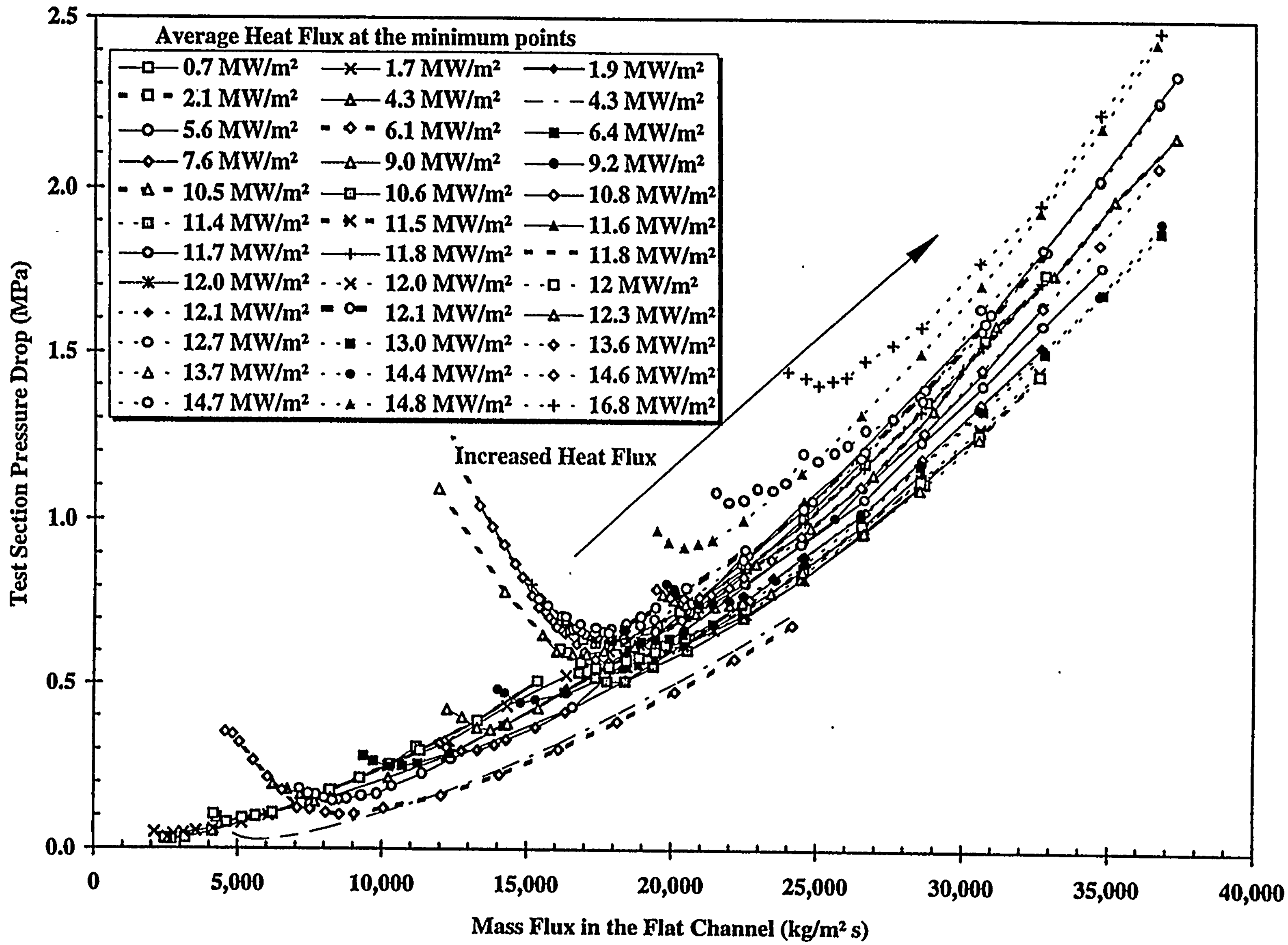


Fig. 3.11. Flow excursion data from thermal-hydraulic test loop experiments.

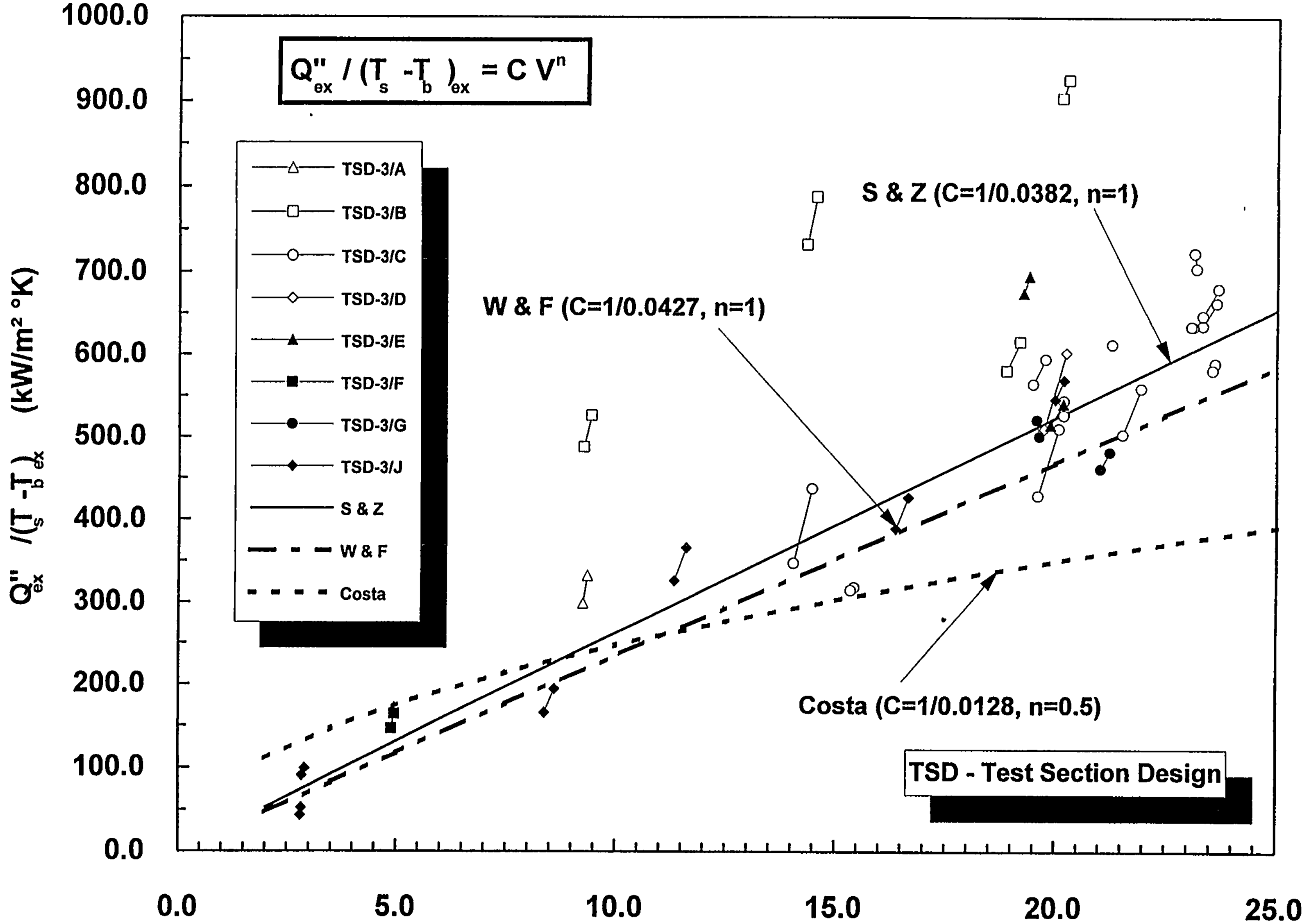


Fig. 3.12. Comparison of flow excursion data from thermal-hydraulic test loop experiments.

$$St_{FE} = 0.0065\ [0.55 + 11.21/(dT_{sub})],\quad Pe_{FE} > 70{,}000 \text{ and} \tag{2}$$

$$Nu_{FE} = 455\ [0.55 + 11.21/(dT_{sub})],\quad Pe_{FE} < 70{,}000 \tag{3}$$

where

dT_{sub} = subcooling temperature difference = $Ts - Tb$ (K),
Nu_{FE} = Nusselt number at FE point,
Pe_{FE} = Peclet number at FE point,
St_{FE} = Stanton number at FE point,
Tb = bulk coolant temperature (K),
Ts = bulk coolant saturation temperature (K).

Figure 3.13 shows the selected modified correlations in terms of *St* number and *Nu* number, respectively, against the ANS data base, including the THTL data of the present study. Statistical evaluation of these modified S&Z correlations shows considerable improvement in the means and standard deviations compared with those of the original S&Z and Costa correlations. Additional investigation of these proposed modifications is still in progress to determine their final acceptable form. A more detailed discussion of the experiments and results and the proposed modifications and their constraints is given in Siman-Tov et al.[33]

In addition to the above FE experiments, which are based on the minimum pressure drop (nondestructive tests), three destructive FE experiments and one true CHF experiment were performed. The results are presented in Fig. 3.14. The CHF experiment was performed at 12 MW/m^2 in a "stiff" system with a practically closed bypass line. It shows a 30% additional margin in velocity (12.3 m/s) compared with the FE velocity of 17.0 m/s at the minimum pressure drop, which is clearly identifiable in Fig. 3.13, and approximately a 50% margin compared with the ANS nominal velocity of 25 m/s. The three destructive FE experiments were performed in a "soft" system, but each with a different by-pass ratio (BPR) as indicated in the figure. As can be seen in the two tests performed at 12 MW/m^2 heat flux and 1.7 MPa exit pressure, the one with larger BPR (6.15) had a burnout critical velocity much closer to the minimum than the one with lower BPR (2.63), demonstrating again the affect of BPR. The third destructive FE was performed at conditions as close to Costa's (1969) as possible. The heat flux (2.29 MW/m^2) used is the same as in Costa, but the pressure (0.38 MPa) was higher than in Costa (0.17 MPa) because of loop limitations, and the inlet temperature (40°C) was set to be lower than in Costa (74.2°C) in order to achieve the same **exit** subcooling (4.3°C). As can be seen in the figure, the minimum pressure-drop point was achieved at a velocity of 4.7 m/s, compared with 5.4 m/s in Costa. When an attempt was made to reduce the velocity a little further, an actual destructive FE occurred at a velocity of 4.2 m/s in spite of the fact that the system was relatively stiff, with a bypass ratio of about 1.0. In this case, the destructive FE occurred much closer to the minimum in spite of a lower bypass ratio of about 1. The higher sensitivity of those conditions to FE is also reflected in the larger negative slope of the dP/dV curve on the left hand side of the minimum in test THTL-09 (similar to Costa conditions) as seen in the figure (left side of the minimum point), compared with the slopes of the other two FE tests. Another difference between the two tests is the closer agreement of the FE heat flux data with the Costa correlation. The measured heat flux in case FE318B was about 12% higher than the Costa correlation, whereas in cases FE212A and FE331A it was about 50% and 60% higher, respectively. This is in agreement with our previous conclusion stated in the PSAR that the Costa correlation may be more conservative at higher velocities (and pressures). In addition, a number of tests were performed at decay heat removal conditions, which are also represented in Fig. 3.12 as test section TSD-3/J. A more detailed discussion of the experiments and results is given in Siman-Tov et al.[28]

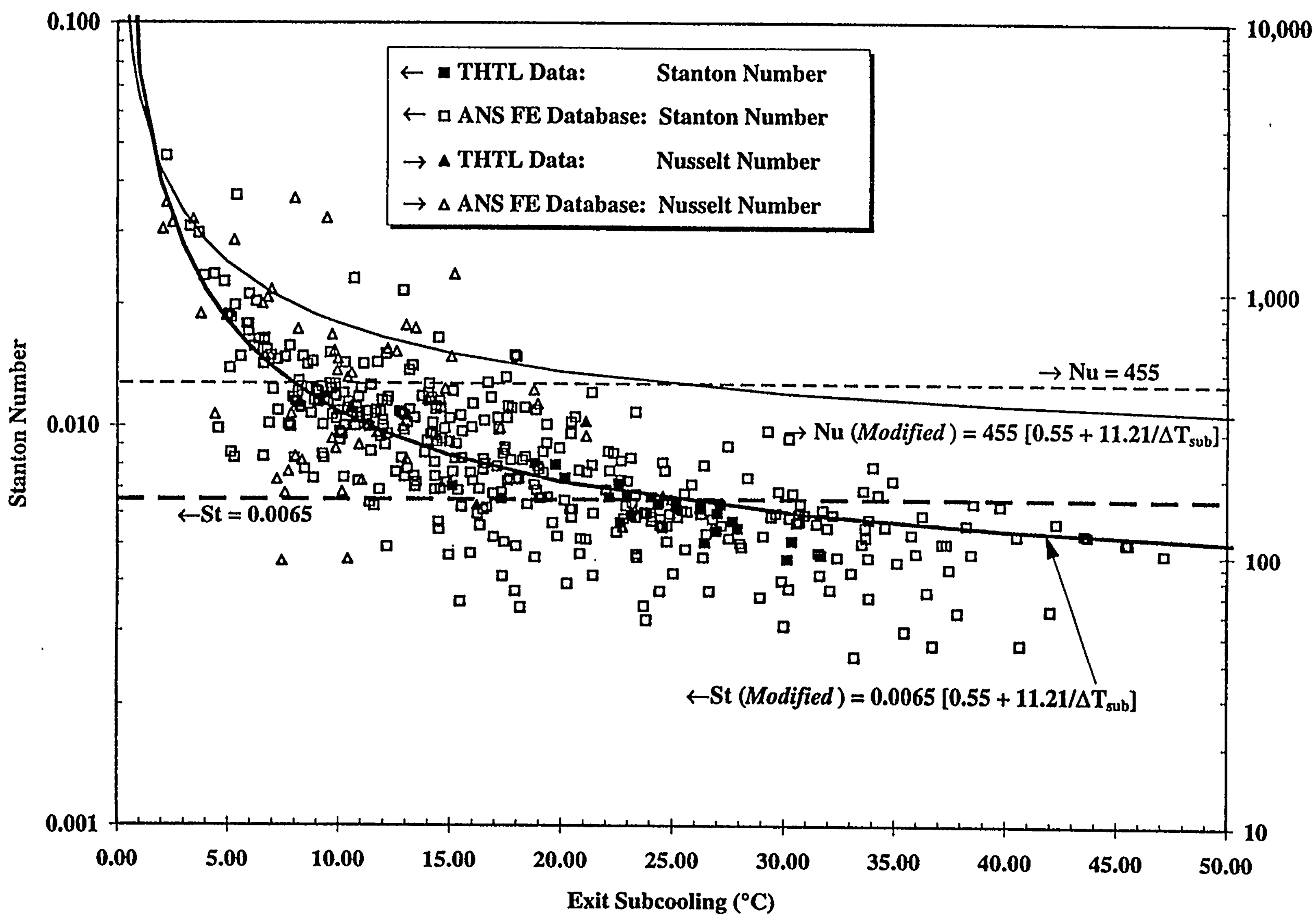


Fig. 3.13. Comparison between the modified and unmodified forms of the Saha and Zuber correlation.

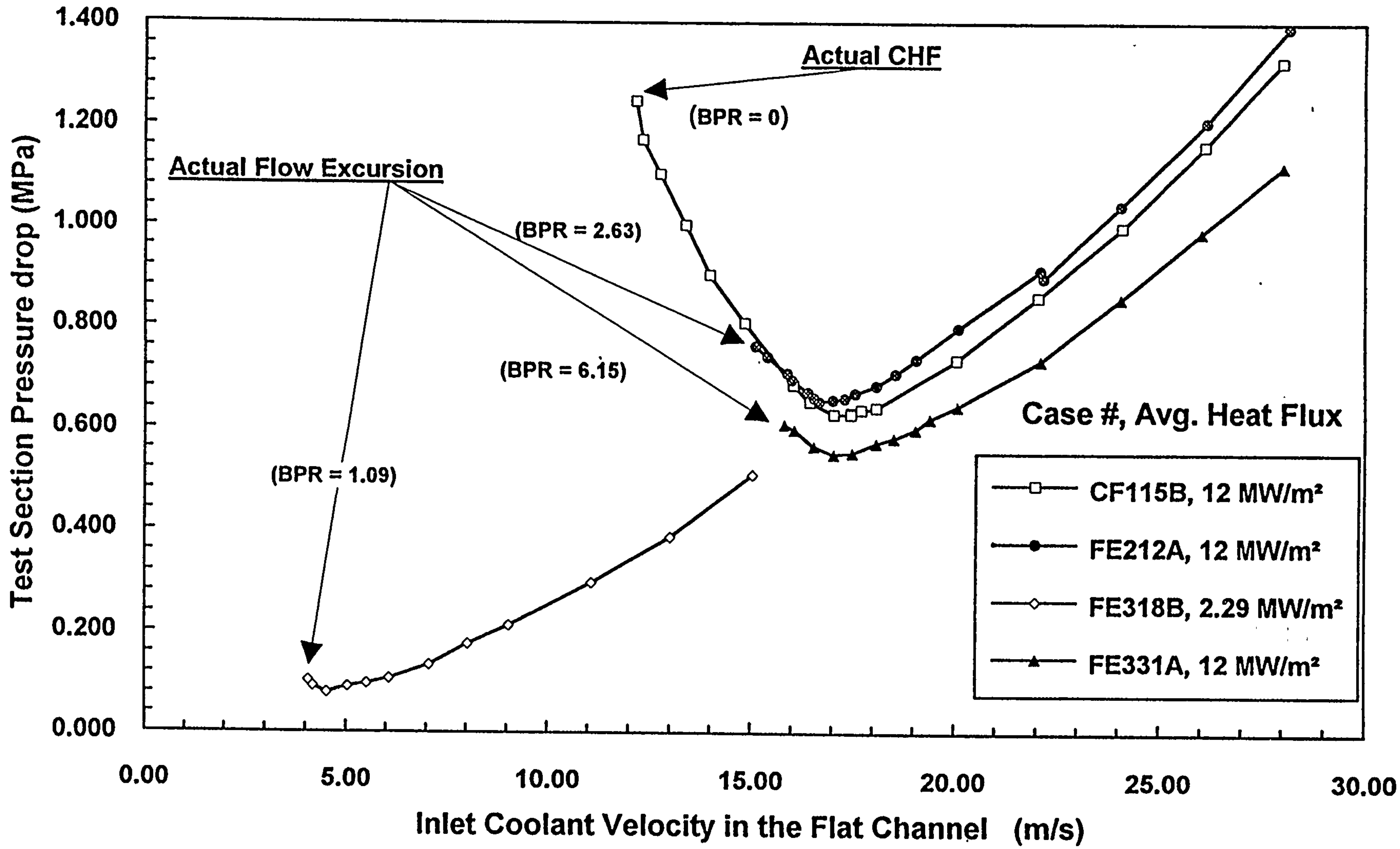


Fig. 3.14. Destructive critical heat flux and flow excursion tests in the thermal-hydraulic test loop.

3.4.1.2.3 Future plans

The initial focus of the THTL experimentation for the ANSR is the determination of thermal limits under ANS nominal conditions using water as the coolant. Plans include additional experiments to be performed with the existing facility and the basic test section design to capture the onset of incipient boiling, single-phase heat transfer coefficients and friction factors, and two-phase heat transfer and pressure drop characteristics.

Also included are plans for off-nominal conditions including low-flow tests simulating shutdown and refueling conditions, low-pressure conditions simulating a loss-of-coolant accident and other selective quasi-equilibrium conditions encountered during transient scenarios (described in Sects. 3.4.4 and 3.4.9 below), the effects of oxide buildup (typical for aluminum), the effects of material and its surface roughness, and the effects of heavy water on the thermal limits [CHF, FE, and incipient boiling].

Near-term plans under this subtask include additional testing of wider span test channels. These tests are aimed at verifying the scalability of the 12.7-mm-span channel data that make up most of the current THTL data base. Testing using the shorter heated length of the 3-element core design (418 mm) will also be performed, with both the 12.7- and 25.4-mm-span test channels. In addition, some shorter test sections will be tested to obtain prototypic exit conditions for high-power CHF testing. Some "fuel defects" type testing may be required that will address effects such as hot spots and hot stripes. Recent advancements in post-fabrication inspection techniques for the fuel plates may affect these requirements, however, and a study will be performed before this testing effort is initiated. Testing under low pressure and velocity conditions is planned to verify thermal limits under transient conditions such as pump coastdown and full power depressurization. Additional scoping tests are planned to examine characteristic time scales for the FE transient and for the oxide spallation phenomenon that has been observed on the aluminum channels during testing.

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 ANSR is designed with the intent of providing 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 nonforced flow conditions.

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. 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.

3.4.2.2.1 Natural Convection Test Facility

It is not expected that this facility must be full scale, but it must incorporate a heated region to simulate the natural circulation conditions properly within the reactor primary coolant loops and must include an upper flow annulus, hot leg piping, heat exchanger, cold leg piping, and pump discharging to the simulated "fuel assembly." Scaled accumulators will also be included in the design. The facility will be designed to allow incorporation of modifications to facilitate use for the off-normal natural circulation tests described in Sect. 3.4.3 and the refueling tests described in Sect. 3.4.9.

3.4.3 Natural Circulation Off-Normal Tests

3.4.3.1 Justification for the Natural Circulation Off-Normal Tests

One of the intended features of the ANSR design is a capability to reject core heat under natural circulation conditions. Thermal limits such as CHF, FE, 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.

Boiling natural circulation flows are subject to several types of instabilities. The nature of the boiling channel performance is strongly linked to the hydraulic properties of the surrounding flow loop. Therefore, it is important to have an experimental arrangement that accurately simulates the pressure-drop and mass flux performance of the ANSR during natural circulation decay heat removal. The objective of the testing will be to evaluate the core thermal limit behavior under natural circulation conditions.

Two-phase natural circulation (in the piping) may not occur as a result of any of the design basis accidents. Thus, tests examining the two-phase natural circulation performance of the reactor may not need to be extensive since larger uncertainties and conservatisms are permissible in analysis of events with very low probabilities. Realistic quantification of margins associated with the single-phase natural circulation performance of the reactor will be difficult, however, unless the two-phase natural circulation performance and associated thermal limits are understood.

3.4.3.2 Description of the Natural Circulation Off-Normal Tests

The nature of the boiling channel performance is strongly linked to the hydraulic properties of the surrounding flow loop. Therefore, it is important to have an experimental arrangement that accurately simulates the pressure drop and mass flux performance of the ANSR during natural circulation decay heat removal. The most prototypic situation would include a bank of parallel heated channels in a natural circulation loop, complete with a riser above the fuel element. The design of the heater simulating the core becomes critical for these tests as a result of the increased importance of properly simulating heat capacity and thermal response for two-phase conditions. The natural circulation loop developed under WBS 1.1.4.2 will be used, and modifications will be made to incorporate specific requirements for two-phase testing. 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, FE, and CHF behavior.

3.4.4 Low Mass Flux Departure from Nucleate Boiling Tests

3.4.4.1 Justification for the Low-Mass-Flux Departure from Nucleate Boiling Tests

These tests are intended to address the approach to thermal limits under pump coastdown-type and pony motor pumped core flow conditions. Low-pressure, low-mass-flux CHF 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 CHF 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 ANSR exist for low-mass-flux CHF where unsteady positive quality flows may be encountered. Prototypic data are needed to establish thermal limits in these situations.

3.4.4.2 Description of the Low-Mass-Flux DNB 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. The thermal limits will be sensitive to a number of ANSR-specific characteristics that must be accurately represented in the test channel and test methodology design:

- Rewetting of the surface is very sensitive to surface conditions such as the oxide film.
- Thermal diffusivity and heat capacity of the heated wall must also be prototypic in order to obtain the proper wall temperature response.
- Simulation of boiling instabilities (period and amplitude) implies that channel length, heat flux, and power-to-volume ratio should be prototypic.

These experiments will be conducted in the THTL with a full-width test channel and will include effects of the oxide layer on the aluminum wall. Time-average pressure drop and mass flux information from the tests will be coupled with the CHF model to allow a lumped parameter simulation of the fuel assembly performance under the applied power and pressure drop situations. Friction factors at low flow (laminar) in narrow rectangular channels will also be determined.

3.4.5 Full Span Tests

3.4.5.1 Justification for the Full 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 ANSR will influence the pressure drop characteristics of the channel, which will in turn influence the position where FE is initiated. Hot stripe and hot spot conditions must be simulated adequately in core thermal-hydraulic analysis to predict core thermal limits correctly.

3.4.5.2 Description of the Full Span Tests

An experimental test section will be developed to determine the effect of increased span-to-gap ratio. In addition, a test section will be designed to examine both hot spots and hot streaks that may exist in the ANSR core. These include both neutronic peaking and localized peaking due to fuel manufacturing defects. 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.

3.4.5.2.1 High flux heat transfer facility

The full span test section will require significantly higher dc power levels for the test section, as well as higher volumetric flow rates than the current THTL (described previously). Results obtained with regard to the aluminum oxide effects on tests run in the THTL will most likely lead to requirements for aluminum test channels, as well, to properly simulate the surface characteristics. This requirement would dictate the use of high current (>100 kA) and low voltage power supplies. To obtain a bypass flow to test section flow ratio of 20 for FE type tests, a volumetric flow capacity of approximately 55 L/s would be required. The facility would also include the necessary heat exchangers, valves, piping, and instrumentation and controls system in order to provide the desired conditions at the test section.

3.4.6 Integral Transient Tests

3.4.6.1 Justification for the Integral Transient Tests

Transient thermal-hydraulic codes that currently are being used to analyze the ANSR design have not been validated for conditions (e.g., subcooling levels, coolant velocities) typical of the ANSR. Experiments must be developed to validate those codes for safety applications. This experimentation must cover both core behavior and more global or system behavior. An integral test platform is needed to perform RELAP5 benchmark testing for models describing phenomena such as accumulator behavior, pressure wave propagation through the primary system, and pump cavitation/air ingestion, as well as overall system behavior.

Accumulator response characteristics determine the system depressurization behavior as illustrated in Fig. 3.15, which shows FE thermal limit ratios for various accumulator sizes during a loss of coolant accident depressurization event (as predicted by RELAP5 models). The RELAP5 accumulator models need to be validated against prototypic accumulator behavior such as isothermalized performance, interface tracking, and gas evolution.

Upon opening of a break (instantaneous or near instantaneous), a rarefaction wave propagates from the break to the core, lowering core pressure and thermal limits. Pressure wave propagation around the loop after a pipe break will be examined to confirm the times predicted between break initiation and trip of the reactor, as well as the magnitude of the pressure waves as they traverse the piping and core. An example of a pressure transient modeled by RELAP5 for two different break sizes is shown in Fig. 3.16.

As shown in Fig. 3.17, the pump suction pressure falls below net pump suction head (NPSH) requirements during some transients, causing pump cavitation. Air ingestion may occur for breaks to limited volume cells. In order to predict coolant flows during this period, pump cavitation and two-phase characteristics need to be quantified.

3.4.6.2 Description of the Integral Transient Tests

Some small scale transient tests will be conducted in the THTL (described previously) to allow many portions of the RELAP5 code to be exercised simultaneously. A multi-loop integrated controller on the facility will be used to provide rapid variations in the applied power to simulate

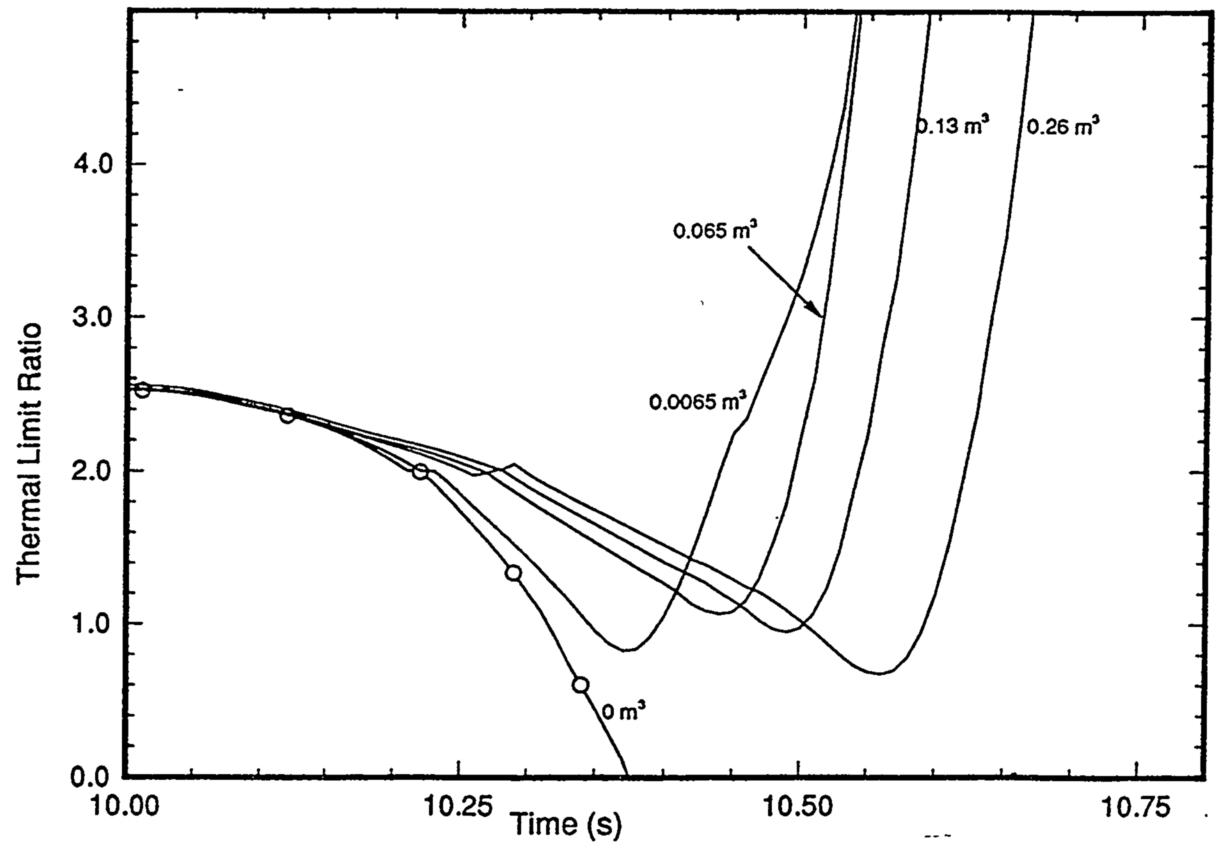


Fig. 3.15. Accumulator response characteristics determine system depressurization behavior.

reactor shutdown. A pump will be used or programmed to provide head characteristics similar to those expected in the reactor main cooling pumps. A break simulator (i.e., rupture disks at two or three locations in the loop) and some additional instrumentation will also be needed. A uniformly heated channel with a coolant bypass will simulate the reactor fuel assembly. The basic T/H phenomena associated with a system piping break and reactor shutdown will be present in this facility. A RELAP5 model of this facility will be developed and used to simulate the response of the system to the imposed transients. The RELAP5 simulation results and the measured transient response of the system can then be directly compared.

3.4.6.2.1 Integral transient test facility

Extension of these small-scale tests will be conducted in a scaled facility (1/5, 1/3, or other), which will be used for integral testing to provide benchmark data for RELAP5 and the dynamic model under ANSR type transients with loop components similar to those of the ANSR. This scaled facility will focus on loop behavior (as opposed to detailed core behavior) and will be used for integral testing to gather pressure and flow response (and other) data. The tests will use a scaled accumulator to evaluate injection rates and accumulator thermal behavior under transient conditions. The experimental testing would require high-speed pressure measurements, including spatial resolution of the wavefront and break flow measurements.

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

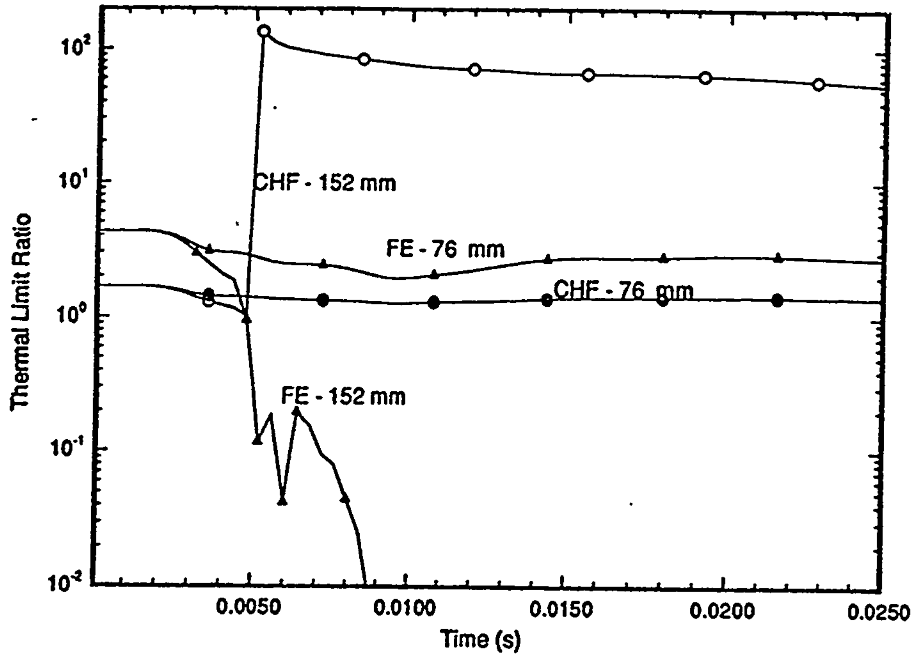


Fig. 3.16. Rarefraction wave causes the FE limit to be exceeded for instantaneous DEGB at core inlet for the 152-mm but not the 76-mm break.

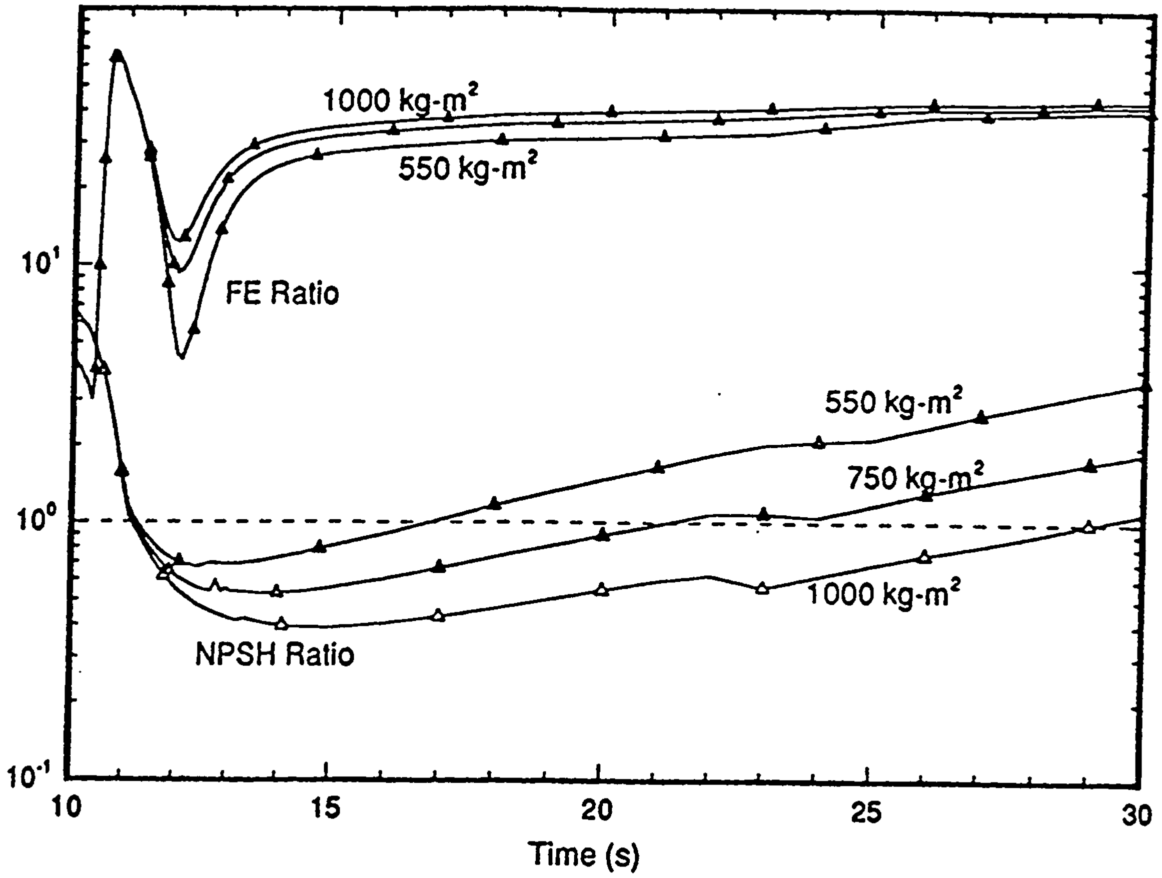


Fig. 3.17. Pump suction pressure falls below NPSH requirements during some transients, causing pump cavitation. Air ingestion may occur for breaks to limited volume cells.

3.4.7 Hydraulic Tests of Nonfuel Components

3.4.7.1 Justification for the 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 under both nominal and off-normal reactor operating conditions. At this time, specific testing needs have not been identified. It is felt, however, that as a more detailed analysis of these components is made, experimental investigation of the thermal and fluid behavior ofsome of these components may be necessary to characterize their performance and operating margins sufficiently. This task has been identified as a potential task; evaluation of its specific requirements is pending.

3.4.7.2 Description of the Hydraulic Tests of Nonfuel Components

Components that may require testing include the outer shutdown rod assembly, hydraulic lines to the outer shutdown rods, tangential thermal beam tubes, transuranium production, pneumatic rabbit tubes for analytical chemistry, isotope production vertical holes, slant irradiation tubes, cold source thimbles, and hot source thimbles. In addition, such items as cooling of the inner control rods, CPBT, and irradiation capsules may need to be experimentally validated. 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.

3.4.7.2.1 Test facility

If tests are needed, it is expected that a dedicated test facility would be required. This test facility would provide basic flow and heat loads over the ranges anticipated for all components to be tested. The facility would comprise a flow loop, power supplies, and associated instrumentation and controls necessary to provide the thermal-hydraulic conditions required. Modular test assemblies for the different geometries of various components would be installed in the flow loop as necessary.

3.4.8 Flow Blockage Tests

3.4.8.1 Justification for the Flow Blockage Tests

Other reactors have been damaged by debris lodged at the inlet to core cooling channels. Such blockage reduces the coolant flow within the channel and may cause fuel damage. The ANS design includes several features to reduce the likelihood of a blockage and to minimize the consequences of any that might occur. The adequacy of these safety features must be assessed as part of the overall reactor safety evaluation. The possibility of a blockage is assessed by a probabilistic examination of potential debris sources upstream of the core inlet and downstream of any inlet strainers that may be deemed necessary. Some finite blockage possibility is expected to remain, and the consequence of this blockage has been reduced by extending the unheated length at the channel inlet. In theory, any perceived flow perturbations would recover within this unheated region, thereby providing adequate cooling to the heated portion of the fuel channel. However, a preliminary appraisal, based upon an extrapolation of available experimental data, indicated that fuel plate damage could still occur for some blockage events. Because the available data were limited and the validity of the extrapolation was questionable, further analysis and experiments were required.

Computational fluid dynamics (CFD) computer codes are well developed and commercially available. An initial attempt to apply one such code to the flow blockage geometry and flow conditions produced unsatisfactory results. An investigation of the causes led to a much improved model and results that are more successful at duplicating the known physics of pertinent flow fields. However, this investigation also revealed the dependence of any numerical analysis on experimental benchmarks. Without such benchmarks, the numerical analysis will give qualitatively correct results, but with an unknowable error band. Therefore, experimental data are necessary to provide a confidence interval for the numerical results. For a given set of benchmarks, the CFD codes can be used to model a broad range of thermal-hydraulic parameters. The numerical models will also be used to model any proposed design modifications resulting from the experimental results.

3.4.8.2 Description of the Flow Blockage Analysis and Testing

Spatial variations in the fuel assembly inlet flow velocity and temperature must be limited to ensure safe operation of the reactor. Debris obstructing the flow channels or upstream structural components provide possible sources of these variations which must be addressed. A combined experimental and computational approach is currently being pursued to define the range of inlet flow velocity and temperature allowable. The flow blockage test facility (FBTF) was constructed to experimentally measure the spatial variation of the heat transfer characteristics downstream of an inlet velocity perturbation in order to establish the minimum blockage size at the fuel assembly inlet that will result in fuel damage. A CFD code will be used to model the experimental test apparatus for validation purposes and, with the expansion of the model, will be used later in the analysis of the ANSR.

3.4.8.2.1 Status

Based on the limited availability of experimental data for the flow conditions expected within the ANS core, and the dependence of the numerical analysis on experimental benchmarks, the FBTF was established. The hub of the test facility is a channel designed to mimic ANS core region flow channels with a span and gap of 80 and 1.27 mm, respectively, and a total channel length of 507 mm. The inlet to the channel was designed to permit the insertion of blockages ranging from 4 to 40 mm and located in both edge and central positions with respect to the channel span. A bypass line was provided to permit variations in flow rate through the channel. A schematic drawing of the facility is shown in Fig. 3.18. Construction of the FBTF, completed in FY 1993, included installation of the test section assembly, electrical utilities, instrumentation, and piping.

Experimental measurements include bulk average temperatures at the inlet and outlet of the test section, fluid flow rates, and upstream and downstream pressures. The flow separation and rotational behavior behind a blockage would be changed by the introduction of any kind of pressure-sensing instrumentation. Therefore, indirect methods of measuring the fluid velocity vectors and pressure fields are used. Figure 3.19 shows a heater strip that covers ~30 mm of the channel's length. One side of the heater strip is exposed to the coolant flow through the channel. The back of the heater is covered with thermochromic crystals. These crystals change color in response to temperature. Because the heater strip is thin and highly conductive, the temperature pattern on the back side can be used to infer the heat transfer coefficients on the channel side. High-resolution optics are used to record the color pattern. A color-to-temperature calibration is made for an unblocked channel with quasi-steady-state temperature conditions. This calibration is then used to transform the recorded color pattern to a temperature field record. The power to the heater strip is adjusted for each test to maximize the color change across the heater face. The measured electrical current for each test is used with the known heater resistance to calculate the average heat flux from the heater face into the channel. (A detailed analysis of the temperature dependence of the heater's electrical resistivity

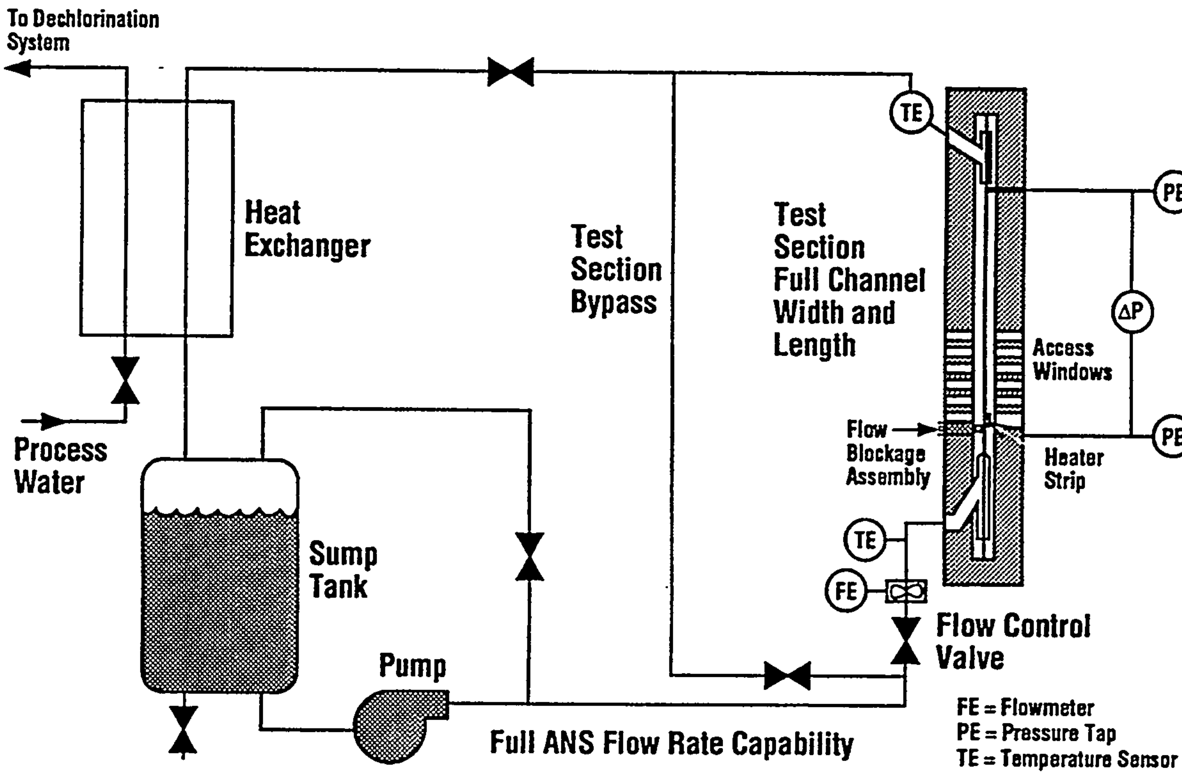


Fig. 3.18. Schematic diagram of the flow blockage test facility.

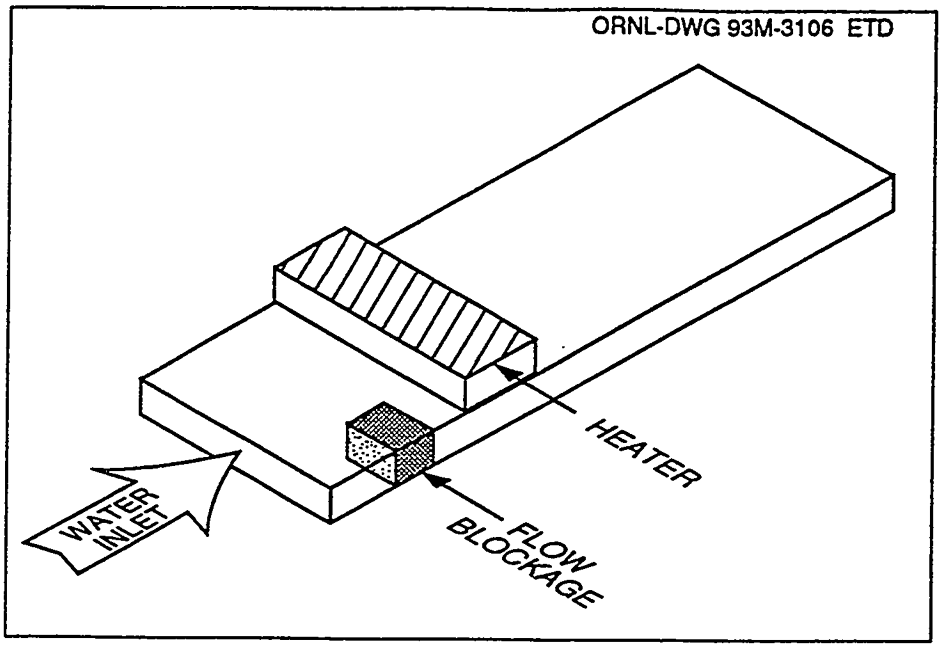


Fig. 3.19. Schematic diagram of the blockage and diagnostic heater arrangements for the flow blockage test assembly.

showed that the local heat flux will not vary from the average value by more than a few percent). The total heat flux is small enough that there is no measurable increase in the temperature of the water flowing past the heater. Given constant temperature fluid, the average heat flux can be used to calculate heat transfer coefficients from the recorded temperature fields.

These temperature fields can only be used to measure the channel wall heat transfer over a relatively small region. A larger heater surface would not offer the same uniformity of heat flux and would therefore produce less accurate experimental data. Therefore, another method of measuring the fluid behavior has been included in the FBTF design. A laser doppler velocimeter will be used to measure the flow fields over an extended length of the channel. Local fluid velocities will then be used to infer heat transfer characteristics on the channel walls. Within the region covered by the heater, the results from the two measurement devices will be compared. The CFD modeling results will also be compared with the body of empirical data for the channel. These comparisons should improve our understanding of the physical phenomena and increase confidence in the overall safety analysis of the fuel channel design.

Primary tests on the FBTF will characterize the flow behavior within the channel behind a variety of blockages under ANS hydraulic operational conditions. For the primary tests, the pressure drop across the channel will be held constant, reflecting operational conditions that would occur during a flow blockage event in the ANSR core. The total flow through the channel will be allowed to vary and will be much reduced for the larger blockages. These tests will provide a direct relationship between the blockage size and span-wise location and the unheated entrance length necessary to ensure safe operation. The primary test results will also be used to benchmark the CFD models.

Secondary tests will expand the range of hydraulic conditions and will be used predominately to benchmark a more extensive range for acceptable CFD models. These tests will include lower flow rates and variable pressure drops through the channel for the same blockage sizes and positions used in the primary tests. Experiments are planned to establish the type of blockage necessary to initiate fuel damage.

3.4.9 Thermal-Hydraulic Testing for Refueling

3.4.9.1 Justification for Thermal-Hydraulic Testing for Refueling

These tests are needed to determine core thermal limits applicable to the fuel element transport process during refueling. Existing data and models are not adequate to define CHF limits in the ANSR fuel element geometry under low-pressure natural circulation conditions that will exist during fuel transfer. This applies when the elements are in the normal vertical orientation or in an off-normal horizontal orientation. Potential refueling accidents include dropping a fuel element, which could ultimately result in the element's lying horizontally on the pool floor.

3.4.9.2 Description of Thermal-Hydraulic Testing for Refueling

Plans are being made to define specific tests needed to develop an adequate thermal limit data base and allow model development for these conditions. This test series is also expected to use the natural convection test facility instrumentation system, developed under WBS 1.1.4.2 and modified to address conditions expected during refueling. As a result, this testing will occur as a follow-on to the natural circulation testing described previously (as shown in Table 3.10). A separate pool and associated hardware may be required to provide simulation of fuel element transport conditions. These experiments will address natural convection cooling capability during movement of the fuel elements.

3.4.10 Full Flow Hydraulic Tests

3.4.10.1 Justification for the Full Flow Hydraulic Tests

Testing under full flow hydraulic conditions is needed to address the questions of whether the orificing and structural designs introduce the desired flow characteristics in a very complicated geometry and whether there are design features that may cause flow-induced structural behavior. The present ANSR core region design includes ten separate orifices to distribute the flow properly. Thermal limits within the core and nonfuel components are dictated by these flows. These tests will therefore examine issues associated with integrated operation of a complete core assembly under ANSR design flow conditions. Although it would be possible to perform some of this testing during start-up of the reactor, problems discovered at that point would have very serious impacts on both cost and schedule.

3.4.10.2 Description of the Full Flow/Full Scale Hydraulic Test Facility

It is expected that a dedicated test facility will be required to perform these tests. This test facility will include pump(s), heat exchangers, and associated piping systems required to obtain full flow and pressure conditions typical of ANSR normal operating conditions. The facility would also be capable of addressing structural interactions and flow distribution effects under some anticipated transient conditions such as transition to pony motor flow. The facility will be capable of hydraulic testing of an actual core assembly (or hardware mockup). Included will be measurement and evaluation of flow distribution between fuel elements and other core components, measurement of pressure distributions within the core region, and evaluation of component stability and vibration under full flow conditions. In addition, measurement of flow patterns established at the core inlet as a result of structural components and evaluation of these effects on fuel channel flow distributions will be made.

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 core development activity. The work includes the reactor physics analyses necessary to develop the concept for reactivity control and to perform a detailed analysis of the concept.

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 3.12, and it will be performed at INEL, the Massachusetts Institute of Technology (MIT), and ORNL. The total estimated costs over the 8-year period covered by this R&D plan are given in Table 3.13, and the associated schedules are shown in Fig. 3.20. 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.

Table 3.12. Summary description of the control concepts work breakdown structure level four tasks

WBS	Task description
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 ANS. 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 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. This task will provide the analytical evaluations necessary to identify reactivity control system poison materials and characterize their performance in an ANS neutron and temperature environment. Without this R&D activity, the ability to control the nuclear reaction during normal operation cannot be established.

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 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 shutdown system have been identified, the poison material must be chosen. Issues to be addressed for the material selection include reactivity worth, burnup rate, behavior of daughter products, nuclear heat deposition during operation, decay heat during shutdown and low-flow conditions, fabrication techniques, mechanical properties, and corrosion resistance. The reactor physics issues will be addressed under this task while remaining issues will be examined under other elements of the R&D program.

Much of the work under this subtask has been completed and conceptual designs for the inner control system, reflector shutdown system, and burnable poison system have been established. There are, however, additional activities described below that must be performed under this subtask.

Table 3.13. WBS level four breakdown of costs for the control concepts activity

WBS Level 3	Level 4	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.5		Control Concepts										
	1.1.5.1	Static analysis of control and safety poison systems	Exp.	69	61	61						191
			Line				31	31	31	45		138
	1.1.5.2	Reactor kinetics	Exp.	37	61	61						159
			Line				31	31	31	51		144
		Subtotals	Exp.	106	122	122	0	0	0	0	0	350
			Line	0	0	0	62	62	62	96	0	282
		Contingency	Exp.	5	12	12						29
			Line				12	12	12	19		55
		Total	Exp.	111	134	134	0	0	0	0	0	379
			Line	0	0	0	74	74	74	115	0	337

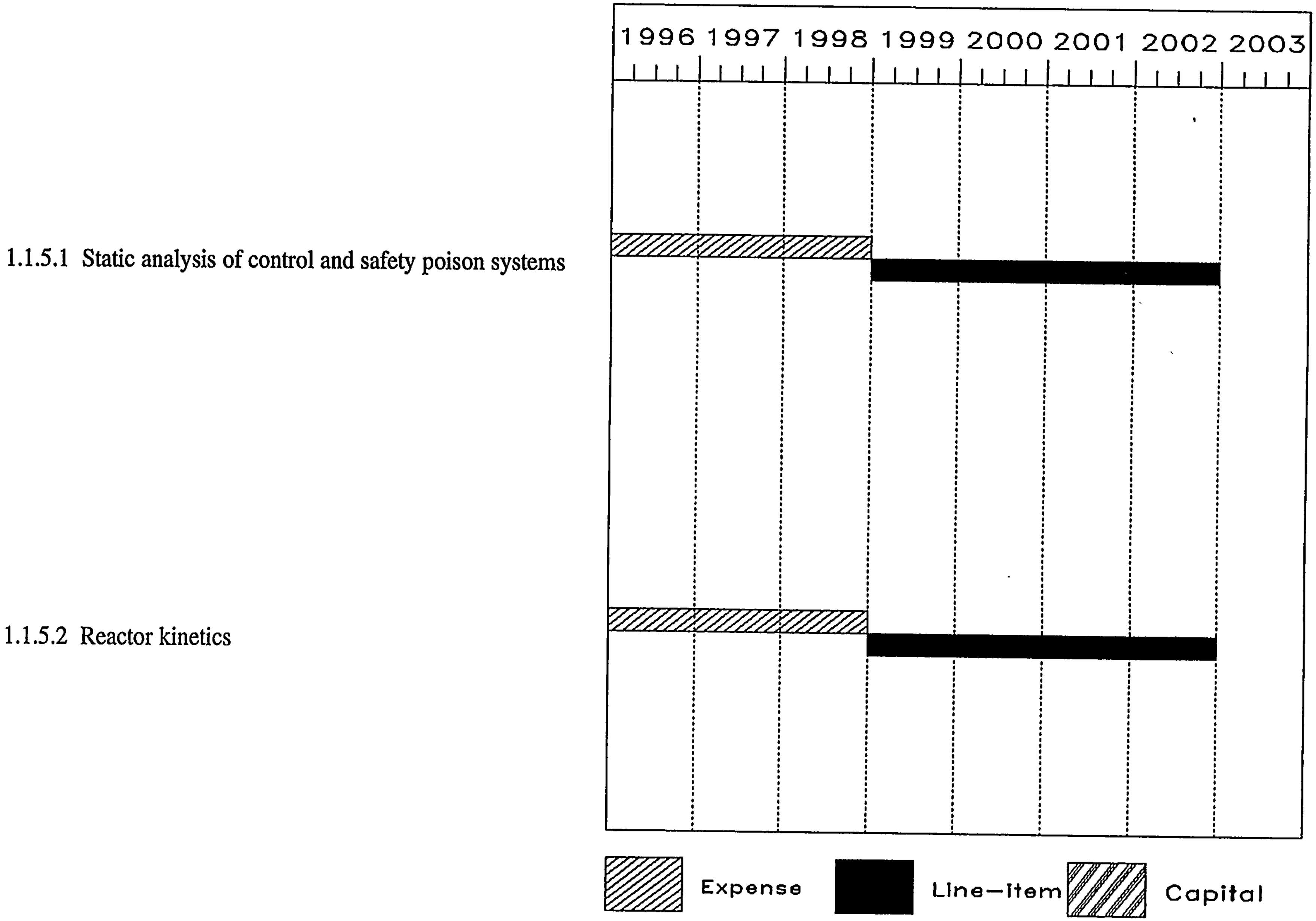


Fig. 3.20. Schedule for WBS 1.1.5 control concepts.

3.5.1.2.1 Determination of safety shutdown system locations

The present location of the safety shutdown rods in the heavy water reflector is such that the required shutdown margin and reactivity insertion rates can be obtained while maintaining minimal neutron loss during normal operations. In addition, care must be taken to avoid spatial conflicts with beam tubes and other experimental facilities. Therefore, the location of the shutdown rods must be examined whenever there is a change in the core and reflector component geometry.

3.5.1.2.2 Development of control system worth curves

Calculations of the change in core reactivity as a function of the control rod location at all points throughout the fuel cycle must be known to ensure adequate control and shutdown capability. The worth curves also serve as input to other system models to determine the required control rod insertion rates and reactor kinetics response. Calculations of the control rod worth curves can be performed with 3-D, explicitly modeled control rods using the MCNP and KENO Monte Carlo codes. Two-dimensional, equivalent models are used in the VENTURE diffusion theory and DORT transport theory codes. Currently, the central control rod worth curves are calculated throughout the fuel cycle using the VENTURE model and at beginning-of-cycle using the MCNP model. Future methods development will allow the use of MCNP and DORT in the evaluation of control rod worth curves throughout the fuel cycle.

3.5.1.2.3 Determination of control systems lifetimes, activation, and transmutation productions

The depletion aspects of the central control rods must be determined to determine the frequency at which the rods must be replaced and the changes of control rod worth curves over several fuel cycles. In addition, activation rates must be determined to ensure that the rods are properly handled upon removal from the reactor core and transmutation products must be examined to evaluate material properties and decay heat. The use of the Monte Carlo depletion strategy briefly discussed under WBS 1.1.1.1 and DORT transport theory depletion are the most appropriate methods for determining the control rod burnup and activation.

3.5.1.2.4 Examination of improved burnable absorber systems

In the current design, boron carbide burnable poison is located in the fuel element endcaps as a means of controlling the core reactivity at beginning-of-cycle. Other burnable absorber materials and locations, such as the filler region as used in HFIR, will be examined as a possible means of providing better reactivity control over the fuel cycle.

3.5.1.2.5 Determination of the impact of control system vibrations on core reactivity

The central controls are cooled using a high velocity flow, which may induce vibrations. The vibrational movement of the control materials may lead to unacceptable changes in the core reactivity. Effects of the magnitudes and frequencies of the control rod vibrations on core reactivity will be examined to ensure that undesirable oscillations in core power do not occur.

3.5.1.2.6 Evaluation of minimum shutdown requirements

The ANS control system is required to meet several minimum shutdown requirements. These include maintaining a multiplication factor below 0.95 independently when the central control rods and the safety shutdown rods are at the worst point in the fuel cycle and the highest worth rod of the group is stuck at its position when the scram signal was initiated.

3.5.1.2.7 Quality assurance and documentation

Appropriate QA plans and documentation will be developed to ensure that modeling and computer code errors are avoided.

3.5.2 Reactor Kinetics Analyses

3.5.2.1 Justification of the Reactor Kinetics Analysis

The analysis of the ANSR in accident scenarios, such as a control rod ejection, and in operational transients, such as startup and shutdown, requires a kinetics analysis capability. Most of the transient situations cannot be determined experimentally and therefore can be analyzed only through the use of analytic models. Furthermore, since the ANS has a high-reactivity core with large power densities, determinating appropriate and accurate analysis methods is extremely important.

3.5.2.2 Description of the Reactor Kinetics Analyses Task

The major reactor kinetics analyses tasks are as follows.

3.5.2.2.1 Determination of appropriate kinetics methods

Since the ANS is unlike any other reactor currently in operation, the appropriate kinetics analysis method must be determined. These methods include the comparison of points kinetics and space-time methods. Currently, a comparison of the use of the point kinetics equations (with reactivity determined using an adiabatic approximation) with a full space-time treatment is being performed at MIT. The space-time method under development provides a means of incorporating neutron transport effects, if necessary. Pending the results of the MIT study, additional work may be required to incorporate the developed space-time methodologies into the present design capabilities.

3.5.2.2.2 Determination of kinetics parameters throughout fuel cycle

The reactor kinetics parameters, reactivity, prompt neutron lifetime, and effective delay neutron fractions are used as input to ANS system models and used to determine the response of the ANS control systems. This subtask includes the determination of these parameters throughout the fuel cycle. Since a heavy water coolant and reflector is used in the ANS, these parameters must include the effects of photoneutron production.

3.5.2.2.3 Quality assurance and documentation

Appropriate QA plans and documentation will be developed to ensure that modeling and computer code errors are avoided.

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 physics 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 performing 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 2000. This schedule allows at least 3 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 3.14. Most of the work under this activity would be performed at an as-yet-undefined location (the facility chosen to perform the critical experiments), and some analysis work would be performed at INEL and ORNL. The total estimated costs (all expense money) for this activity over the 8-year period covered by this R&D plan are given in Table 3.15, and the associated schedules are shown in Fig. 3.21. 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 are to be measured, along with the experimental conditions, must be known before a site for the experiments can legitimately be discussed. 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 might be made as part of the critical experiments will be prepared under this task. The measurements on the list will then 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. Proposed sites will be discussed with the staff to determine the feasibility, practicality, and costs of performing experiments.

After a site has been selected, a detailed 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 1997 as the reference plan.

Table 3.14. 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, development of detailed cost estimates of facility modifications required to conduct experiments, identification of experimental parameters to be measured, formation of team for completing the critical experiments task, and development of 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 support the safety analysis of the critical experiments, 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 the critical experiments and measure physical parameters necessary to confirm the ANS core physics 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.

3.6.1.2.1 Status

A detailed list of measurements that might be made as part of the ANS critical experiments program has been compiled. These potential tests have been divided into sixteen technical areas described in the following paragraphs.

1. **Critical configurations.** Establish control rod critical configurations analogous to beginning of life (BOL) with and without mock-ups of reactor experimental facilities, such as beam tubes and irradiation facilities.
2. **Relative fission density distributions.** Measure the radial and axial fission density distribution in both the upper and the lower fuel element (± 2%). This would be done at a variety of radial and axial locations for three upper and three lower removable plates with a 3-mm spatial resolution for the measurements. These measurements would be performed by gamma ray scanning of the special removable plates. The mechanical details of these tests will be worked out between the ANS Project staff and the critical facility personnel. Measurements will be performed for the BOL configuration only. It may also be useful to study the effects of

Table 3.15. WBS level four breakdown of costs for corrosion tests and analysis

WBS Level 3	Level 4	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.6		Critical Experiments										
	1.1.6.1	Develop experimental plan for crital experiments	Exp.	51	138							189
	1.1.6.2	Preanalysis of critical experiments	Exp.	209	52							261
	1.1.6.3	Modification of facilities for critical experiments	Exp.		1863	1770						3633
	1.1.6.4	Perform critical experiments	Exp.				1898	1022				2920
	1.1.6.5	Analysis of critical experiments	Exp.				254	254				508
		Subtotals	Exp.	260	2053	1770	2152	1276	0	0	0	7511
		Contingency	Exp.	13	205	177	215	128				738
		Total	Total	273	2258	1947	2367	1404	0	0	0	8249

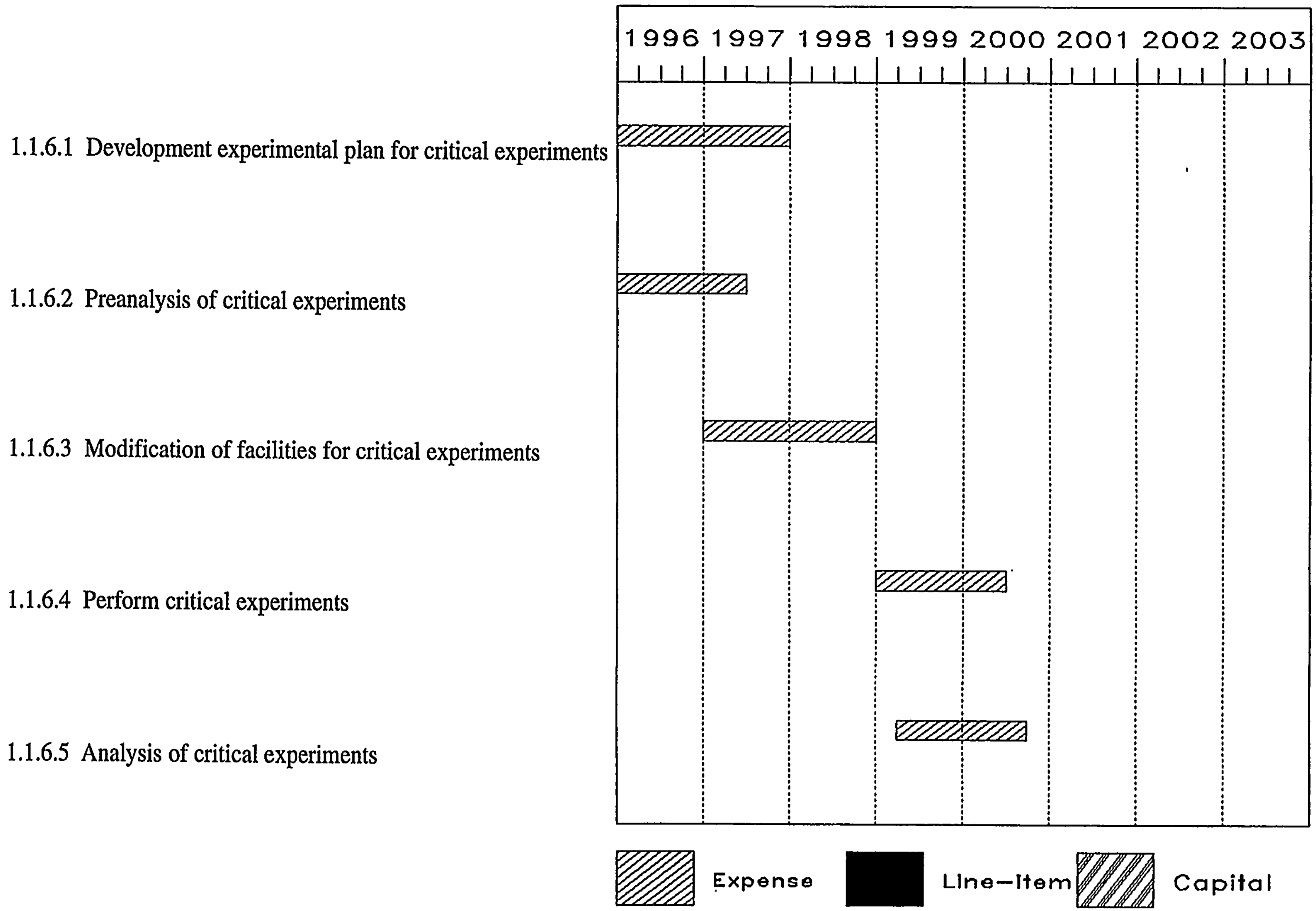


Fig. 3.21. Schedule for WBS 1.1.6 corrosion tests and analysis.

mockups of special experiments (such as an irradiation target) on the fission density distributions.

3. **Absolute fission density**. These measurements would be made at selected points in the core to an accuracy of ±5%.
4. **Control rod calibrations**. ANS control rods will be calibrated as a function of position for the BOL and end-of-life (EOL) configurations by the reactor period method and the inverse kinetics rod drop (IKRD) method. The ^{252}Cf source-driven noise analysis, break frequency noise analysis (BFNA) and modified source multiplication (MSM) methods may also be used to determine the total worth. The EOL reactivity level will be simulated by placing poison in the fuel elements.
5. **Safety rod calibration**. ANS safety rods will be calibrated as a function of position for the BOL configuration and the simulated EOL condition by the reactor period method and the IKRD method. As in the case of the control rod calibration tests, the ^{252}Cf source-driven noise analysis, BFNA, and MSM methods may be used to determine the total worth.
6. **Temperature coefficient of reactivity**. This parameter will be measured at three uniform temperatures between room temperature and 80°C (e.g., 30, 60, and 80°C for BOL configuration).
7. **Reactivity coefficients**. The reactivity associated with selected samples of materials such as fuel plate worths and void coefficients will be measured (the latter by use of expanded plastics or some other device to simulate voids). A total of 20 to 25 test configurations are anticipated for this category.
8. **Neutron spectrum**. The neutron spectrum at various locations in the reflector and central hole region of the core is very important to experiment facility design. Therefore, measurement of spectrum is proposed for the critical facility plan. The ratio of fast flux (>0.1 MeV) to the thermal flux (<0.625 eV) is required to <10% accuracy at approximately 5 locations. The ratio of the epithermal flux (0.625 eV to 100 eV) to the thermal flux at the isotope production facility positions (approximately three locations) is also required to <10% accuracy.
9. **Flux per fission**. Measurement of absolute neutron thermal flux at the peak reflector flux position per core fission is needed. Absolute flux measurements should be performed to an accuracy of <3% using coincidence counting techniques or other suitable methods.
10. **Gamma ray measurements**. The gamma ray to neutron flux is of interest to experiments. Thegamma ray energy deposition should be measured to <±1% accuracy as a function of position in the D_2O tank and in simulated beam tube locations. The gamma ray flux and energy distribution should also be measured at these locations. These measurements will also be important for use in confirming heat loads for components in the D_2O reflector tank and for experiment design.
11. **Instrumentation and control measurements**. The purpose of these measurements is to verify that the start-up and other instrumentation will perform as required. These will be planned by the project staff. Additional detectors required to perform these measurements will be supplied by the project, if necessary.
12. **Approach to critical**. A reverse approach to critical experiment will be performed to evaluate methods to monitor the initial ANSR start-up. This experiment will include control and shutdown rod withdrawal, which is as prototypic as practical. Techniques such as MSM, BFNA, and ^{252}Cf source-driven noise method would be used for monitoring.
13. **Handling of fuel outside of reactor**. Experiments simulating how fuel is handled outside the reactor will be performed. Subcriticality of each fuel element submerged in light water will be measured. These tests will confirm absorber/isolator (supplied by ORNL) design to ensure safe fuel loading of ANS. The methods used will include the ^{252}Cf source-driven noise analysis. One of the purposes of this test will be to examine the applicability of the ^{252}Cf noise method for

QA testing of all future core elements. An additional tank with a minimum of 305 mm of external reflection may be necessary to perform these experiments. In addition, subcriticality will be measured for each fuel element in heavy water with absorbers/isolators to establish standards for monitoring the loading of the fuel in the ANS.

14. **Light water contamination.** The measurement of the partial insertion of light water up to 20% into the core is desirable. Separate measurements of the effect of light water in three separate regions (central region, inner fuel annulus, and outer fuel annulus) would be requested, but combination measurement is acceptable if separate effect measurements are prohibited. Control rod worths will also be measured under these conditions to determine impacts on their worth curves. No measurements of light water contamination of the reflector water are expected to be required.
15. **Reactivity worths.** Reactivity worths of various mockups of experiments will be measured, including off-normal conditions such as flooded beam tubes. It is anticipated that approximately 25 component reactivity worth measurements will be required.
16. **Kinetic experiments.** Various experiments will be performed to evaluate the kinetic parameters of the system. The prompt neutron decay constant will be obtained from BFNA with the value at delayed criticality being equal to β_{eff}/l. The neutron importance distribution could be measured by moving a small ^{252}Cf source within the system.

Initial discussions have been held with organizations at five potential experiment site locations. The discussions indicate that most if not all of these tests can be performed within the budget allocated for this task.

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 major intentions of the critical experiment program 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 preanalyzed to determine the practicality of the test.
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 input to these safety analyses.

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 analysts and 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 start-up tests cover areas such as subcriticality monitoring, reactivity measurements during refueling, and simulation of start-up.
2. Reactor criticality measurements include criticality measurements at BOL and possibly simulated EOL configurations.
3. Tests for direct validation of calculational methods include identification of critical rod configuration at BOL, reactivity coefficients, relative fission density distribution, neutron spectrum, light water contamination, and kinetics parameters.
4. Reactor performance measurements include neutron and gamma ray fluxes at various experimental facility locations per fission in the core.
5. Reactor operation experiments include 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 documenting 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 ANALYSES—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 tests, and analyses 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 September 1997. 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 1998. 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 1999. 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 analyses activity is divided into the seven WBS level four tasks summarized in Table 3.16. 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 8-year period covered by this R&D plan are given in Table 3.17, and the associated schedules are shown in Fig. 3.22. 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

Table 3.16. 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 analysis—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	Structural dynamic 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 CPBT. Experimental studies are planned 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 focus of this work will be to determine 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	Materials issues—This task addresses materials related to design needs such as *ASME Boiler and Pressure Vessel Code* approval of aluminum and development of fracture mechanics for aluminum.
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.

Table 3.17. WBS level four breakdown of costs for the materials data, structural tests, and analysis activity

WBS Level 3	Level 4	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.7		Material Data, Structural Test, And Analysis										
	1.1.7.1	Design support	Exp.									0
			Line	68	270	270	270	270	105	50		1303
			Cap.		45			45				90
	1.1.7.2	Fuel plate stability test and analysis	Exp.	535	613	533	192					1873
			Line									0
			Cap.		414	799	150					1363
	1.1.7.3	Fuel plate analysis	Exp.	193	97	97						387
			Line									0
			Cap.									0
	1.1.7.4	Structural dynamic tests and analysis	Exp.	1117	2488	951	352	65	30			5003
			Line			654	654	327				1635
			Cap.	1000	464		424	108				1996
	1.1.7.5	Material properties	Exp.	1300	2022	1108	748	147	107	32		5464
			Line									0
			Cap.									0
	1.1.7.6	Materials issues	Exp.	350	868	801	551	100	81			2751
			Line									0
			Cap.									0
	1.1.7.7	Material properties data base	Exp.									0
			Line	97	211	117	117	117	105			764
			Cap.									0
		Subtotals	Exp.	3495	6088	3490	1843	312	218	32	0	15478
			Line	165	481	1041	1041	714	210	50	0	3702
			Cap.	1000	923	799	574	153	0	0	0	3449
		Contingency										
			Exp.	175	609	349	184	31	22	3	0	1373
			Line	8	96	208	208	143	42	10	0	715
			Cap.	100	185	160	115	31	0	0	0	591
		Total	Exp.	3670	6697	3839	2027	343	240	35	0	16851
			Line	173	577	1249	1249	857	252	60	0	4417
			Cap.	1100	1108	959	689	184	0	0	0	4040

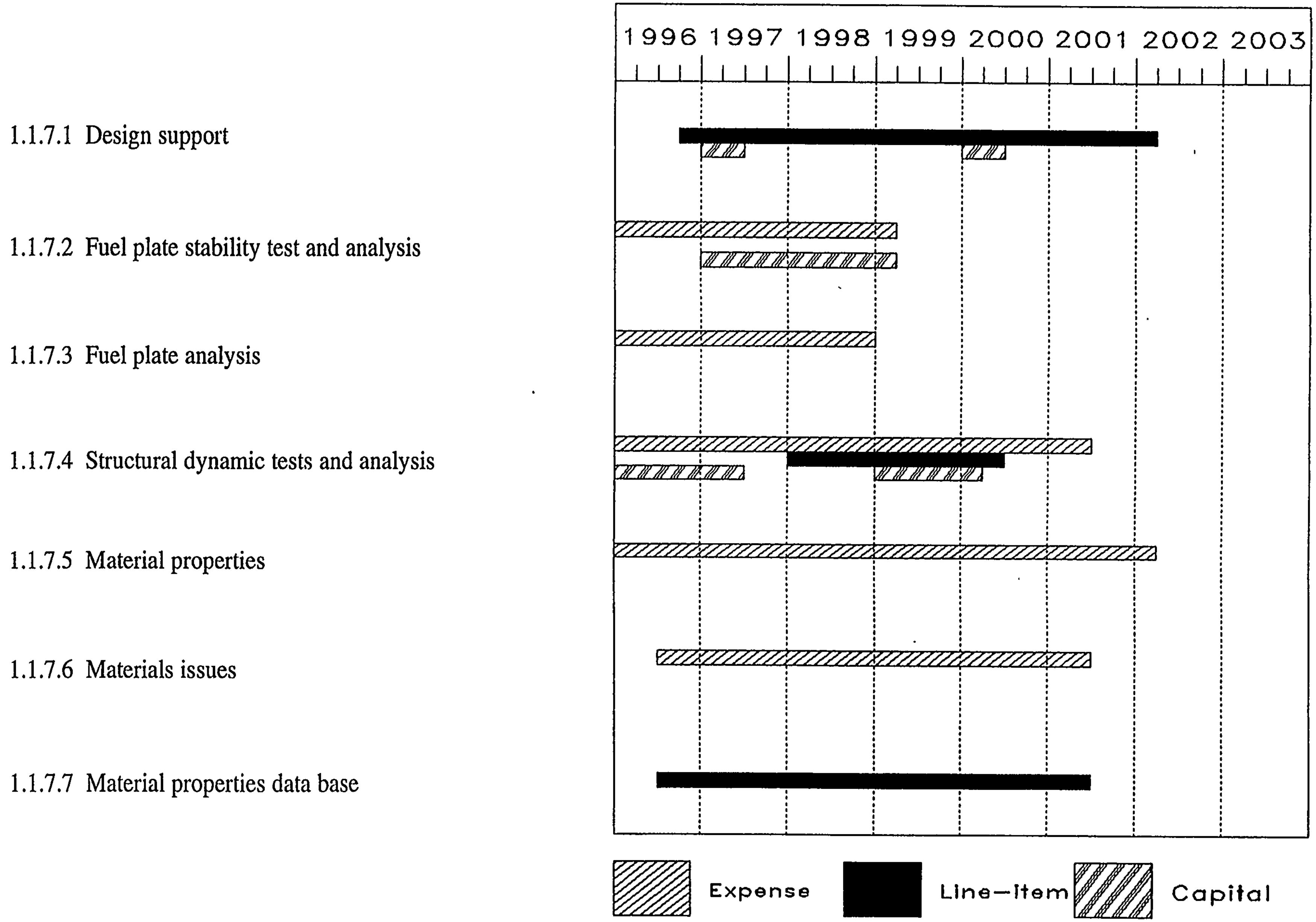


Fig. 3.22. Schedule for WBS material data, structural tests, and analyses.

the work is in direct support of the Title I or Title II design activities. Subsections 3.7.1 through 3.7.7 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 Task

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. This task provides design support as needed to develop these models.

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 such as the reflector vessel head and nozzles that require structural analyses. Input for the "design report" will be provided for "code-stamped" components. Assistance also will be provided in the preparation of design specifications and review of design reports required by the *ASME Boiler and Pressure Vessel Code.*

3.7.2 Fuel Plate Stability Tests and Analyses

3.7.2.1 Justification for the Fuel Plate Stability Tests and Analyses Task

Closely spaced arrays of fuel plates cooled by water flowing through the channels between the fuel plates have long been used in research reactors. Since the work of Stromquist and Sisman in 1948,[34] it has been known that very high flow velocities past fuel plates can cause the plates to deform and fail. 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 be allowed to impede the flow of coolant through the reactor core. Fuel plate stability tests and analytical model analyses must be performed to ensure that excessive plate deflection will not occur in the ANS. 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 ANSR 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.[35] 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 addition to their involvement in the Miller instability phenomenon.[36]

For nonisothermal plate stability analysis,[37] ABAQUS[38] 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.

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 under this WBS. However, the flow rates and pressures required to reach plate instability are large for aluminum plates, and thus a major test facility is required to perform these tests. In the early stages of this task, it has been determined that data from plastic (epoxy) plate tests can be used to extrapolate to aluminum plate behavior. Although this is certainly not a substitute for the aluminum plate tests to be performed later, the lower modulus of elasticity of epoxy plates (compared with aluminum) reduces the critical velocity so that early data can be obtained with modest flow rates and pressures that do not require a major facility.

3.7.2.2.1 Status

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.[36] No instability was observed during the test.

The facility shown in Fig. 3.23 has been used to conduct tests on arrays of five plates for both the upper and lower fuel elements to examine their structural response to coolant flow.[39,40] 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. One of the objectives of this experimental effort was to examine the capability of the Sartory model to predict the plate response to coolant flow. In comparing the experimental data with the calculated collapse velocity from the Sartory model,[34] it is apparent that they do not correlate. Relative to a collapse velocity for the ANS plates, the experimental data imply that a collapse velocity does not occur within a velocity range of more than twice the planned operating flow velocity of the ANS.

Another very important observation from the experimental data is that the plate deflection is a function of the pressure difference across the plate, which in turn is a function of the flow velocity. This observation is important because, if the pressure difference can be determined, the plate response can be predicted.[41] At the operating velocity of the ANS, the maximum plate deflection determined from the tests is 6% of the channel opening. This does not mean that the channel cross-sectional area decreases by 6%. The plate deflects in an "S" shape from its unloaded involute shape; that is, part of the plate deflects in one direction and part of the plate deflects in the opposite direction as pressure is applied. Because of this deformed shape, the cross-sectional area changes less than the percentage change of the maximum deflection.

The next series of tests will be performed during FY 1995 with arrays of five aluminum plates. These tests will focus on (1) determining flow-induced deflection of dummy aluminum fuel plates and (2) investigating flutter of dummy aluminum fuel 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, and for the adequacy in fabrication of each component. 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 required for these tests; testing is scheduled to begin in FY 2000.

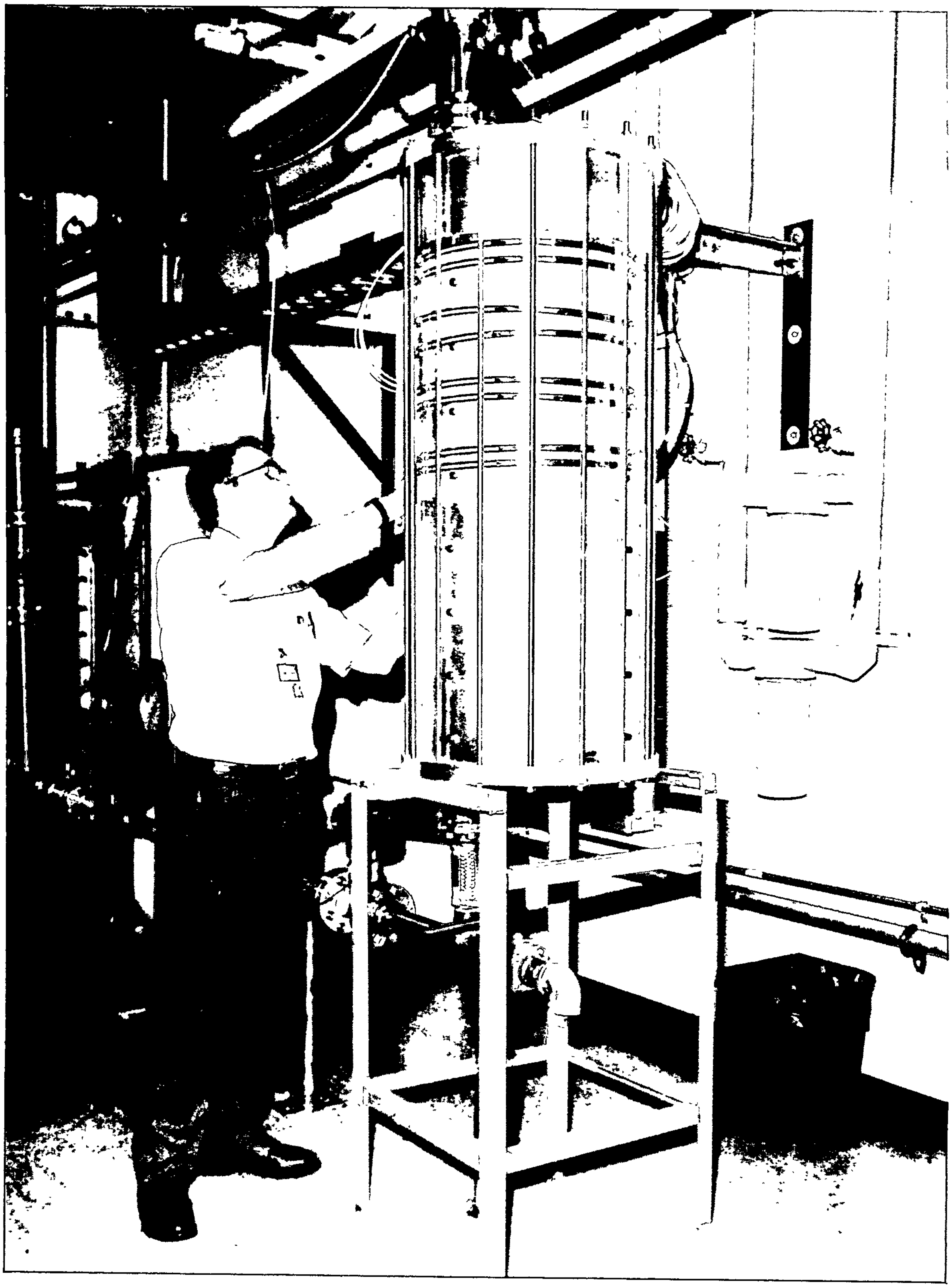


Fig. 3.23. ANS multiplate flow stability test facility.

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 is accommodated by allowing one of the support cylinders to rotate freely. There is no such mechanism for accommodating the axial differential expansion. The yearly indication is that some ANS involute fuel plate distortion due to this thermal expansion mismatch is probable. If the deformation excessively reduces the width of the coolant gap between two adjacent plates, heat removal would be impeded, and the fuel plates could overheat and be damaged. Therefore, the changes in gap must be determined and provided as input to the thermal hydraulic analyses. This task provides the analytical models and their validation through analysis of benchmark tests.

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

For structural analysis, ABAQUS,[38] a widely used and highly respected general-purpose finite element structural analysis computer program, is 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 are used to determine maximum deflection due to thermal expansion. These analyses are then used to determine the maximum change in gap between plates.

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 Structural Dynamic Tests and Analyses

3.7.4.1 Justification for Structural Dynamic 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 could affect 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. Substantial vibration of the control elements could cause contact between these components and the core inner cylinder. This contact would cause wear and could result in a failure to scram or impurities in the coolant flow stream. 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, for flow-induced vibration (FIV) large enough to be of concern. The requirement that sections be kept thin to limit the temperature resulting from nuclear heat deposited in them makes vibration more likely. Components of particular concern are the fuel elements, the inner control elements, the shutdown rods, and the CPBT. Without this task, the extent of vibration of important components and its consequences would not be known.

3.7.4.2 Description of the Structural Dynamic 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 using the NASTRAN[42] computer code is in progress or is planned to refine the current FIV estimates. The first refinement made was the added hydro-elastic 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 was included in our modal calculations. Alternate support configurations at the top of the control elements were studied.

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. These results will be directly correlated with measured values from the control element test facility (CETF) 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 directly predict the response of the structure, but will provide valuable design guidance and basic knowledge of the control element structural 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 CETF forced-flow test.

Scram dynamics are being simulated throughout the entire 1200-mm travel of the control rod. This simulation produces a time-history response of the rod mass that can be used for structural design purposes and that provides a direct measure of scram performance. The simulation accounts for all the forces acting on the rod during scram, including the velocity-dependent fluid forces. These shear, buoyancy, and pressure differential forces are described in ref. 43 along with typical acceleration, velocity, and displacement time histories of the baseline configuration. These simulations are essential to the specification and understanding of the springs and damper system required to give the desired scram performance. At the same time, they prevent excessive structural loads on the system that might buckle or overstress the rod assembly. The simulations will provide immediate insight into the controlling parameters that determine scram performance.

One major test facility is expected to be needed to complete this task. This test facility will be used to perform tests that will confirm control rod mechanical behavior. The tests and the required test facility are described below.

In addition to the control element tests, it is perceived that it is necessary to compare the plates' natural frequencies with potential exciting frequencies of the reactor system. If these frequencies match, then some modifications will be needed to avoid possible plate failure. 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, are already available for use in these tests.

Control element test facility—The CETF will be used to benchmark analytical models and to provide final confirmation of control element FIV characteristics. The facility will also be used to establish assembly/disassembly procedures, to determine operational characteristics, and to determine operational reliability. Figure 3.24 shows a schematic of the CETF based on the conceptual design configuration.

Recent calculations have indicated that a major revision to the baseline design is required to prevent the possibility that a broken rod might be ejected from the core because of flow forces. The 0.9-MPa (134-psi) pressure drop across the CETF must be reduced to as low as 0.3 MPa (43.5 psi), or the drag effect of the flow on the rod correspondingly reduced, or the rod mass increased. Consequently, further study of the control element performance requirements will be carried out before detailed design of the CETF.

A full set of three control rod assemblies will be tested 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 on 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 conceptual design phase has begun; testing is scheduled to begin in FY 1996.

3.7.5 Material Properties Evaluations

3.7.5.1 Justification for the Material Properties Evaluations Task

The extremely high neutron and gamma flux in ANS means that certain components will be subjected to high neutron fluences concurrent with high temperatures. It is important to know the effect of this high fluence (including rate effects) on the various materials used in the ANSR. Embrittlement is an especially important characteristic that must be understood. Without material properties evaluations, the lifetime limits of various components cannot 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 the 6061-T651 aluminum alloy, which has been selected for the CPBT, the reflector vessel, and possibly the cold source as well. Capsules will be irradiated in the HFIR to study the

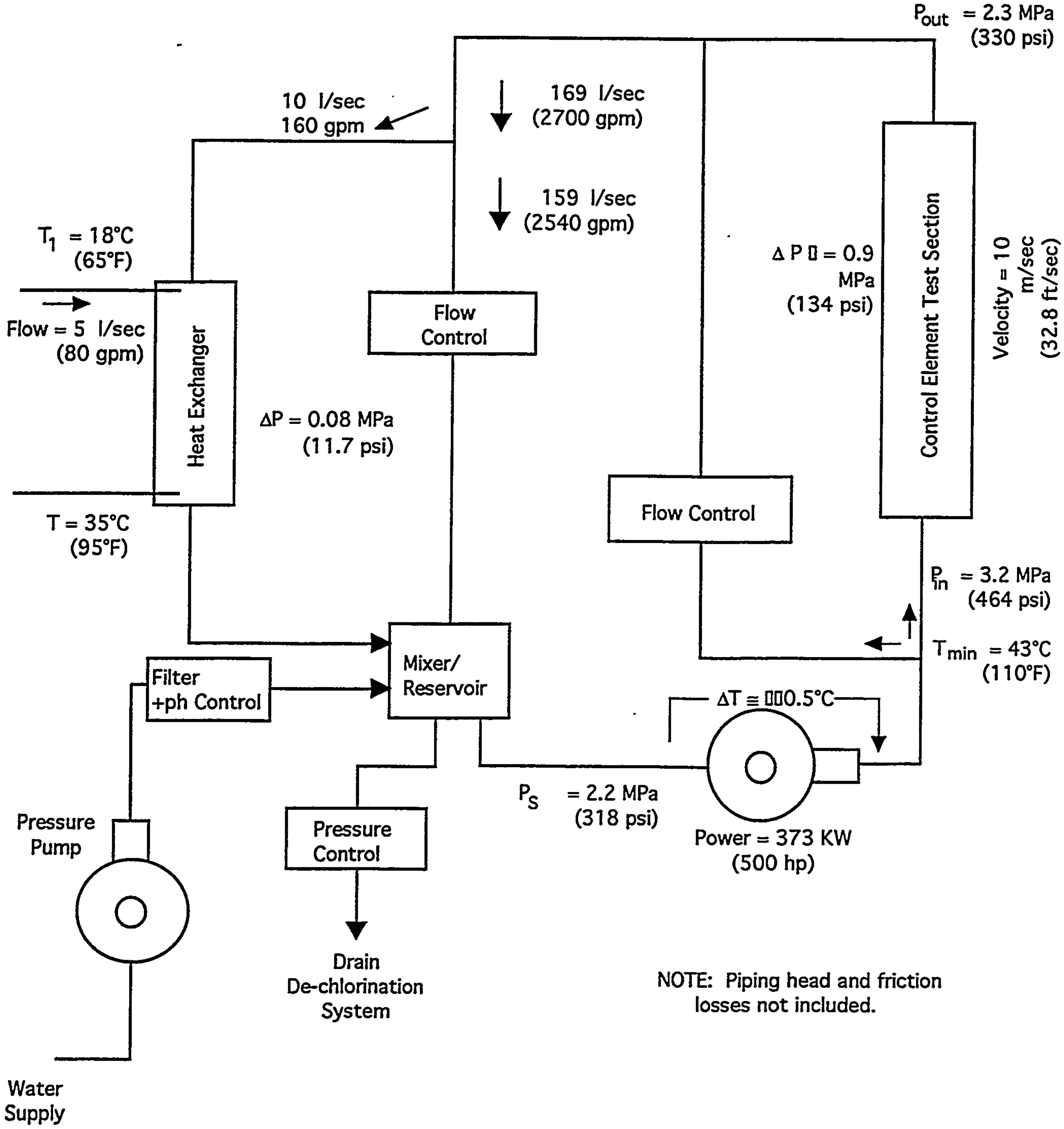


Fig. 3.24. Schematic of the control element test facility.

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 of operation for the reflector vessel. Three capsules contain material typical of the CPBT that is irradiated at a thermal-to-fast neutron ratio of ~2, close to the predicted ratio for the CPBT. These irradiations have been completed, and specimens from the first capsule have been tested. Irradiation was done at a temperature of ~95°C, the operating design temperature for the ANS, to a fluence of ~10^{26} neutrons/m^2 (thermal). Subsize tensile and compact specimens were irradiated. Post-irradiation testing of the specimens was conducted from room temperature to 150°C. The yield and ultimate tensile strengths increased after irradiation, and the total elongation decreased, as expected. However, the fracture toughness at 25 and 95°C was not degraded by irradiation and decreased only slightly at 150°C.

The CPBT will be required to sustain a persistent hoop stress that may cause accelerated creep under irradiation, resulting in permanent barreling of the CPBT. The extent of such deformation cannot be estimated because there are no radiation creep data for 6061 alloy, nor for other aluminum alloys. To fulfill this need, an experiment has been initiated in the HFIR to measure radiation creep in miniature, pressurized tubes of 6061-T651 alloy. We hope to get the first data from this experiment in the early months of FY 1995. Plans are being developed to tackle the remaining outstanding issue for irradiated aluminum—the effects of a highly thermalized neutron spectrum. This is an important matter because much of the radiation-induced change in the mechanical behavior of aluminum is caused by a precipitate of transmutation-produced silicon, which is produced by thermal neutrons.

Some irradiation tests also are planned on the control material and on material used for springs in the control rod scram mechanism. Some testing also will be done on these materials to establish baseline properties for unirradiated materials. Detailed test plans for these irradiations have not been completed yet.

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 ANSR vessel surveillance program will adhere to the requirements of ASTM E-185-82 and will satisfy the intent of 10 CFR 50, Appendix H. This WBS will provide the development of such a surveillance program. It should be noted, however, that exceptions may be necessary when these procedures and requirements are applied to nonferrous materials and/or when circumstances make them unfeasible for an experimental research reactor.

3.7.6 Materials Issues Task

3.7.6.1 Justification for Materials Issues Task

Various materials-related issues must be addressed. For example, 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 Section 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 Section III standards.

3.7.6.2 Description of the Materials Issues Task

After careful consideration of candidate materials, 6061-T6 aluminum was selected as the reference CPBT structural material. Although it has been used previously in research reactors, 6061-T6 aluminum has not been included in the *ASME Boiler and Pressure Vessel Code* for Class 1 nuclear construction. The project formally requested the *ASME Boiler and Pressure Vessel Code* Committee to include 6061-T6 aluminum as an acceptable material for Class 1 components in 1991. That request was considered and approved by a series of Code committees up through the main committee of the *ASME Boiler and Pressure Vessel Code* Committee. Formal approval was obtained in 1994. *Code Case N-519, Use of 6061-T6 and 6061-T651 Aluminum for Class I Nuclear Components, Section III, Division 1*[44] provides the requirements that must be met in order to obtain an ASME Boiler and Pressure Vessel Code stamp on ANS aluminum vessels.

Fracture-mechanics evaluation of critical components will be required. 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 that 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[45] 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) as part of this WBS.

This WBS subtask will also provide a sound basis for fracture assessment that includes the preparation of a deterministic methodology for preventing nonductile rupture in unirradiated and irradiated 6061-T6 aluminum. Adaptation of the probabilistic fracture methodology used for ferritic components of 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.

3.7.7 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; materials selection and design calculations must be based on a single reviewed and approved source of materials data to ensure accuracy and uniformity. A readily accessible, centralized materials data base specifically addressing 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

A primary goal of the task is the timely documentation and efficient distribution of reliable materials information generated in the ANS program. To accomplish this, a materials information gathering and distribution activity called the Advanced Neutron Source Materials Information System (AMIS) was formed. Initially, materials information is being compiled and documented in *draft* form. That is, the information will be unreviewed and "as-received." This will provide urgently needed data on a quick-turnaround basis to support those design applications whose schedules demand immediate estimates of material properties. This starting effort is being performed by a small core group of materials and design experts. Later in the task, the draft information will become the basic input for an in-depth analysis and development of "best-fit" values to be submitted to an extensive peer review. The result will be a collection of reliable data capable of supporting final design decisions, regulatory scrutiny, and QA requirements.

Initially, the information is being formed into a draft hard copy called the ANS materials databook. More than 250 pages of technical information have been prepared. Much of these data have been prepared in electronic format to facilitate conversion into an electronic data base. The Infodex Database System from Information Indexing, Inc., of Garden Grove, California, has been chosen as the main software for the ANS materials databook. The basic version of this software has been installed on the Energy Systems CD-ROM Library network. It contains an extensive collection of material properties taken from vendor publications. The ANS materials databook will contain selected portions of the Infodex Database supplemented by information generated by AMIS.

An AMIS advisory group is being formed to provide administrative and technical guidance to the AMIS activity. This will include selection of hard copy and electronic formats, input on data needs, data analysis and preparation, and review and approval of developed information.

3.8 COLD SOURCE DEVELOPMENT—WBS 1.1.8

One of the major objectives of the ANS Project is to provide a source of low-energy neutrons for experimenters. These neutrons are produced by two liquid deuterium cold sources within the reactor reflector and close to the core. This task provides the following:

1. a foundation of basic design principles,
2. testing and analysis to support the cold source design activity (WBS.4.10), and
3. the demonstration of a working prototype.

The major milestone of this WBS element is:

To complete validation testing of the overall concept by January 2001.

However, it is not practical to enact a single fully representative single test. Instead, four separate but complementary tests will together satisfy the milestone:

1. simulated full power heat flux testing of the thimble assembly,
2. cooldown and liquid fill testing,
3. control systems testing, and
4. refrigerator testing to full specification.

The cold source development activity is divided into the five WBS level four tasks summarized in Table 3.18. This work would be performed at the National Institute of Standards and Technology [(NIST) Boulder)], ORNL, the University of Virginia, and at least one other site. The total estimated costs for this activity over the 8-year period covered by this R&D plan are given in Table 3.19, and the associated schedules are shown in Fig. 3.25. 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

Table 3.18. Summary description of the cold source development work breakdown structure at level four

WBS	Task description
1.1.8.1	Neutronics analysis—This subtask provides for the development of computer models and subsequent physics analyses necessary to define the cold source geometry. Data produced will include flux levels and heat loads that will be used to develop the design and make material selections.
1.1.8.2	Thermal and hydraulic analysis—This subtask provides for the computer analysis of the thermal and hydraulic behavior of the cold source vessel design. This effort includes evaluation of the internal baffling required to achieve the required heat transfer coefficients and detailed loop evaluations. Thermal-hydraulic analyses of off-normal events will be performed in support of WBS 1.1.8.5. This subtask also covers physical modeling using a surrogate fluid to validate the design and provide benchmarking for the more complex computer analysis.
1.1.8.3	Structural analysis and design development—This subtask supports the development of major cold source components. It also provides for the design and construction of test facilities to evaluate their concept validity, reliability, and integrity under conditions as authentically reproduced as possible. The final phase of this subtask toward the end of the R&D program will be testing of collective components to validate their overall design concept.
1.1.8.4	Cold source instrumentation and control—This subtask covers the development of temperature and pressure control systems for the test program facilities. The development experience gained will subsequently be applied to the final cold source systems and also will point the way to the requirements for interlocks and information retrieval. A pressure control system has already been set up for the first test program, which is a liquid nitrogen cryogenic loop to be used initially to prove the liquid deuterium circulator design. This is a PC-based distributed control system with windows-based automation software that will allow final candidate pressure control configurations to be evaluated for subsequent application. Later test loops will allow other parameters to be examined in a similar manner.
1.1.8.5	Safety analysis and parameter definition—This subtask will provide for the identification and development of safety goals and requirements. A safety philosophy will then be developed to ensure that these requirements are subsequently met. Analyses and tests performed under this WBS and under the ANS accident analysis WBS 1.2.3.2 will be evaluated to quantify risks. This task also will eventually provide material for the cold source safety sections for the ANS preliminary and final safety analysis reports.

Table 3.19. WBS level four breakdown of costs for the cold source development activity

WBS Level 3	Level 4	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.8		Cold Source Development										
	1.1.8.1	Neutronics analysis	Exp.	97	120	50	50					317
			Line				25	50	50	30		155
	1.1.8.2	Thermal and hydraulic analysis	Exp.	436	917	553	227					2133
			Line				23	200	105	60		388
	1.1.8.3	Structural analysis and design development	Exp.	250	703	787	390					2130
			Line	130			390	330	550	661	65	2126
	1.1.8.4	Cold source instrumentation and control	Exp.	50	135	100	170	210	130	75		870
			Line	23								23
	1.1.8.5	Safety analysis and parameter definition	Exp.	10	30	50	70	45	35	19		259
			Line							16	50	66
		Subtotals	Exp.	843	1905	1540	907	255	165	94	0	5709
			Line	153	0	0	438	580	705	767	115	2758
		Contingency	Exp.	42	191	154	91	26	17	9		530
			Line	8		0	88	116	141	153	23	529
		Total	Exp.	885	2096	1694	998	281	182	103	0	6239
			Line	161	0	0	526	696	846	920	138	3287

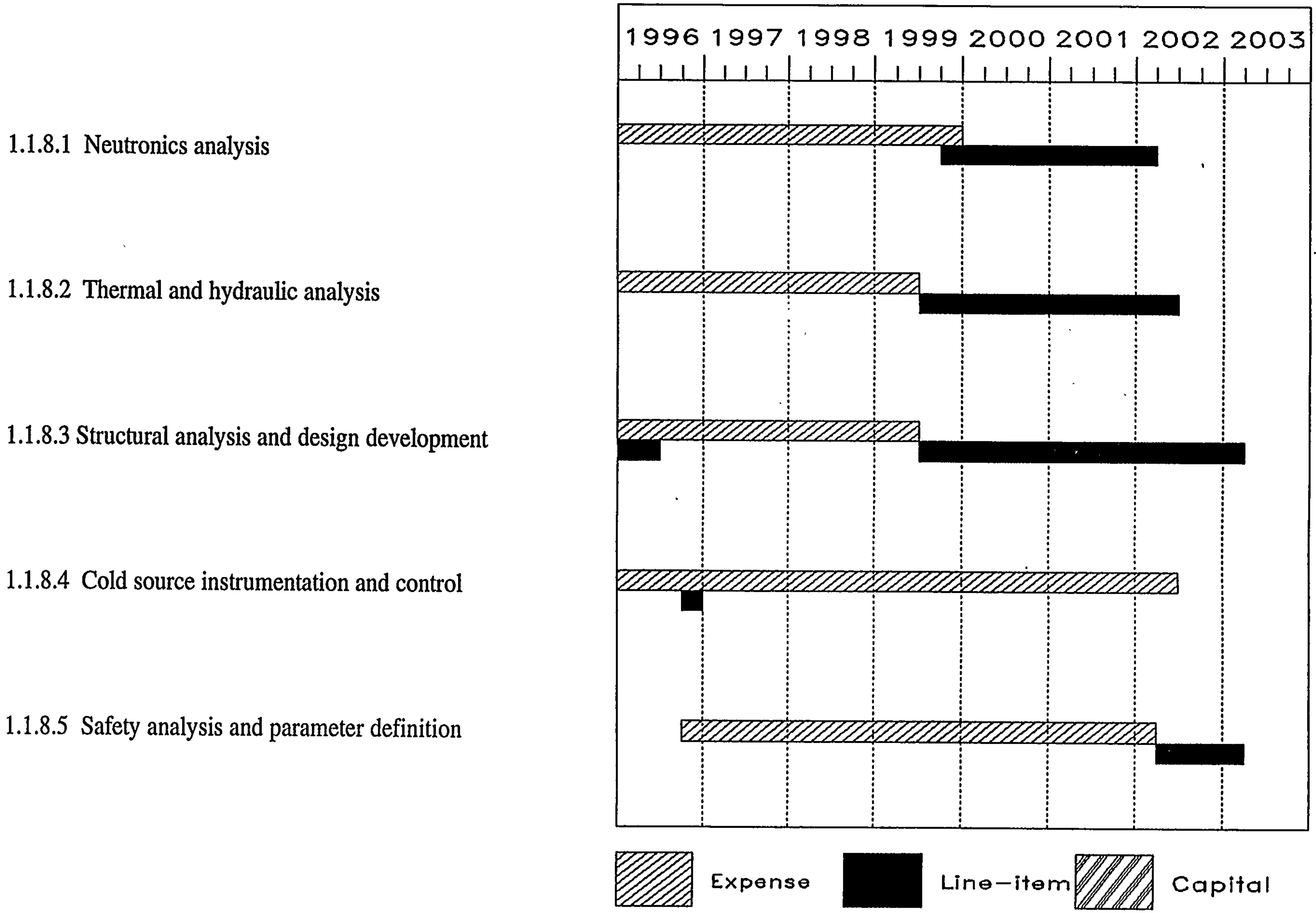


Fig. 3.25. Schedule for WBS 1.1.8 cold source development.

development phase. Subsections 3.8.1 through 3.8.5 provide more detailed information on the WBS level four tasks under this activity.

3.8.1 Cold Source Neutronics Analysis

3.8.1.1 Justification for the Cold Source Neutronics Analysis Task

The efficiency of the cold source moderator, in terms of cold neutron flux available to the experimenters, depends on the following factors:

1. the neutron flux entering the moderator vessel,
2. the materials of construction, and
3. moderator geometry and its resulting neutron flux perturbations.

The design of the complete thimble assembly must represent the best possible balance of these three parameters. This task provides the physics evaluations necessary to optimize this balance. Another element of this task addresses the need for an evaluation of the heat loads produced by candidate design configurations under normal and off-normal reactor conditions. These heat loads directly affect requirements for the refrigerator design (WBS 1.6.5) and therefore must be well understood. Without this task, it would be impossible to optimize the design or to predict the overall performance of the cold sources.

3.8.1.2 Description of the Cold Source Neutronics Analysis Task

Computer modeling will be used to analyze cold neutron fluxes, heat loads, irradiation damage to structural materials, and material activation levels that must be considered for maintenance activities. These calculations will be applied to four areas:

1. optimization of the cold neutron fluxes (to support the project objectives),
2. evaluation of heat loads generated within the cold source moderator and infrastructure (to support the cold source design WBS 1.4.10),
3. determination of neutron fluence and spectrum at cold source component locations (to support cold source irradiation damage studies WBS 1.1.7.5), and
4. examination of the cold source components activation levels (to support the design and procedures for maintenance activities WBS 1.4.10 and WBS 1.7.3).

Validation of the analysis methods will be covered under this task and will include analytical evaluation of existing cold sources and comparison with measured data obtained on these existing devices.

3.8.1.2.1 Status

Initial studies performed under this task concentrated on basic moderator and geometry decisions. Liquid hydrogen geometries were compared with liquid deuterium geometries. These studies validated the conclusions of early ILL work that indicated that a higher gain factor over the range of interest could be obtained from deuterium if the larger volume required could be accommodated. Concepts such as using a liquid nitrogen precooler to reduce the liquid deuterium inventory were examined but found to be ineffective. Cavities in the moderator vessel were investigated for gains comparable to those obtained at ILL. In fact, it was determined that although a cavity reduces the overall cold neutron flux, gains of 30 to 50% in the cold neutron current could be obtained. A cavity geometry was therefore adopted as part of the reference concept.

Based on these parametric studies and other input from WBS 1.1.8.2 and WBS 1.1.8.3, a general cold source design concept was developed under WBS 1.4.10. The cold source thimble for this concept is shown in Fig. 3.26. This general concept was modeled in two and three dimensions using various material combinations. Evaluations performed with these models indicate that cold neutron flux levels approximately 5 to 6 times those of ILL can be achieved. Cryogenic heat loads for this initial cold source geometry have been estimated, using two independent techniques, and are shown to be within the range of 25 to 30 kW for each cold source.

An ILL cold source model has been developed and compared with ILL published data. Heat load comparisons were good; the model underpredicted the reported heat load by about 10%. Spectrum comparisons with measured data over the energy ranges of interest were good over some portions of the range and not as good in others. A better comparison was obtained using German evaluated cross-section data for liquid deuterium, but agreement over certain areas was still not good, and further work to close this envelope is planned.

3.8.2 Thermal and Hydraulic Analysis

3.8.2.1 Justification of the Thermal and Hydraulic Analysis Task

Failure to provide adequate cooling of the moderator vessel walls under full power operation would result in hot spots that could subsequently lead to an uncontrolled rise in temperature and possible local boiling of liquid deuterium. This could cause pressure surging severe enough to activate the safety systems and shut the reactor down. It is therefore paramount that adequate heat transfer coefficients be maintained between the liquid deuterium and the vessel walls. As this is a function of the liquid velocity, the design of the necessary internal flow baffling must be carefully evaluated. It is essential that flow behavior through the gas and liquid transition stages be fully understood, because failure to keep fluid and hardware cooling in step with each other could also lead to surges that, in a serious case, could prevent the system from filling.

Moderator vessels must be annealed at 100°C at the end of each fuel cycle. Reactor decay power will probably be used to supply the necessary heating power, but the temperature rise of the vessel will be controlled by circulating deuterium gas at a volumetric flow rate of about 0.012 m^3/s. Since all instrumentation must be located some distance from the vessel, it is essential that the fluid temperatures always represent those of the vessel. This task therefore is an essential element in the development of a baffle design that is reliable under all conditions. It is also vital to gain a thorough understanding of the heat transfer mechanisms involved at all phases of the cooling and operation.

3.8.2.2 Description of the Thermal and Hydraulic Task

Fluid flow behavior, under all conditions of operation, will be understood using the following techniques:

1. computer modeling of the flow passages through the moderator vessel and
2. physical full-scale modeling using a surrogate fluid.

This work will be coordinated with the neutronic and pressure analyses that make up part of WBS tasks 1.1.8.1 and 1.1.8.3 and that are discussed in Sects. 3.8.1 and 3.8.3 of this report. It will be supported by a program to be carried out by NIST (Boulder) to generate hitherto unavailable thermophysical data for liquid deuterium. A facility will be built to flow liquid deuterium across a

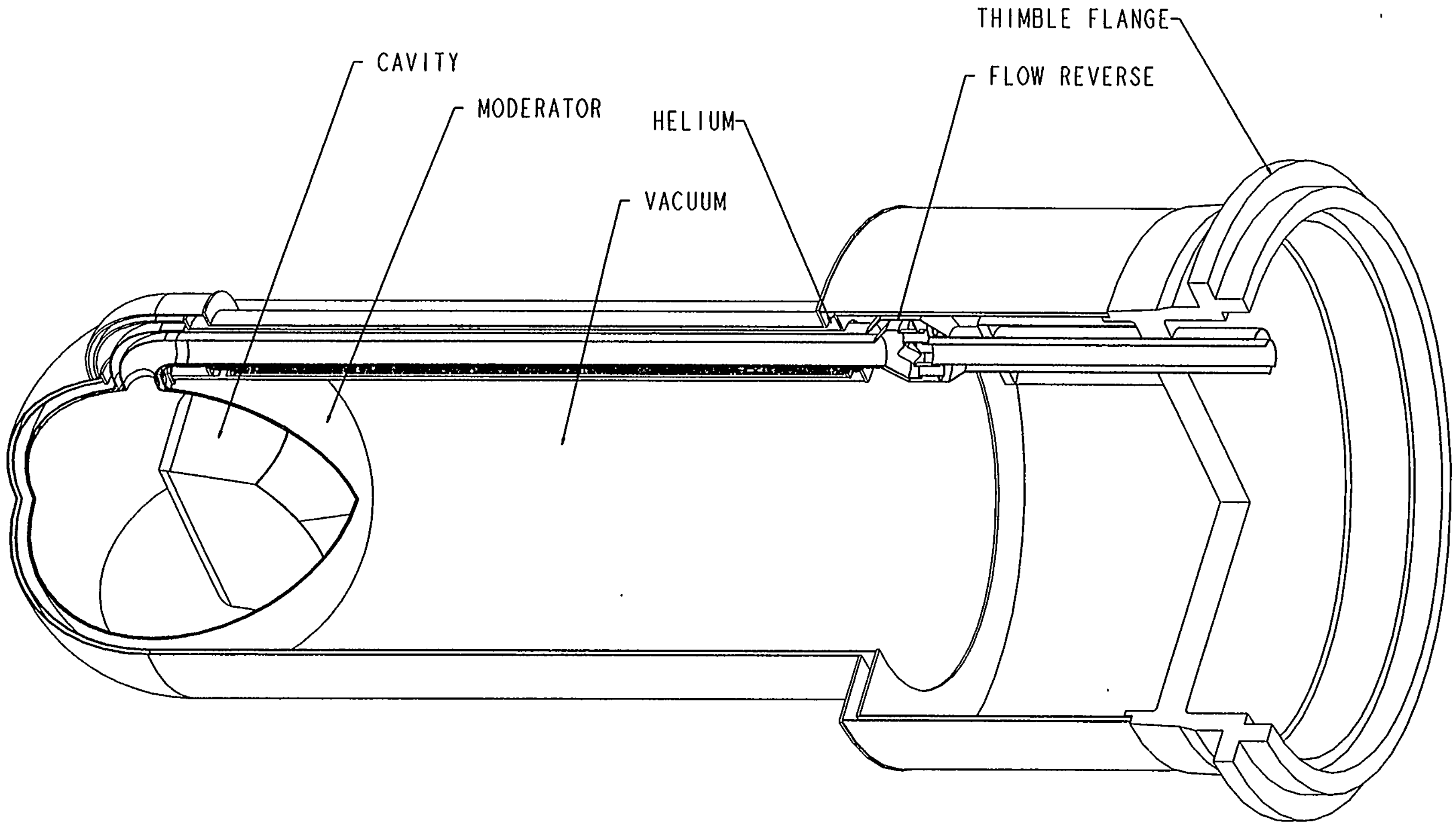


Fig. 3.26. Cold source thimble assembly.

reference strip of specific surface area. A heat load will be applied to the reference strip and corresponding heat transfer coefficients calculated over the required range of flow. This facility will also allow subcooled boiling phenomena to be studied visually.

3.8.2.2.1 Status

The design configuration used for the analysis calls for a thin-walled spherical vessel inside the main vessel, with a 2-mm gap between the two vessels. Liquid deuterium passes through the service pipe outer annulus into the cavity between the two spheres. This maintains the velocities required to provide the necessary heat transfer coefficients to remove the heat generated in both vessels. The liquid then returns through a hole in the bottom of the inner vessel to the inner service pipe at the top. The annulus was modeled using the computational fluid dynamics program CFDS-FLOW3D, with a hemispherical 2-D axisymmetric geometry for the hemisphere closest to the reactor core. The outer shell was assumed to be 1.5-mm thick and the inner shell 1-mm thick. The model factored in heat generation for both vessels, turbulent flow of the liquid, and heat transfer in the liquid within the annular flow field.

A number of plots were made for varying mass flow rates, and it was demonstrated that removal of the estimated 15 kW of heat from the vessel, in addition to a further 15 kW of heat deposited in the liquid, is attainable. A small region of superheating was indicated, and further runs were made with the gap at the waist line of the vessel reduced to 1 mm. This almost totally eliminated superheating, but it was clear that any subcooled boiling that might take place would be of such small order that the microscopic bubbles would collapse in the return leg of the transfer line.

A further run was made to examine the effects of relative movement between the two vessels. This did not appear to present serious problems, as a degree of self-compensation in velocity introduced a corrective factor in the heat transfer coefficients. A 3-D model will subsequently be created to examine the effects of the internal cavity on the flow patterns and possible perturbations in the cooling of the vessel.

3.8.3 Structural Analysis and Design Development

3.8.3.1 Justification for the Structural Analysis and Design Development Task

The cold source system comprises many safety-related systems and pressure vessels that contain deuterium at pressures up to 0.4 MPa. Special test facilities will be constructed to demonstrate the operation and reliability of key components of the system. The operational integrity of the final cold source systems depends heavily on this task, and this task will supply input to the cold source design activity (1.4.10) through the title II design phase.

3.8.3.2 Description of the Structural Analysis and Design Development Task

The scope of this task can be divided into seven areas of activity.

1. A stress analysis and an evaluation against the pressure vessel codes will be conducted for all pressure vessels to be used in the various test loops. This will establish all design principles for vessels to be used in the final cold source system. Vessels that make up the thimble assembly will require special evaluation because they will be required to operate in particularly arduous conditions of radiation and temperature change. Also, because the heat load is directly proportional to the thickness, the heat load must be considered in the thickness optimization

process. Input will be provided to the vessel design tasks (WBS 1.4.10) to ensure that all vessels will meet the requirements of the ASME Pressure Vessel Code Sect. VIII.

2. Test loops will be developed and built to demonstrate the operability and integrity of seven key operations of the final cold source system:

   A. A test program will be implemented to demonstrate long-term operation of the liquid deuterium circulators and to examine their operational characteristics to help identify early warnings of breakdown.
   B. Cryogenic valves will be developed and tested to demonstrate the isolation of circulators during replacement.
   C. Tests will be performed to confirm flow and heat transfer mechanisms operating in the moderator vessel at all levels of reactor power.
   D. The ability of the system to cool and fill with liquid within the required time scale of 20 to 24 hours will be validated. This test will also demonstrate the stability of the cold source during power level changes.
   E. The ability of the main heat exchanger to remain stable during cooldown, liquefaction, and normal full load operation will be confirmed.
   F. Testing will be performed to confirm the stability of the transfer lines under all operational modes, including fault conditions in which high gas velocities will be experienced.
   G. The ability of the control systems to respond to rapid changes in heat load will be confirmed. These tests will also provide input to the development of the interlocks required to maintain system integrity under faulted conditions.

   These test programs, including a full load test of the refrigerator, which will be performed under WBS 1.4.10, will represent validation of all the design principles developed for the final cold source system.

3. A logic will be defined for selecting and evaluating structural materials used in the cold source concept. The long-term suitability and integrity of materials used in the cold source will be established and, where necessary, testing will be performed to provide validation data.
4. A development effort will be performed to establish design principles for the cold source vacuum systems. Vacuum pumps will be assembled into inert-gas-blanketed pressure vessels to form vacuum stations. This is necessary to maintain the double containment philosophy and to allow problems, such as removal of heat from the pumps, to be addressed.
5. A subtask will be established to develop a double containment philosophy that will establish requirements for enclosing the entire system in an inert gas blanket. The design of components that must be replaced during operation, such as circulators and vacuum stations, must allow this to be done without breaking the double containment barrier.
6. A vent system will be developed which allows the inventory of each cold source to be contained at a reduced pressure of 0.12 MPa.
7. Metal hydride units will be developed to transport deuterium between the reactor building and the detritiation building.

3.8.3.2.1 Status

A cryogenic test facility has been built. A schematic diagram of the facility is shown in Fig. 3.27. Initial experiments to be performed in this test loop will address pump tests using liquid nitrogen. The experiment space set-up for these initial tests is shown in Fig. 3.28. Plans for later tests have been developed. A schematic of a heat test is shown in Fig. 3.29.

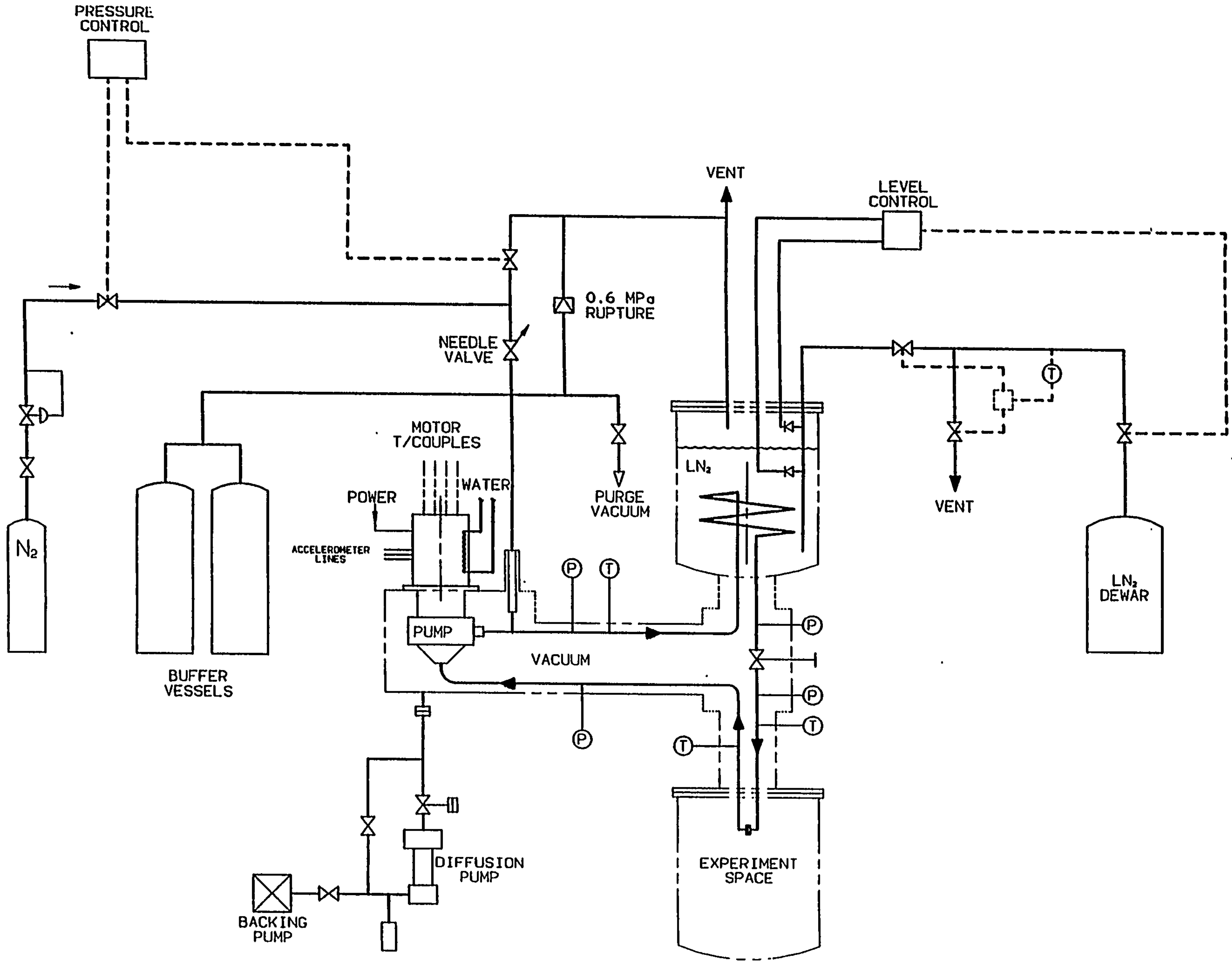


Fig. 3.27. Cryogenic loop test facility schematic diagram.

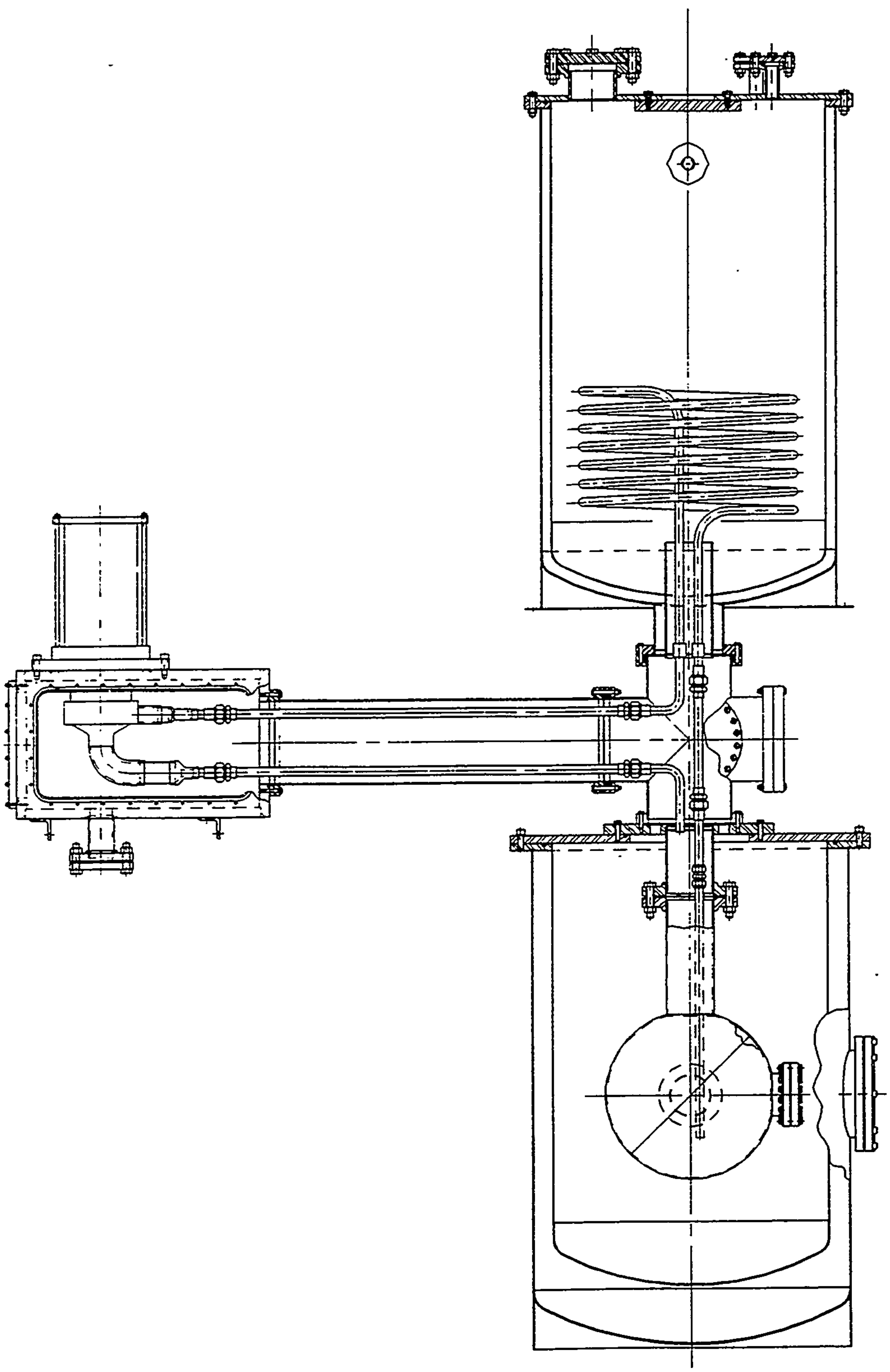


Fig. 3.28. Cryogenic loop test facility experiment space.

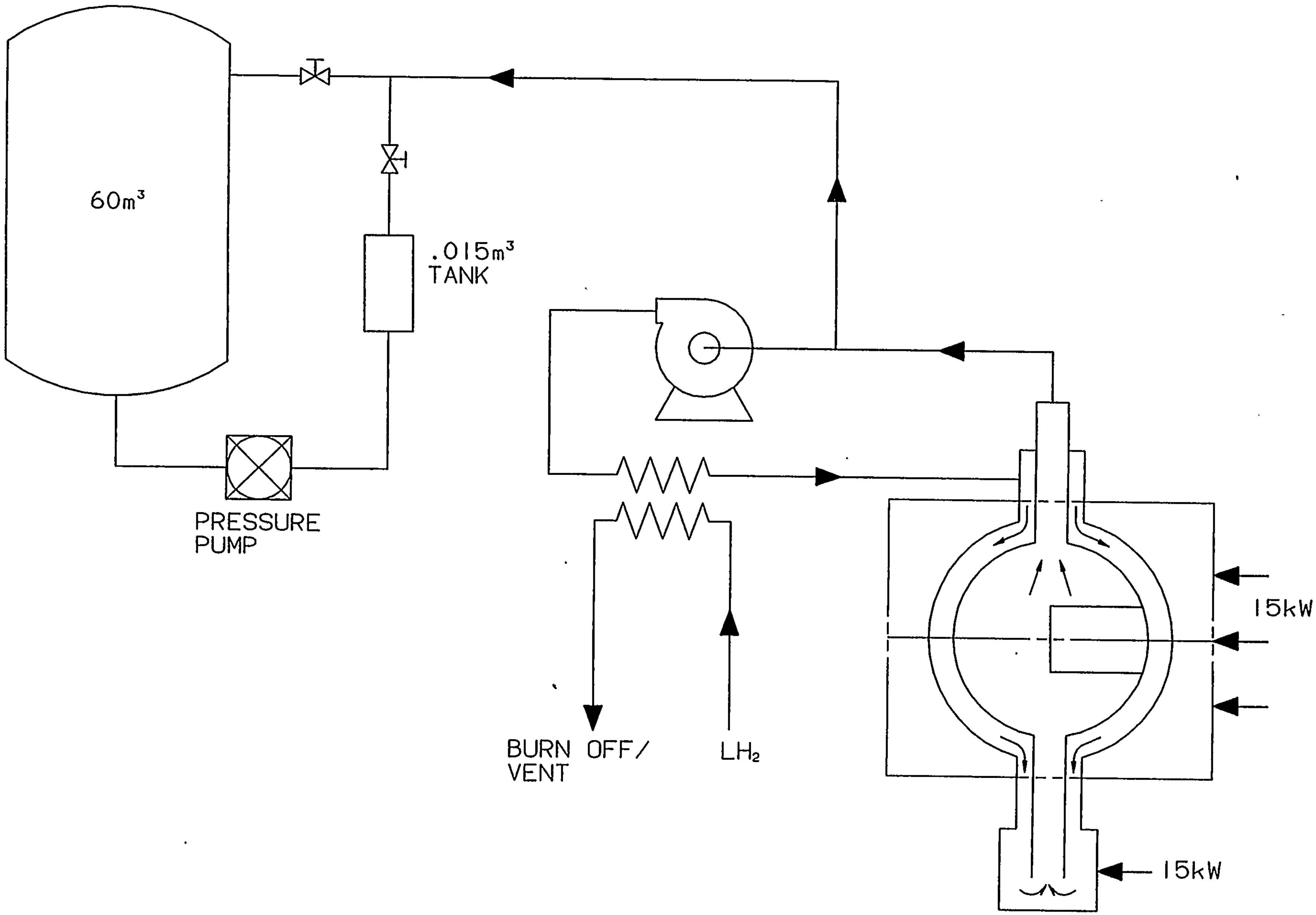


Fig. 3.29. Schematic of a heat test configuration.

Potential metal hydride unit concepts have been appraised. The appraisal indicated that uranium offers the best solution, partly because of its high storage capacity but also because of its operational temperature-pressure relationships. Uranium hydrides progressively break down into fine powder with successive cycling, and this would demand very careful design and handling. A unit size sufficient to contain one-third of the capacity of one cold source (as shown in Fig. 3.30) was estimated to offer the best compromise between handling and the number of unit changes required for the complete loading or unloading of a deuterium inventory. This unit size also proved to be close to optimum for overall construction costs. Other materials studied included titanium iron, uranium zirconium, uranium aluminum, and uranium silicon. The final choice of uranium, however, was based on present availability; new alloys could well be developed by the time a final design study is initiated.

3.8.4 Instrumentation and Control

3.8.4.1 Justification of the Instrumentation and Control Task

Operation of the cold source will depend heavily on the reliable operation of the control systems. Pressures and temperatures must be held within operational limits at all times. Similar control will berequired for the various test loop programs making this task essential to ultimate safety and reliability.

3.8.4.2 Description of the Instrumentation and Control task

The requirements of the instrumentation and control system fall into five areas.

1. Means of measuring the temperature and pressure at salient parts of the cold source circuit must be provided. This subtask will investigate the feasibility of reliable cryogenic flow measurement for gas or liquid operation. In addition, this subtask will provide appropriate test loop temperature and pressure instrumentation.
2. Control systems will be developed that are capable of applying up to 32 kW of heater power to the refrigeration loop in response to one of three temperature sensors, depending on the prevailing mode of operation. System pressures will also be controlled under all conditions of operation.
3. A system will be developed to provide constant monitoring of the liquid deuterium circulators to give early indication of possible failure. This will require the resolution of the output of vibration sensors, current monitors, and motor temperature sensors.
4. Safety barriers will be required to isolate low-current circuits from high voltages. This is necessary to limit possible energy release to an intrinsic safety level in the event of a system fault. All power equipment in the system will be specified to meet the hazardous gas "safety" requirements.
5. Fail-safe design features will form an integral part of the systems, and a general philosophy of interlocks will be developed to take appropriate action in the event of malfunction or incorrect operation.

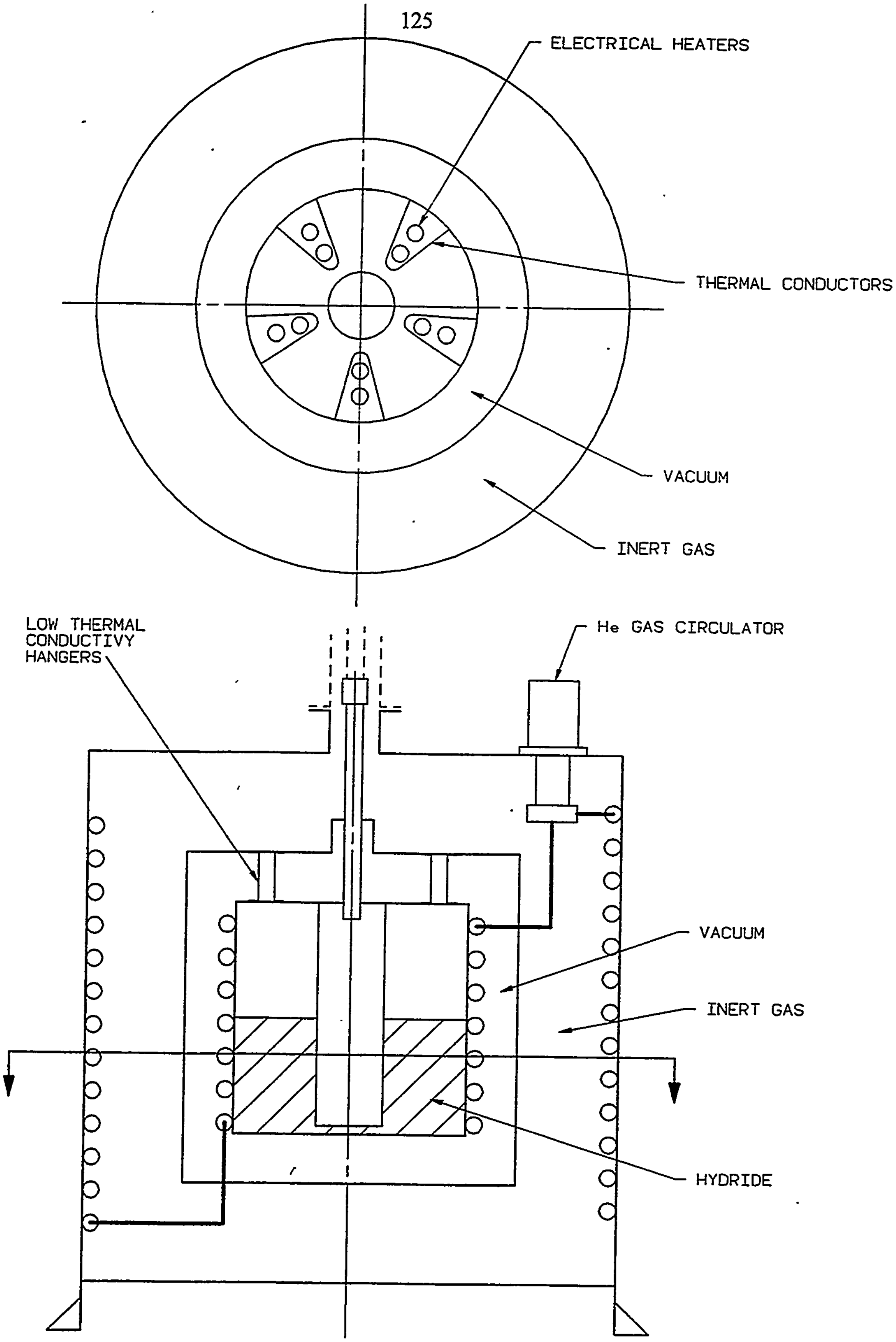


Fig. 3.30. Conceptual hydride bed design for ANS.

3.8.5 Safety Analysis and Parameter Definition

3.8.5.1 Justification for the Cold Source Safety Analysis and Parameter Definition Task

The two cold source systems proposed for the ANS contain a combined mass of 38 Kg of liquid deuterium. This represents a potential risk to workers and to the reactor. The ANS *Plant Design Requirements* document has adopted the goal that there should be no cold source accidents within the design basis (i.e., at frequency $>10^{-6}$/year) that lead to severe fuel damage. The cold source risk to affected workers should be comparably low. Developing a design philosophy to meet these very stringent overall safety goals is an essential task. Stresses and temperatures throughout the whole cold source boundary must be maintained within acceptable limits for the complete spectrum of reactor design basis events, as well as the comprehensive range of cold-source initiated events. This must be demonstrated to a high degree of confidence by analytical and experimental means as required. This subtask provides the evaluations necessary to ensure that these risks are consistent with the ANS project safety goals.

3.8.5.2 Description of the Cold Source Safety Analysis and Parameter Definition Task

This task addresses the potential safety issues associated with the cold source. Reviews of existing cold source SARs and input from the cold source review team will serve as a basis for establishing a cold source philosophy. This philosophy will be developed further by specific evaluations of ANS cold source risk. Support will be provided to the ANS Project probabilistic risk assessment group (WBS 1.2.3.4) to identify cold source accident initiators and probable frequencies. The potential consequences of off-normal cold source events will be examined both under this WBS and under the ANS accident analysis, WBS 1.2.3.2. The emphasis under WBS 1.2.3.2 will be on infrequent limiting events, typically involving interaction with or the release of energy to the surroundings following the escape of LD_2 from the cold source boundary. Based on the consequence evaluations, recommendations will be established for maximum frequency goals for various cold source conditions. These frequency goals will then be used to recommend a general design philosophy to the cold source design activity (WBS 1.4.10). This will ensure that the acceptable consequences of cold source failures or accidents are consistent with the expected frequency of those events. This task will also supply the documentation in the PSAR and FSAR necessary to support the safety case for the cold source.

This task will also be responsible for identifying cold source safety interface requirements with other systems. For example, some cold source failures will require emergency shutdown (scram) of the reactor, and the appropriate interfaces with the reactor protection system development tasks (WBS 1.1.12.2 and 1.1.12.3) must be defined. Another example is the air handling and treatment requirements of the "safe room" that must be interfaced with the development of the reactor building environmental control systems (WBS 1.6.3).

3.8.5.2.1 Status

A general cold source safety philosophy has been developed. Emphasis is on maintaining integrity of the D_2 containment boundaries because D_2 is not dangerous unless it mixes with water or air. Also, if LD_2 does not enter the reflector vessel, it cannot mix with the reflector D_2O to cause oscillations that might damage reactor components. This general philosophy has led to the concept of a safe room (as shown in Fig. 3.31) where most of the cold source operation and maintenance activities would be performed. It has also led to the design of separate inner and outer thimbles with a monitored gas barrier to give an early warning of possible heavy water leakages.

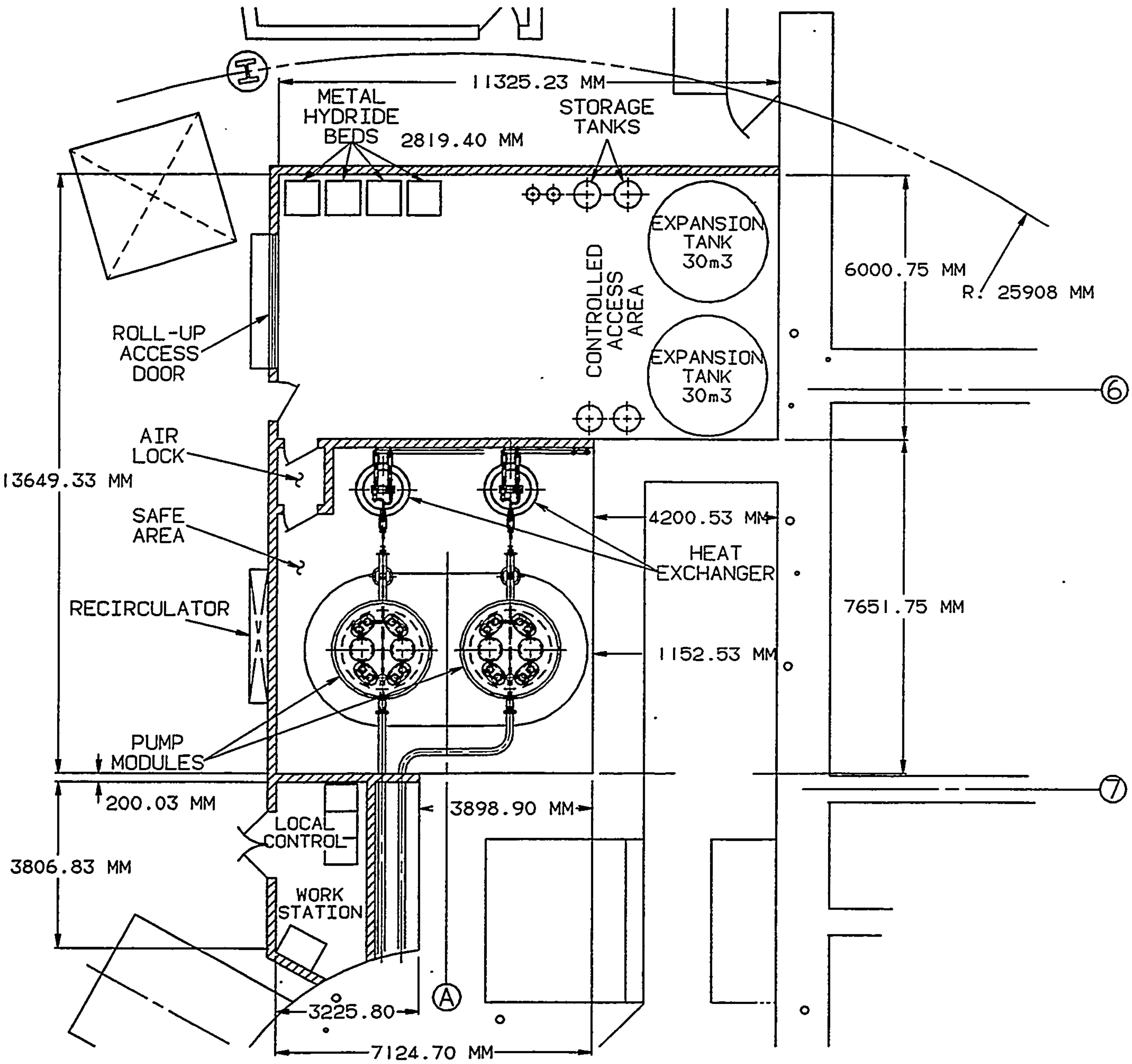


Fig. 3.31. ANS cold source safe room layout.

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 to deliver neutrons to beam experiments and to perform 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 1999. 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 3.20. Most of this work would be performed at Brookhaven National Laboratory, NIST, ORNL, and other subcontractors, including several universities. The total estimated costs for this activity over the 8-year period covered by this R&D plan are given in Table 3.21, and the associated schedules are shown in Fig. 3.32. 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.

Table 3.20. 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—The quality of any neutron beam depends on a very high signal-to-noise ratio. This aspect is addressed in this task by determining the best ways to minimize noise in the beams by using filters.
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	Monochromator 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—New technologies that are continually being developed for satellite and astronomical use will be investigated for their suitability for high-resolution neutron detection by a convertor, as opportunities arise.
1.1.9.5	Instrument systems development—New methods will be investigated for ultra-high-precision time-of-flight spectroscopy based on neutron spin precession at the Larmor frequency.

Table 3.21. WBS level four breakdown of costs for the beam tube, guide, and instrument development activity

WBS Level 3	Level 4	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.9		Beam Tube, Guide, and Instrument Development										
	1.1.9.1	Beam transport system development	Exp.	150	338	404	195	175	140			1402
			Line									0
			Cap.				617	271				888
	1.1.9.2	Polarizer development	Exp.	75	201	448	291	204	100			1319
			Line						107	107	107	321
			Cap.									0
	1.1.9.3	Monochrometer development	Exp.	210	242	451	423	398	311	145		2180
			Line						111	104	67	282
			Cap.									0
	1.1.9.4	Detector development	Exp.	127	186	569	950	780	120	120		2852
			Line						134	174	174	482
			Cap.									0
	1.1.9.5	Instrument systems development	Exp.	300	648	1019	1226	633	150	134		4110
			Line					289	280	267	267	1103
			Cap.									0
		Subtotals	Exp.	862	1615	2891	3085	2190	821	399	0	11863
			Line	0	0	0	0	289	632	652	615	2188
			Cap.	0	0	0	617	271	0	0	0	888
		Contingency	Exp.	43	162	289	309	219	82	40		1144
			Line					58	126	130	123	437
			Cap.				123	54				177
		Total	Exp.	905	1777	3180	3394	2409	903	439	0	13007
			Line	0	0	0	0	347	758	782	738	2625
			Cap.	0	0	0	740	325	0	0	0	1065

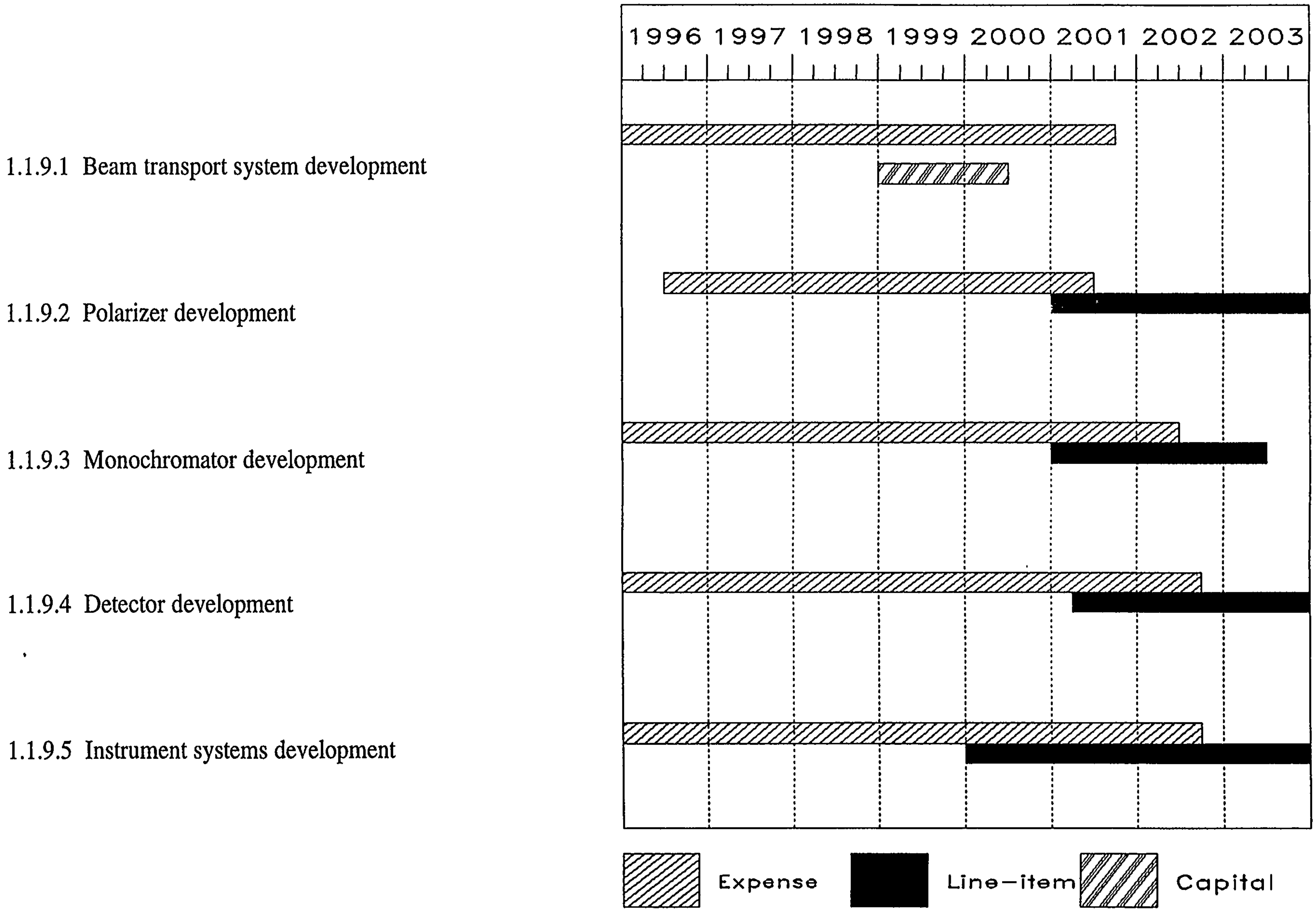


Fig. 3.32. Schedule for WBS 1.1.9 beam tube, guide, and instrument development.

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 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 that would seek to minimize the beam dimensions and the experimental requirements that 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 as follows.

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.
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, NIST, 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.

3.9.1.2.1 Status

Three items highlight the work that has been completed: (1) optimal beam tube locations and geometries have been determined for the conceptual design report reactor core geometry; (2) a new type of neutron filter for cold neutron beams, the neutron optical filter, has been designed and tested by simulation; and (3) the optimal techniques for very-cold neutron transport and for ultra-cold neutron production have been determined.

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 of 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 through 6 will be used to fabricate a linear neutron spin-echo set-up to test prototype devices and to assess the utility of different nonmagnetic materials (e.g., nonferrous bearings) in polarized-beam spectrometer construction.

3.9.2.2.1 Status

The main accomplishment to date is that a new design for neutron-optical thin-film polarizing supermirrors has been developed and tested. This led to the design and testing of a practical divergent-beam polarizer that resulted in an "R&D 100" award.

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 monochromator development task has been divided into six subtasks.

1. The use of different crystalline materials for focusing crystal monochromators will be evaluated theoretically for both 1-D and 2-D 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.

3.9.3.2.1 Status

This task has not been initiated.

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.

3.9.4.2.1 Status

The initial activity of this task was to examine existing detector technology and detector development programs. This was accomplished by sending questions to every major neutron scattering facility in the world. An evaluation of these surveys has been performed and published.[46] This led to the identification of detector development priorities and the best ways of addressing their development.

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 (1) methods of making the modularity more global to minimize the number of different components and (2) 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.

3.9.5.2.1 Status

Two items highlight the work that has been performed under this task: (1) an international workshop on instrumentation was held, and the results have been documented as an ORNL conference report;[47] (2) fail-safe interlocks that hold radiation levels as low as reasonably achievable (ALARA) at instruments have been designed and tested for practicality in "real life" experiments.

3.10 HOT SOURCE DEVELOPMENT—WBS 1.1.10

The hot source development activity includes the reactor physics, thermal-hydraulics, and initial structural analyses necessary to support the development of an ANS hot source facility. In addition, tests will be performed as necessary to demonstrate operation of the concept. The R&D team for this WBS will work closely with the engineering design team (WBS 1.4.11) in order to ensure fidelity with the plant and experiment design systems.

This WBS element contains one major project milestone:

> Complete hot source prototype tests by the end of September 2000. 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 nine months before the completion of the Title II design phase.

The objective of the hot source is to provide neutron beams with an increased current in the short wavelength range of approximately 0.4 to 0.8 Å with a performance at least equivalent to that of the ILL hot source. This is to be achieved by rethermalizing the background "room temperature" Maxwellian neutron flux in the reflector vessel with a moderator operating at 2400 K or above. An optimized design requires compromises between design features to increase the thermal efficiency (which tend to increase the size of the penetration in the heavy water) and the need to minimize the reduction in the background thermal flux caused by the penetration that drives design in the direction of minimizing the size and amount of materials. An additional consideration is the need to satisfy the safety requirements associated with reactor operation.

Because geometry, heating rates, fluxes, and possibly materials for the hot source in ANS will be sufficiently different from all other existing hot sources, a dedicated R&D program is needed to assure that the hot source will meet the design requirements. The hot source development activity is divided into five WBS level four tasks summarized in Table 3.22. Most of this work would be performed at ORNL, INEL, and other subcontractors to be determined. The total estimated costs for this activity over the 8-year period covered by this R&D plan are given in Table 3.23, and the associated schedules are shown in Fig. 3.33. 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.

Table 3.22. Summary description of the hot source development work breakdown structure level four tasks

WBS	Task description
1.1.10.1	Neutronics analysis—This task will provide the development of models, data (e.g., hot 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 hot source regions.
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	Safety analysis and parameter definition—This task will identify potential accident scenarios involving the hot source and determine the response of the hot source and the reactor to limiting scenarios in cooperation with the engineering design team.
1.1.10.5	Hot source testing—This task will use data from subtasks 1.1.10.1, 1.1.10.2, 1.1.10.3, and 1.1.10.4 to develop and demonstrate a prototype hot source system. 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 FSAR.

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 neutron energy transfer efficiency of the moderator material depends on the material cross sections. Therefore, significant neutronics support is needed to develop a hot source concept.

3.10.1.2 Description of the Neutronics Analyses Task

Parametric studies will be performed using 3-D 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. Parametric studies will need to be carried out to characterize the neutronic effects of candidate coolant and cooling shroud materials. Activation analyses will be performed to determine the long-term impact of the very high flux field on the hot source material composition

Table 3.23. WBS level four breakdown of costs for the hot source development activity

WBS Level 3	Level 4	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.10		Hot Source Development										
	1.1.10.1	Neutronics analysis	Exp.	139	185	70	64	48	48	37		591
			Line							46	46	92
			Cap.									0
	1.1.10.2	Thermal analysis	Exp.	200	229	214	143	83	48	48		965
			Line					18	46	128	84	276
			Cap.									0
	1.1.10.3	Stress analysis and structural design	Exp.	112	156	159	99	54	25			605
			Line			48	18	18	36	46	46	212
			Cap.									0
	1.1.10.4	Safety analysis and parameter definition	Exp.	374	260	77						711
			Line		113	472	323					908
			Cap.									0
	1.1.10.5	Hot source testing	Exp.		20	122						142
			Line				282	282	96	19		679
			Cap.			1072	1362	977				3411
		Subtotals	Exp.	825	850	642	306	185	121	85	0	3014
			Line	0	113	520	623	318	178	239	176	2167
			Cap.	0	0	1072	1362	977	0	0	0	3411
		Contingency	Exp.	41	85	64	31	19	12	9		261
			Line		23	104	125	64	36	48	35	435
			Cap.			214	272	195				681
		Total	Exp.	866	935	706	337	204	133	94	0	3275
			Line	0	136	624	748	382	214	287	211	2602
			Cap.	0	0	1286	1634	1172	0	0	0	4092

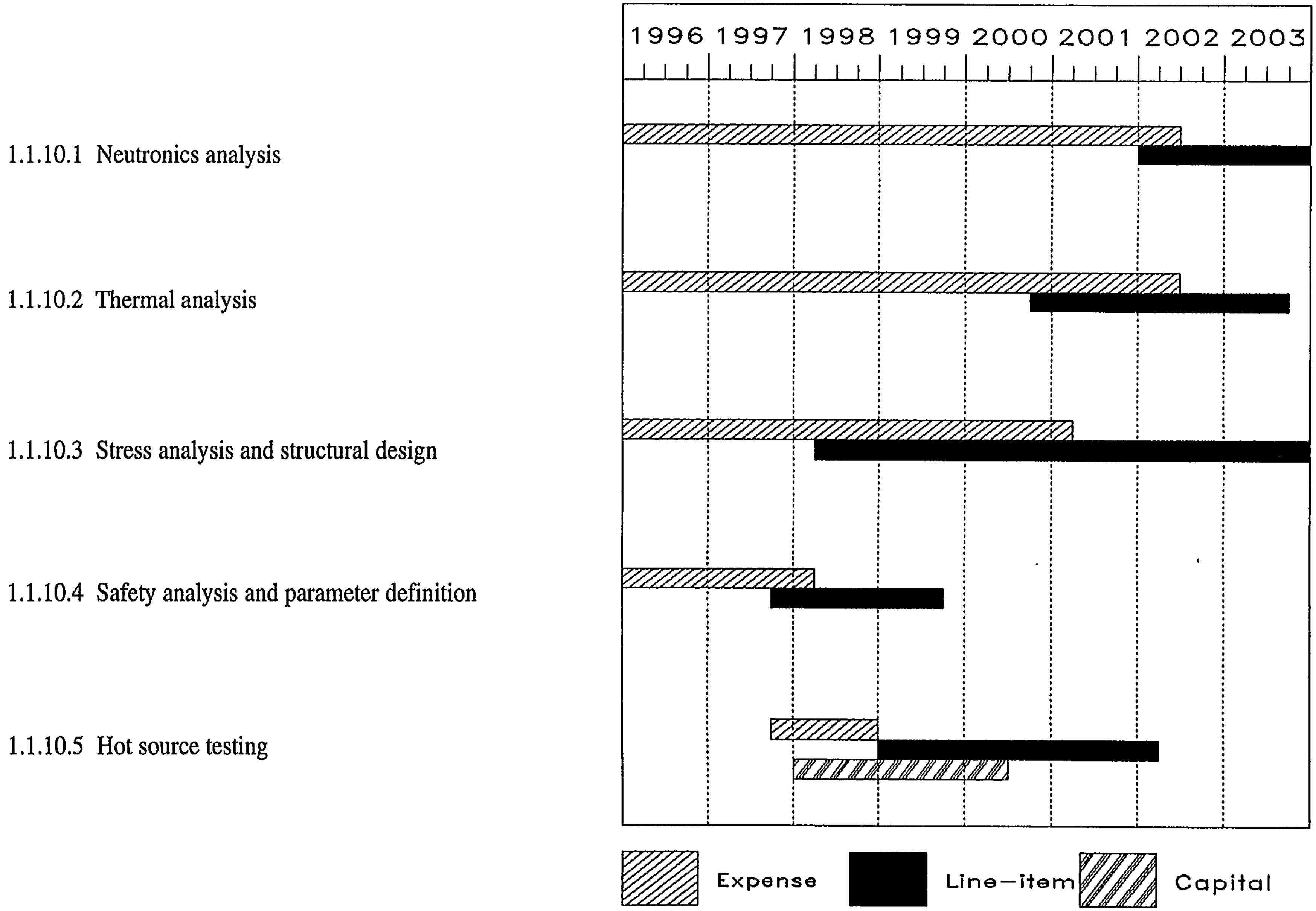


Fig. 3.33. Schedule for WBS 1.1.10 hot source development.

and to supply source terms that must be considered in the design of shielding for normal operation and maintenance activities. 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 physics analyses will be performed as necessary to support the design team.

3.10.1.2.1 Status

A review of the ENDF-III and ENDF-IV graphite S(α,ß) data at ORNL indicated inconsistencies. Further evaluations determined that the models used to generate the S(α,ß) data for ENDF-III, IV, and V used single precision computational methods that introduced inaccuracies for graphite, D_20, and other materials. The models were rerun with double precision computation and the inconsistencies were eliminated. All ANS hot source calculations were performed with the corrected data. This data has been sent to the National Data Center as a correction to the evaluated nuclear data files.

Three-dimensional Monte Carlo (MORSE) and two-dimensional discrete ordinates (DORT) calculations have been performed for the ANS hot source concept. Comparisons of the two models indicate excellent agreement for neutron wavelengths less than 0.502 nm. Over the wavelengths of interest (0.04 nm to .09 nm), the difference in the calculated currents at the hot source beam tube mouths was only about 5%.

The 3-D Monte Carlo model has been used to obtain neutron fluxes at the ANS hot source 5.5- and 6.5-m experimental stations. These calculations were performed using the 39N-44G processed 2000 K graphite cross sections.

3.10.2 Thermal Analysis

3.10.2.1 Justification for the Thermal Analysis Task

Accurate thermal characterization of the hot source is required to optimize performance with the conflicting goals of minimum thermal flux perturbation and the need for a high temperature moderator. Higher temperatures are achieved by increasing the size of the moderator and the amount of insulation. Both of these options increase the penetration size and decrease the amount of heavy water in the area, which causes perturbations that reduce the local thermal flux. The accuracy is expected to be achieved by complex thermal models that are later validated by data in the literature and by a dedicated test program. The work proposed in this WBS provides for the development of these models and their validation.

3.10.2.2 Description of the Thermal Analysis 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, to evaluate various insulator configurations, and to determine hot source temperature distributions. These calculations will also be used to evaluate the importance of thermal asymmetries that may exist because of asymmetric heat loading or design details. Initial calculations will be performed using simple analytic models, with more advanced numerical modeling incorporated into the task as required. Interfaces with the neutronic task within WBS 1.1.10.1 will be necessary to ensure that spacial heating is properly incorporated in the thermal hydraulic models. Interfacing with the structural tasks (WBS 1.1.10.3) will allow stress analysis that is consistent with the thermal analysis. Detailed analyses will be performed as necessary to support the design team.

As stated earlier, the ANS hot source must be capable of operating at or above 2400 K. Moderator materials in conjunction with candidate insulator materials will be examined to determine their compatibility with this goal and their ability to optimize the neutron current in the short wave lengths of interest. The sublimation characteristics of these materials will also be addressed by this subtask. A selection of moderator, insulator, and structural materials will be made based on their thermal characteristics. Thermal measurements will be performed as necessary to determine thermal behavior over a range of temperatures under appropriate gas composition and pressure.

3.10.2.2.1 Status

Two carbon-bonded carbon-fiber (CBCF) materials and one graphite felt material have been identified as candidate hot source carbon insulation materials. One of the CBCF materials was manufactured by the ORNL Metals and Ceramics (M&C) Division, while the second was supplied by UCAR Carbon Company. The WDF-graphite felt was supplied by the National Electrical Carbon Corporation (NECC). Thermal conductivity measurements for these three candidate insulation materials have been made at Southern Research Institute, Thermophysical Properties Research Laboratories, and the M&C Division at ORNL.

As a result of these measurements, a good set of conductivity data has been obtained for ORNL CBCF in helium at atmospheric pressure over the temperature range of interest for the ANS hot source. Data have been collected for as-fabricated material as well as for material receiving heat treatment at 3000 K for one hour. An equation fit was performed to the data of the material heat treated at 3000 K, and a simple model was found that predicts the data well. Conductivity data of ORNL CBCF in helium at various pressures up to one atmosphere were used to make extrapolations of the effect of helium pressure above one atmosphere.

UCAR CBCF thermal conductivity data collected are inconsistent; however, this material is clearly not as effective an insulator as the ORNL material, so collection of additional data for this particular UCAR material is not planned. It may, however, be desirable to perform conductivity measurements on higher density UCAR material (e.g., 0.4 g/cc) that may perform better than the ORNL material at high temperatures because thermal radiation heat transfer, which becomes important at higher temperatures, is less effective in the higher density material.

A reasonable set of conductivity data has been collected for the NECC WDF graphite felt in helium at atmospheric pressure and in vacuum over the temperature range of interest for the ANS hot source. At lower temperatures (i.e., below 1700°C), the felt is a more effective insulator than ORNL CBCF. This is a result of its density being about half that of ORNL CBCF. On the other hand, radiative heat transfer is more effective in the lower density felt, causing it to have a higher thermal conductivity at higher temperatures.

Several issues have been identified that may require further measurements. It has been postulated that the thermal conductivity of fibrous carbon materials may be sensitive to the thickness of the insulation. Thus, either analytically and/or experimentally this question must be addressed. In addition, testing is required to investigate the stability of insulation thermal conductivity with time at temperature.

3.10.3 Stress Analyses and Structural Design

3.10.3.1 Justification for the Stress Analyses and Structural Design Task

There are two separate structural problems that must be addressed: (1) structural integrity of the double-walled zirconium alloy containment vessel and (2) structural integrity of the 275-mm diam graphite sphere that is designed to operate at a temperature $\geq$ 2400 K. Assurance that the structural integrity of the containment vessel is adequate is complicated by the anisotropic behavior of zirconium, the poor thermal conductivity of zirconium and high internal heat generation because of nuclear heating that will result in thermal stresses, and the four nozzles. Irradiation induced dimensional changes and property changes because of irradiation must also be considered. Mechanical loads because of internal pressure, seismic events, and possible vibrations must be analyzed and evaluated. It will be necessary to get a Code Case approved for the zirconium alloy if it has to be constructed to meet ASME Section III.

Graphite is anisotropic and is subject to considerable dimensional change when irradiated. The rate of dimensional change is sensitive to the temperature at which irradiation occurs. Data are not currently available for irradiation at temperatures $\geq$ 2400 K. The dimensional changes caused by irradiation are often large enough to result in calculated internal stresses that are of sufficient magnitude to cause cracks to propagate. Fortunately, graphite also creeps when irradiated. This reduces the internal stresses that would otherwise cause cracking. Thermal creep will also reduce the internal stresses at 2400 K. Unfortunately, when the hot source is shut down, there will be no time for creep to occur so cracking is most likely to occur at that time.

3.10.3.2 Description of the Stress Analyses and Structural Design Task

The details of this task have not been fully defined. Because the thermal stresses and the irradiation-induced stresses, as well as the effect of irradiation on properties, are important in predicting the structural behavior, interfaces with WBS 1.1.10.1 and WBS 1.1.10.2 will be important. 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 Safety Analysis and Parameter Definition

3.10.4.1 Justification of the Safety Analysis and Parameter Definition Task

The hot source high temperature graphite block represents a potential risk to workers and to the reactor. The ANS *Plant Design Requirements*[3] has adopted the goal that the median probability of severe core damage or meltdown resulting from all potential internal events should not exceed 1×10^{-5}. In addition, a risk goal of 1×10^{-6}/yr has been established for on-site workers and visitors within 1.6 km of the ANS facility. The hot source risk must be comparably low to be consistent with these overall facility goals. Therefore, developing a design philosophy to meet these very stringent safety goals is an essential task. This subtask provides the evaluations necessary to assure that these risks are consistent with the ANS Project's safety goals.

3.10.4.2 Description of the Safety Analysis and Parameter Definition Task

Safety analysis will include evaluating hot source thermal behavior because of reactor transients as well as transients initiated because of the hot source itself. Initial safety evaluations will emphasize identification of potential off-normal scenarios, selection of limiting scenarios by preliminary analysis, identification of solution methodologies, and documentation of the conclusions. These initial evaluations will be used to provide input to the hot source test plan that will be developed under WBS 1.1.10.5. Depending on conclusions reached during this initial phase, analysis may range from the form of numerical conduction calculations for simple overpower transients up to two-phase liquid vapor calculations that may be necessary for heavy water ingress transients. The type and severity of the transients to be examined will dictate the detail of these evaluations.

3.10.5 Prototype Development and Demonstration

3.10.5.1 Justification for the Prototype Development and Demonstration 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 hot source concept. 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. In addition, operating constraints and other information deemed necessary to perform start-up testing of the hot source will be established. If performance flaws are identified, these tests will provide the opportunity for design change prior to installation into the ANS. Finally, testing will be performed to support safety analysis as required.

3.10.5.2 Description of the Prototype Development and Demonstration Task

The hot source development R&D team will provide analysis support as necessary to complete the design of the hot source concept as well as a prototypical experiment. 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. Testing will also address data needs for safety analysis. 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 shielding activity is divided into four WBS level four tasks summarized in Table 3.24. Most of this work would be performed at ORNL and INEL. The total estimated costs for this activity over the 8-year period covered by this R&D plan are given in Table 3.25, and the associated schedules are shown in Fig. 3.34. 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 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. It should be noted that because of the high importance of the shielding task (discussed in Sect. 3.11.4) to the design team, most of the work that has been performed and reported here is focused on WBS 1.1.11.4.

Table 3.24. Summary description of the neutron and gamma transport and shielding level four tasks

WBS	Task description
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 because of 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	Component 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	Analysis support to shielding design—A limited amount of shielding support is supplied by this task. Work will focus on evaluation of unique shielding problem. Methods and models to be used in design and evaluation of shields also will be developed under this task.

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). Particular design considerations that will be addressed are the performance of reference and alternative beam tube geometrics within the reflector vessel and the impact of neutron supermirror guides and their locations. In addition, the shielding design analysis must rely on the prediction of neutron and gamma leakage from the beam tubes as calculated in the analysis under this 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 the performance of analyses 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 starts with the neutron and gamma fluxes within the reflector system and calculates the highly anisotropic flow of neutron and gamma rays through the beam tubes. The

Table 3.25. WBS level four breakdown of costs for the neutron and gamma transport and shielding activity

WBS Level 3	Level 4	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.11		Neutron and Gamma Transport and Shielding										
	1.1.11.1	Beam transport analysis	Line	83	86	64	55	55	55			398
			Cap.									0
	1.1.11.2	Impact of reflector components	Line	106	110							216
			Cap.									0
	1.1.11.3	Component heat load and activation	Line	127	131	248	248	248				1002
			Cap.									0
	1.1.11.4	Analysis support to shielding design	Line	684	710	715	713	624	508	373		4327
			Cap.	46			46					92
		Subtotals	Line	1000	1037	1027	1016	927	563	373	0	5943
			Cap.	46	0	0	46	0	0	0	0	92
		Contingency	Line	50	207	205	203	185	113	75		1038
			Cap.	2			9					11
		Total	Line	1050	1244	1232	1219	1112	676	448	0	6981
			Cap.	48	0	0	55	0	0	0	0	103

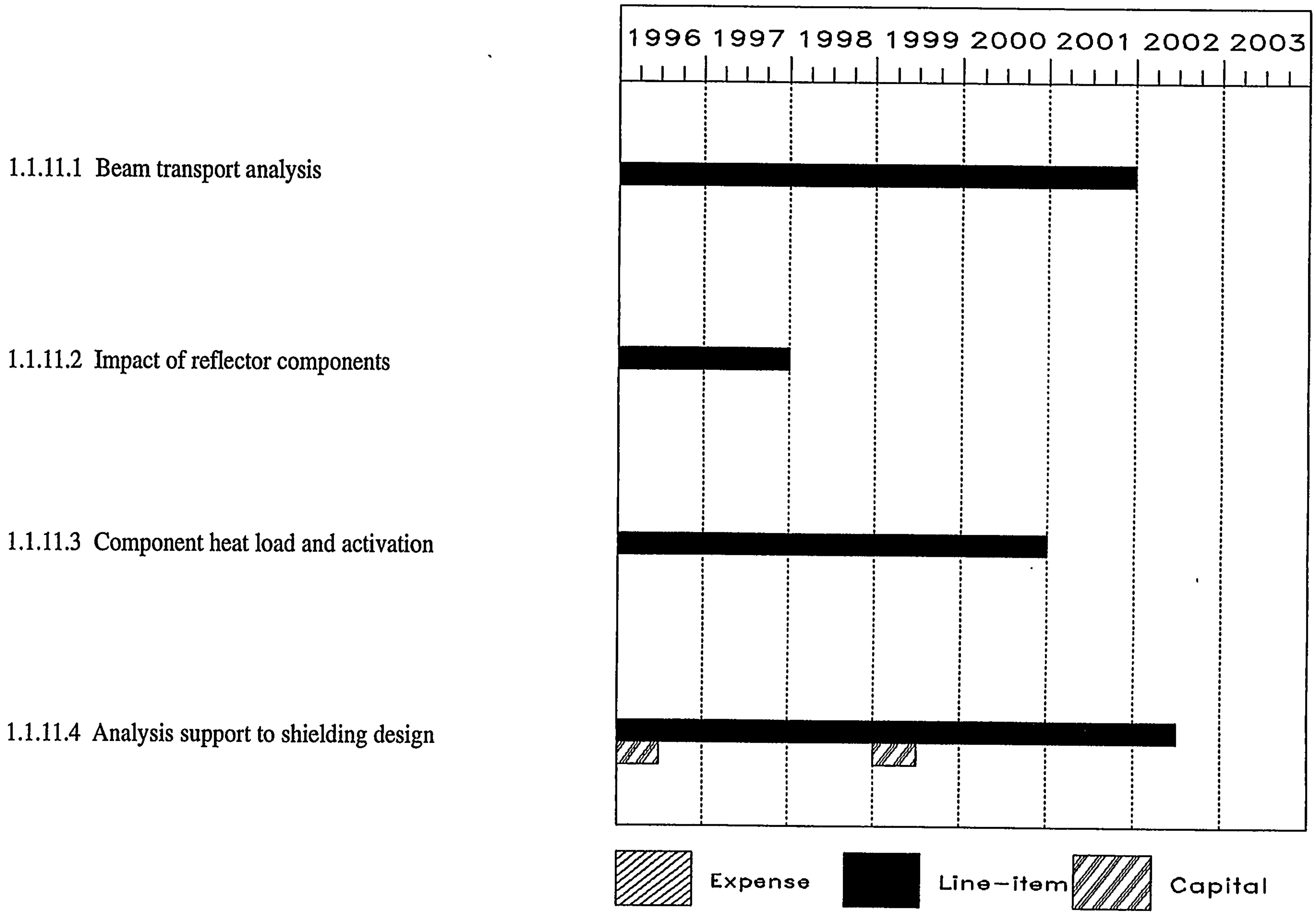


Fig. 3.34. Schedule for WBS 1.1.11 neutron transport and shielding.

neutron angular flux spectrum and gamma angular flux spectrum will be calculated within the beam tube to the point of exit from the reflector vessel. These spectra will be compared for various design options to determine the optimal beam tube thickness, cross-sectional geometry (e.g., elliptical versus circular), and possible conical widening of the tube with distance from the core.

For such difficult problems, modeling compromises between accuracy and calculational feasibility are necessary. Various modeling options will be evaluated, and modeling of HFIR and ILL will be performed to validate modeling assumptions. To date, the analyses have been carried out bycoupling two 2-D discrete ordinate calculations. More detailed 3-D Monte Carlo and discrete ordinates calculational models will be validated against HFIR and ILL and then incorporated in the analysis as the core and beam tube designs progress.

2. This subtask supplies the neutron and gamma leakages from the beam tubes as a source term to support shielding analyses. Simple models will be developed as conservative estimates of source terms exiting the beam tubes and the subsequent transport of the neutrons and gammas through ex-vessel shielding configurations. More detailed source terms will be developed using more detailed models as required.

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. This task will provide crucial perturbed flux data that, along with other data from WBS 1.1.11 as described in Sect. 3.11.1, will be used to determine the optimal beam tube design.

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 MCNP Monte Carlo models of the core/reflector system, with explicit modeling of each reflector component. In addition, a smeared reflector component model will be developed for use in the fuel cycle analysis model; validation will be carried out under task 3.1.1. Modeling is updated to follow the design evolution and to assess component design options.
2. Using the models developed in subtask 1, analyses will be performed to determine the effect of reflector components, individually and collectively, upon core reactivity and neutron flux densities. These effects will be determined at beginning of cycle, middle of cycle, and end of cycle using methods developed under task 3.1. 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 thermal-hydraulic studies performed under WBS 1.1.1 and studies carried out under WBS 1.1.7 to assess the material and structural integrity of each component. These heat load predictions are crucial input to the design of the cold source and beam tubes. Depending upon the heat load distributions, the cold source design may require a vastly different heat removal system that could greatly affect the design of the cold source thimble and related latching systems.

Activation of each component is also important information that must be obtained to evaluate handling needs and requirements during component replacement and/or maintenance. Data on activation of components within the reflector vessel will also need to be generated to assess material damage and component lifetimes. 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 of validating heat load calculation will be determined as part of this subtask and appropriate validation exercises will be performed. One possible source for validation data that will be assessed under this subtask is comparison against a NIST cold source benchmark. A detailed heat load analysis has been performed for the conceptual core geometry and is documented in ORNL/M-3993, *Detailed Heat Load Calculations at Beginning, Middle, and End of Cycle for the Conceptual Design of the ANSR.*[48]
2. Neutron fluxes and gamma fluences are determined per fuel cycle from the coupled neutron/gamma flux calculations over the fuel cycle. These fluences are provided to the materials evaluation task (WBS 1.1.7). This subtask will also include analysis required for optimization of isotopes production targets. A detailed evaluation of fluxes over the cycle for the various components has been performed and reported as ORNL/M-3738, *Detailed Flux Calculations for the Conceptual Design of the ANSR.*[49]
3. The neutron and gamma flux data calculated under subtask 2 are input to a depletion code for evaluating the activation within each component as a function of time after shutdown. This subtask will include determining how to incorporate time-dependent heat loads into RELAP5. The energies and magnitudes of radioactivity associated with the activation of components are determined and used in the component replacement and handling studies under WBS 1.3 and 1.4.

3.11.3.3 Justification for Support to the Design of Shielding Task

Components and personnel must be shielded from harmful neutron and gamma radiation resulting from reactor operations and out-of-core fuel and component transfers 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, one must rely on accurate analysis of gamma and neutron fluxes. Below is a current list of shielding design issues that will require supporting analyses.

In general, most of the studies described are necessary to determine and/or minimize the neutron and gamma dose rates to which operating personnel, refueling personnel, maintenance workers, or experimenters would be exposed during normal routine operations. This would certainly be true of many of the experimental systems, including the thermal beam tube penetrations, the through-tube loading station, the monochromators, the large slant beam tube (LSBT) and its isotope separation on line (ISOL) station, the slant cold guide tube, the cold guide tunnel, and many other experimental systems. It is also true of many maintenance or refueling operations where radioactive components are moved through the various pools or canals. Such operations may involve the movement of spent fuel, irradiated control rods, and the CPBT, as well as isotope targets, the transuranium rods, and other irradiation capsules or pneumatic tubes. Because many of these components will emit highly energetic gammas and will be moved through pools of heavy water, photoneutron production will have to be explicitly included in the analyses. (That is, in such cases, the standard point kernel gamma shielding codes commonly used by architect engineers will no longer suffice.) Shielding calculations will also be necessary to design a shipping cask for the spent fuel, as well as other transfer casks for the cold source, the hot source, and the through-tube. The object of these calculations will be to reduce external dose rates to acceptable limits. The size and weight of these casks will dictate the type of auxiliary handling equipment that must be provided and may have some marginal impact on the time required for periodic maintenance operations.

Most of the shielding studies described below are specifically needed to support further design work in the near future that is needed to remedy known problems. Moreover, much of this design work depends on an ongoing iterative interaction between shielding specialists and the project design team. For example, in the case of the through-tube shielding analysis, the results of preliminary analyses will determine the source terms impinging on the loading station, which will then be used to define the dimensions and thicknesses of components used in the design of the loading station itself. Likewise, in the case of the supermirror guide shielding, the space for equipment and personnel is extremely limited (especially near the biological shield). It is therefore desirable to determine, as soon as possible, whether adequate space for the necessary shielding is available and whether the required shielding will preclude or impact any of the other planned operations in this area. The "equipment transfer and manipulator cell" (and the associated heavy water and light water pools) will also require extensive shielding analyses. Some of the highly radioactive components mentioned in the first paragraph may be raised up out of the heavy water pool, placed into this dry cell for examination, and then lowered into the light water pool. Based on preliminary findings, it now appears that designers and shielding analysts will have to go through several iterations of the design of that cell before it can be finalized.

Other specific areas of more immediate concern to designers include the thermal beam tube penetrations, the hot and cold source penetrations, the monochromator drums, and the guide tubes leading to the monochromator drums.

A. During normal operation, with the shutter open on the thermal beam tubes, the neutron and gamma fluxes emerging from the collimator will be required input for analyses of the monochromator drums. At times, when the shutter is closed on a particular thermal beam tube, experimenters or other personnel may be standing close to these shield plugs during normal operation. Dosage rates there will be of concern—especially if any of the three monochromators close to the biological shield have been rolled back out of the way for maintenance. Periodically, the thermal beam tube shield plugs must be replaced. Gamma activation levels in the shield plug will then be needed to make preliminary estimates of the amount of shielding required for the transfer cask. This will determine the overall size and weight of the cask, which must be known to determine whether any special handling equipment may be needed.

B. There are a variety of concerns with respect to the hot source penetration during normal operation and during the periodic replacement of the hot source.

1. When the shutters are closed, experimenters or other personnel may be standing close to these shield plugs during normal operation.
2. Results of preliminary analyses may affect the design of the plumbing (e.g., cooling water tubes) required to cool the Zircaloy shells surrounding the hot source thimble.
3. As with the thermal beam tube shield plugs, the activation gamma sources in the shutter drums will be needed to make preliminary estimates of the amount of shielding required for the transfer cask used periodically to remove the shutter drums. This cask will also have to be capable of containing the hot source itself. Thus additional activation analyses will be required for the Zircaloy components associated with the hot source (such as the Zr shells, Zr support posts, and Zr support plate) because activated gamma sources in these components are expected to be extremely high and may well set the shielding requirements for this particular yet-to-be-designed cask.
4. During shutdown with the shield plugs and hot source removed, the amount of lead or iron shielding that would be required to protect maintenance personnel in the immediate area during the replacement operation must be determined. Preliminary analyses indicate that the amount of shielding required may be so bulky that, instead of using temporary shields, the biological shield may have to be redesigned to incorporate permanent window-like shields that could be raised or lowered into position when needed. Further analyses in this area will be required once the revised shielding configuration is available.

C. Because experimenters will be working within about a meter of the monochromator drums, several different types of shielding issues must be addressed during normal operation.

1. Is the rear wall of the drum thick enough to provide adequate shielding against the high neutron and gamma fluxes entering the drum through the beam tube? Note that these beams may be on the order of tens to hundreds of rem/hour during normal operation.
2. What are the neutron and gamma dose rates at the exit collimator? While thermal neutrons of a very narrow energy range are intentionally directed toward the exit collimator, fast neutrons and gammas scattering randomly within the drum (because of the impact of the beam with the inner rear wall of the drum cavity) will also escape through the exit collimator. Depending on the flux levels and dosage rates at the exit collimator, it may also be necessary to determine the angular distribution of the neutron and gamma fluxes leaving the exit collimator. This information will be needed as input to other studies where one may wish/need to develop a map of the fluxes and dosage rates in the beam room under normal operating conditions.

3. What are the neutron and gamma fluxes and dose rates near the entrance of the monochromator? As in (2), fast neutrons and gammas scattering randomly within the drum (because of the impact of the beam with the inner rear wall of the drum cavity) will also backscatter and escape through the opening formed by the movable wedges that are necessary to admit the beam into the monochromator drum. Potentially even more important is the fact that portions of these movable wedges close to the beam entrance may be only 10 or 20 mm from the edge of the beam. While the incoming thermal beam is tightly collimated, the incoming fast neutrons and photons will be less collimated. Some divergence of the 100 × 200-mm beam is certainly to be expected over the 6- to 9-m length of the long glass guide tubes leading to the monochromator drum—at least for fast neutrons and gammas. Even a small amount of divergence causing neutrons to hit these movable wedges can create a relatively large amount of backscattered radiation. This must be evaluated before one can begin to design the saddle shield to be placed in front of the monochromator drum and/or evaluate the effectiveness of the saddle shield design initially proposed. (Note that the required thickness of this saddle shield has yet to be determined and will require at least some preliminary shielding calculations in this area.) Even if the monochromator is temporarily removed from service (by closing the thermal neutron beam tube shield plugs leading to the incoming neutron guide tubes), there will be several shielding issues related to monochromator drums:

   a. The inner wall of the monochromator drum will be highly activated. What will be the dosage rates outside the drum (particularly near the exit collimator and the main beam entrance) after shutdown due to the highly activated material (steel) inside the drum?
   b. More important, what will be the dosage rate in the proximity of the monochromator drums (at shutdown or during normal operations) due to the activation of the movable steel wedges? During normal operation, a particular set of wedges will always be close to the main beam and will become highly activated. Although most of this activation is expected to occur on or near the inner portion of the movable wedges that are close to the main beam, other portions of the steel wedges will also become activated to some extent. If the monochromator drum were always rotated to select diffracted neutrons scattered though a single angle, these activated wedges closest to the main beam would be largely shielded by the saddle shield in front of the monochromator drum. In practice, however, the monochromator drum will periodically be rotated to select neutrons at different scattering angles. As the drum is rotated, those highly activated wedges that were previously shielded by the saddle shield in front of the monochromator will now move with the midsection of the drum and will be rotated to positions no longer shielded by the saddle shield. To the extent that workers and experimenters may have direct physical contact with these newly exposed activated wedges, one must verify that the resulting dosage rates are acceptable or must design additional shielding to make those rates acceptable. The final design of the monochromator drums will depend on additional shielding work in this area.

3.11.3.4 Description of the Support to the Design of Shielding Task

This task provides the analysis and evaluation of special shielding issues. These shielding problems require sophisticated neutron/gamma transport model development of the configuration of the areas to be shielded. Shielding requirements for each problem will be determined and design options for shielding will be evaluated.

3.11.3.4.1 Status

At this time, 15 of these special shielding issues have been identified for the R&D program. detailed descriptions of these problems and any progress toward their resolution are given below.

3.11.3.4.2 Cold guide tunnel

The 14 cold neutron guides exit the biological shield and enter a region referred to as the "cold neutron guide tube tunnel." This is a concrete enclosure designed to protect personnel in the ground floor beam room from neutron and gamma ray leakage from the cold neutron guides. Preliminary analysis has indicated that near the outside of the biological shield, some 500 mm of borated polyethylene shielding will be required around the guides; the concrete enclosure must be a heavy type such as barytes concrete; and the glass beam tube must be constructed of borated glass. When the design layout of this configuration is completed, a more rigorous analysis must be completed that includes the 3-D geometry effects and determines the shielding requirements as a function of distance from the biological shield. The preliminary calculations have proved the value of the analytic first collision source technique used to calculate the source in the guide tube and the surrounding shield material. Because of the asymmetric geometry around the cold guides, the first collision source may have to be used in a 3-D TORT or Monte Carlo calculation.

3.11.3.4.3 Thermal beam tube penetrations

A plan view of a shield plug on a horizontal thermal beam tube is shown in Fig. 3.35. The beam tubes and their source terms are similar enough that a solution for one beam tube will suffice for all. Given the source at the hot end of the beam tube and along its length, the primary objective is to calculate the neutron and gamma fluxes at the hot end of the shield plugs, as well as the neutron and gamma fluxes and dosage rates downstream of the shield plugs near the outer surface of the biological shield, with the shutter open and closed during normal operation. Activation analyses will also be necessary to determine the activation gamma sources in the steel portions of the plug. Because of the multiple off-center penetrations in a single beam tube shield plug, 3-D MCNP calculations or multiple 2-D RZ DORT calculations with different boundary sources must be performed.

3.11.3.4.4 Hot source penetration during normal operation

The analysis of the two hot source shield plugs with the shutter open and/or closed will be similar to the analysis of the horizontal thermal beam tube shield plugs. The shield design analysis will rely on the hot source analysis task to provide source terms. Depending on the size of the penetrations leading to the cooling tubes (and other associated plumbing associated with the hot source), the shield plug analysis itself may or may not be somewhat more complicated than that for the regular beam tube shield plugs. The primary objective is to calculate the neutron and gamma fluxes and dose rates downstream of the shield plugs near the outer surface of the biological shield with the shutter open and closed during normal operation. Activation analyses will also be necessary to determine the activation gamma sources in the steel portions of the shield plug. Because of the multiple off-center penetrations in a single beam tube shield plug, 3-D MCNP calculations or multiple 2-D RZ DORT calculations with different boundary sources will have to be performed.

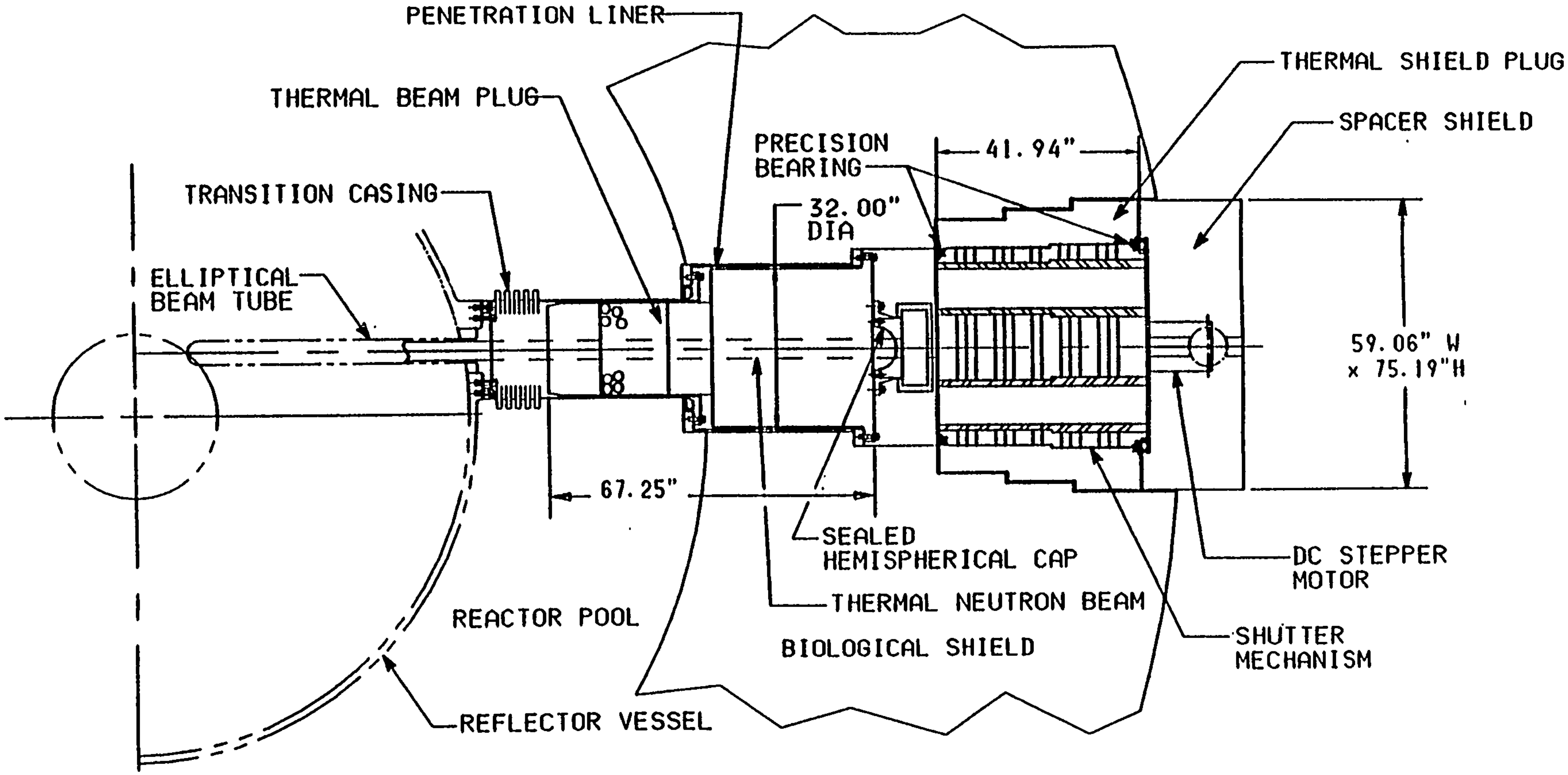


Fig. 3.35. Plan view of ANS beam tube shield plug.

3.11.3.4.5 Hot source penetration during shutdown with shield plugs removed

The hot source beam tube shield plugs (HB-4) and the Zircaloy-clad hot source still must be removed and replaced periodically over the life of the facility. Before this maintenance operation, the fuel will have been removed from the core, and the reflector vessel will have been drained at least below the reflector vessel holes. The removal of the outer shield plugs (two in the case of HB-4) will leave a 1.7-m-diam hole through the outer portion of the biological shield, and removal of the inner shield plug will leave a 1.25-m-diam hole through the inner portion of the biological shield. There would also be a 0.54-m-diam hole through the vessel itself for the removal of the hot source thimble. The purpose of this subtask is to evaluate shielding requirements by determining the neutron and gamma fluxes and dose rates at the outer edge of the biological shield under these conditions. Dosage rates through the hole (several days after shutdown) will depend on (1) the activity associated with aluminum in the reflector vessel; (2) the activity associated with the outer portion of the CPBT; (3) the activity associated with all of the beam tubes; and (4) the activity associated with the Zircaloy shells around the hot source, the support tubes, and the mounting plate, all of which make up the hot source and all of which may become highly activated. Because all of the activation gammas are less than 2.23 MeV, because the D_2O has been drained, and because of the 3-D nature of the problem, a direct line of sight (or point-kernel) will probably be used for this analysis.

3.11.3.4.6 Thermal beam tubes during shutdown

This task refers to determining the neutron and gamma dose rates outside the biological shield during the removal and replacement of the horizontal thermal beam tube shield plugs. Except for the slightly smaller diameter of the penetrations though the vessel (typically 0.32-m-diam here vs 0.54-m-diam for the hot source), the considerations related to this task are essentially the same as for the hot source task above. The results from the hot source task may be regarded as a fairly realistic limiting case for the removal of the other beam tube shield plugs.

3.11.3.4.7 Large slant beam tube

The analysis of the LSBT shield plug with the shutter open and/or closed will be similar to the analysis of the horizontal thermal beam tube shield plugs, even though the shield plug in this case is much further from the reflector vessel than in the horizontal thermal beam tube case. The LSBT consists of a straight 152.4-mm-diam tube that extends up along a slanted diagonal line from a point inside the reflector vessel (located on the centerline of the system, 449.6 mm above the midplane), through the corner of the vessel, and through the light water in the concrete vault, and continues on a diagonal path up through the biological shield to the hot end of the shield plug (located ~7.92 m from the corner of the reflector vessel). The opposite (cold) end of the LSBT shield plug is located in a separate room on the second floor of the reactor building more than 9.14 m above the reactor. Between the reflector vessel and the concrete vault, the LSBT is located coaxially inside a 203.2-mm-diam pipe, and that portion of the LSBT passing through the biological shield is located coaxially inside a 355.6-mm-diam stainless steel pipe. In the initial analysis, a simple solid-angle approach will be used to determine the neutron and gamma fluxes at the hot end of the beam tube shield plug when it is closed. A global 2-D analysis of the core and reflector may provide the source terms for the hot end of the LSBT. Analysis of the beam tube shield plug itself would be performed similarly to that for the horizontal thermal beam tube shield plugs.

3.11.3.4.8 Slant cold guide

The description of the slant cold guide tube and corresponding shield plug is almost identical to that for the LSBT. While the hot end of the slant cold guide tube is also located inside the reflector vessel, it is located at the midplane of the system, 704.8 mm from the system centerline; and the distance from the reflector vessel to the hot end of the slant cold guide tube shield plug is 7.92 m rather than 7.44 m. Note that the cold end of the slant cold guide tube shield plug is located in the same second-floor room as that of the LSBT. Initially, the results for the LSBT will be multiplied by the ratio of the neutron and gamma fluxes at the hot end of the slant cold guide tube to the neutron and gamma fluxes at the hot end of the LSBT.

3.11.3.4.9 Through-tube

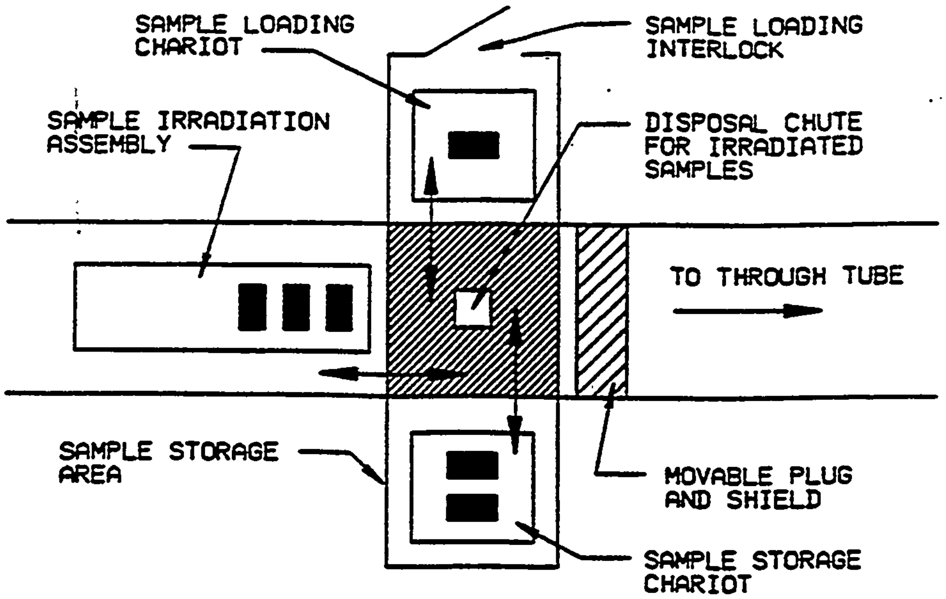
The horizontal through-tube is designed to allow samples to be loaded at one end (via the through-tube loading station at HB-10) while fluxes viewed from the other end (HB-5) are largely without the high background gamma radiation normally associated with the other beam tube whose hot ends are normally located in the reflector. This tube, which runs clear through the reflector vessel, is located in a horizontal plane 300 mm below the midplane and is offset from the CPBT by a small distance. Mechanically, the beam tube shield plug at the viewing end (HB-5) is identical to the horizontal thermal beam tube penetrations, while the penetration at the loading end (HB-10) is of a special design (Fig. 3.36).

Since the through-tube does not "dead end" in the reflector (like the other horizontal thermal beam tubes), the emerging neutron and gamma fluxes at either end cannot be calculated simply and easily using the solid-angle methods used for the other horizontal thermal beam tubes. The angular flux distribution in the reflector based on the 2-D RZ global analysis of the core and reflector could be coupled to a second 2-D RZ calculation along the centerline of the through-tube. This second 2-D RZ analysis could then be used to determine the neutron and gamma fluxes emerging from either end of the through-tube. These, in turn, could serve at boundary sources for the analysis of the beam tube shield plug (HB-5) and/or the loading station (HB-10). Alternately, one could request that the angular-dependent boundary fluxes at either end be generated as part of the 3-D MCNP analyses of the core and reflector being performed by INEL.

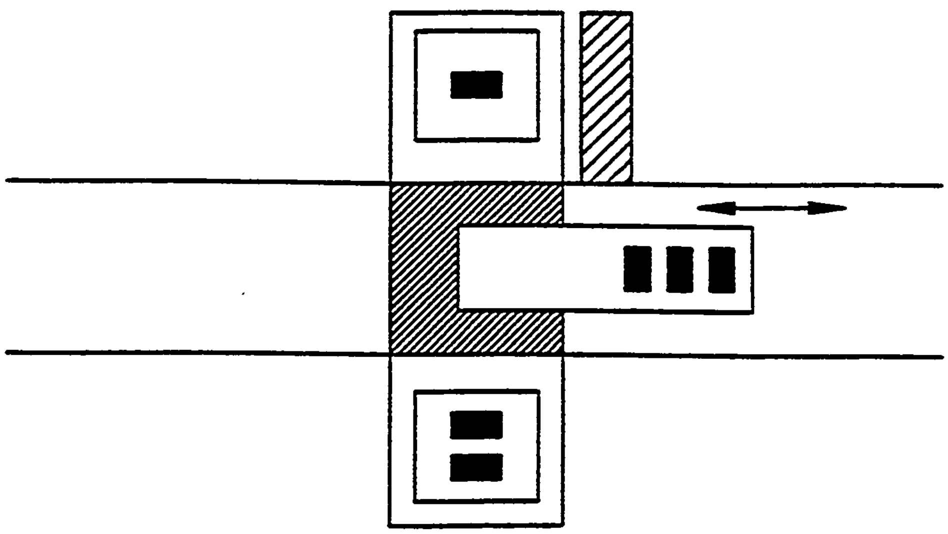
Either way, the fluxes and dose rates downstream of the beam tube shield plug (HB-5) could then be calculated similarly to those for the other horizontal therm beam tube shield plugs. Initially, the results of those previous analyses could be scaled up or down to reflect the intensity of the neutron and gamma sources. While less rigorous, these results could be made available in a more timely fashion. The method for analyzing the dosage rates outside the loading station will be determined as the details of the design become known. Tentatively, 2-D or 3-D methods similar to those previously employed for the horizontal thermal beam tube shield plugs could be used. Results of preliminary shield analyses may be used to help define the dimensions and thicknesses of components used in the design of the loading station.

3.11.3.4.10 Supermirror guide shielding

Several of the large 91,000-kg monochronomator drums are located 6 to 9 m away from the large biological shield surrounding the core and reflector vessel (i.e., 6 to 9 m downstream of the shield plugs on the corresponding horizontal beam tubes). Boron-lined glass tubes extending from the beam tube shield plugs to these distant monochronomators will serve as guides for the thermal neutrons between these two points. While the thermal neutrons in these extended tubes will be tightly collimated by borated plates in the beam tube shield plug penetrations, the fast neutrons (and



SCHEMATIC PLAN VIEW OF THROUGH TUBE LOADING STATION WITH ALL ELEMENTS WITHDRAWN AND INTERLOCK OPEN FOR INTRODUCTION OF SAMPLE. THE INDICATED ELEMENTS CAN BE BROUGHT INTO THE WORKING AREA FOR VARIOUS SAMPLE MANIPULATIONS.



SCHEMATIC PLAN VIEW OF THROUGH TUBE LOADING STATION DURING INSERTION OR WITHDRAWAL OF THE SAMPLE IRRADIATION ASSEMBLY.

Fig. 3.36. Schematic diagram of through-tube loading station operations.

gammas) entering these tubes will have some divergence and will tend to leak out of the tubes radially (in a near axially-tangential direction). A long saddle shield composed of blocks of shielding material above, below, and to either side of these long glass tubes will be necessary to protect people working several feet to either side of these long tubes. Both the composition and required thicknesses of these blocks have yet to be determined.

The required shielding at either end of these glass tubes (i.e., near the beam tube shield plugs and near the monochronomator drums), as well as the required shielding along the length of the glass tubes, must be determined. Near the beam tube shield plugs, flux levels and dosage rates will likely be governed by the neutron and gamma scattering density in materials near the cold end of the beam tube shield plugs and/or the associated shield plug saddle shield(s). Neutron and gamma radiation scattering from these regions will first have to be determined from 2-D or 3-D analyses of the beam tube shield plugs. This will serve as one of the source terms in the analysis of the saddle shields for the glass guide tubes near the hot end. Along the length of the glass guide tubes, neutrons (and gammas) will tend to leak out of the tubes radially (in a near axially-tangential direction) as a result of the natural divergence of the (fast) neutrons and gammas in the beam itself. This source of radiation will likely determine the required thickness of the saddle shield along most of the length of the glass guide tube(s).

At the far end of the glass guide tubes, close to the monochronomator drum, there will also be a third source of neutron and gamma radiation due to the backscattering of radiation from the drum itself. While the monochronomator saddle shield will be designed to absorb most of this radiation, some fraction of this backscattered radiation will inevitably enter the conduit formed by the saddle shield over the glass guide tube. When setting the thickness of the saddle shield over the glass guide tube, this source of radiation must also be considered. A combination of analytical, deterministic, and stochastic techniques will have to be devised to determine all three types of source terms described above and to determine the required thickness of the saddle shield along the entire length of the glass guide tube(s).

3.11.3.4.11 Cask for cold source and thimble and cask for hot source and thimble

Periodically, both the hot source and the cold source must be replaced. This will involve the use of a transfer cask to transfer these highly activated components from the reflector vessel to an on-site long-term storage facility. Tentatively, both the hot source and the cold source will use the same transfer cask. In both cases, the transfer cask will first be used to transport the outer beam tube shield plugs to the storage area. The cask will then return to get the inner beam tube shield plugs and the hot (or cold) source thimbles, respectively. All of these components will be highly activated. It is presently anticipated that the activated source terms associated with the hot source (especially the Zircaloy components associated with the hot source) will represent the limiting case for the cask design, although the activated steel components associated with the beam tube shield plugs must certainly be evaluated as well. Because of the complicated 3-D geometry of the hot source (i.e., the graphite ball surrounded by Zr shells and cooling water, all supported in a cantilever fashion inside a large vacuous enclosure inside the reflector), the neutron fluxes needed for activation analysis of the hot source thimble will have to be determined by a 3-D MCNP analysis. The same is true for the components making up the cold source thimble (even though the materials used here are not nearly as prone to neutron activation as the Zircaloy used in the hot source). Likewise, the neutron and gamma fluxes in the steel portions of the hot source and cold source beam tube shield plugs will have to be determined by a 3-D MCNP analysis so that these fluxes may be used in the prerequisite activation analyses. Given the activation sources just described, 1-D or 2-D discrete ordinate shielding calculations will be used to determine the necessary shield thicknesses for the design of the cask. One-dimensional calculations will be used for the preliminary scoping analyses, and 2-D calculations may optionally be used in the final analyses.

3.11.3.4.12 Cask with through-tube

The horizontal through-tube also must be replaced periodically. This will involve the use of a transfer cask to transfer this highly activated aluminum component from the reflector vessel to an on-site long-term storage facility. Because of the length of the horizontal through-tube, the cask used for the through-tube will be different from that used for the hot and cold sources. It is hoped, for example, that the through-tube cask will have an outside diameter of less than 760 mm and that it will weigh less than 21,000 kg. The actual cask is yet to be designed, and it is not known if these objectives can be met. Evaluations similar to those identified for the cold source and hot source thimble casks will be performed.

3.11.3.4.13 Monochromator/drum shielding issues

Three of the monochronomator drums (T-1, T-11, H-4) are normally located immediately adjacent to the biological shield, while six others (H-2, T-3, T-4, T-6, T-8, and T-12 in Fig. 3.37) are located ~6 to 9 m away from the biological shield. These are connected to their respective horizontal thermal beam port shield plugs via long glass guide tubes. The monochronomator drums shown in Fig. 3.38 are large: 2.9-m high × 31-m-diam steel drums having a solid thick immovable bottom section, a solid thick immovable top section, and a complicated thick-walled midsection (composed of many pieces) that can rotate about a vertical centerline. Circumferentially, two-thirds of this midsection is solid steel. A 1-m-thick rear wall has a single, fixed large hole in the back that serves as the exit collimator; it allows neutrons (within a very narrow energy band scattering in characteristic directions off the crystal inside the monochronomator's cavity) to exit the drum and continue along a secondary flight path toward the adjacent experimental station. Circumferentially, the remaining pie-shaped third of this midsection is divided into two series of 1-m-long, relatively thin, pie-shaped wedges. These can move up or down vertically as the midsection of the drum is rotated to select neutrons of different energies that scatter at different angles off the crystal located in the drum's internal cavity. The wedges are all split at the midplane (in a staggered, stepwise fashion) so that those above the midplane are attached to and ride on one circumferential track, while those below the midplane are attached to and ride on another circumferential track. Near the long glass guide tube conducting the neutron beam from the reactor to the monochronomator, the two tracks separate, with the top one moving up and the lower one moving down. Thus, as the midsection of the drum is rotated to select neutrons of different scattering angles, those wedges in that portion of the drum facing the incoming neutron beam will always be open to the incoming beam regardless of how much or how little the midsection of the monochronomator drum is rotated.

A mixture of semi-analytic methods, 1- and 2-D discrete ordinate analyses, and 3-D Monte Carlo analyses will have to be used for these various problems. Semi-analytic methods or Monte Carlo methods may be used to estimate the divergence of the beam entering the monochronomator. One- or (probably) two-dimensional codes may be used to ascertain the distribution of scattered radiation within the cavity. Two-dimensional discrete ordinate or 3-D Monte Carlo codes may be used to estimate the amount of radiation streaming out through the exit collimator or backscattering out through the entrance of the monochronomator. The actual methods used in each case will be selected as the analysis proceeds.

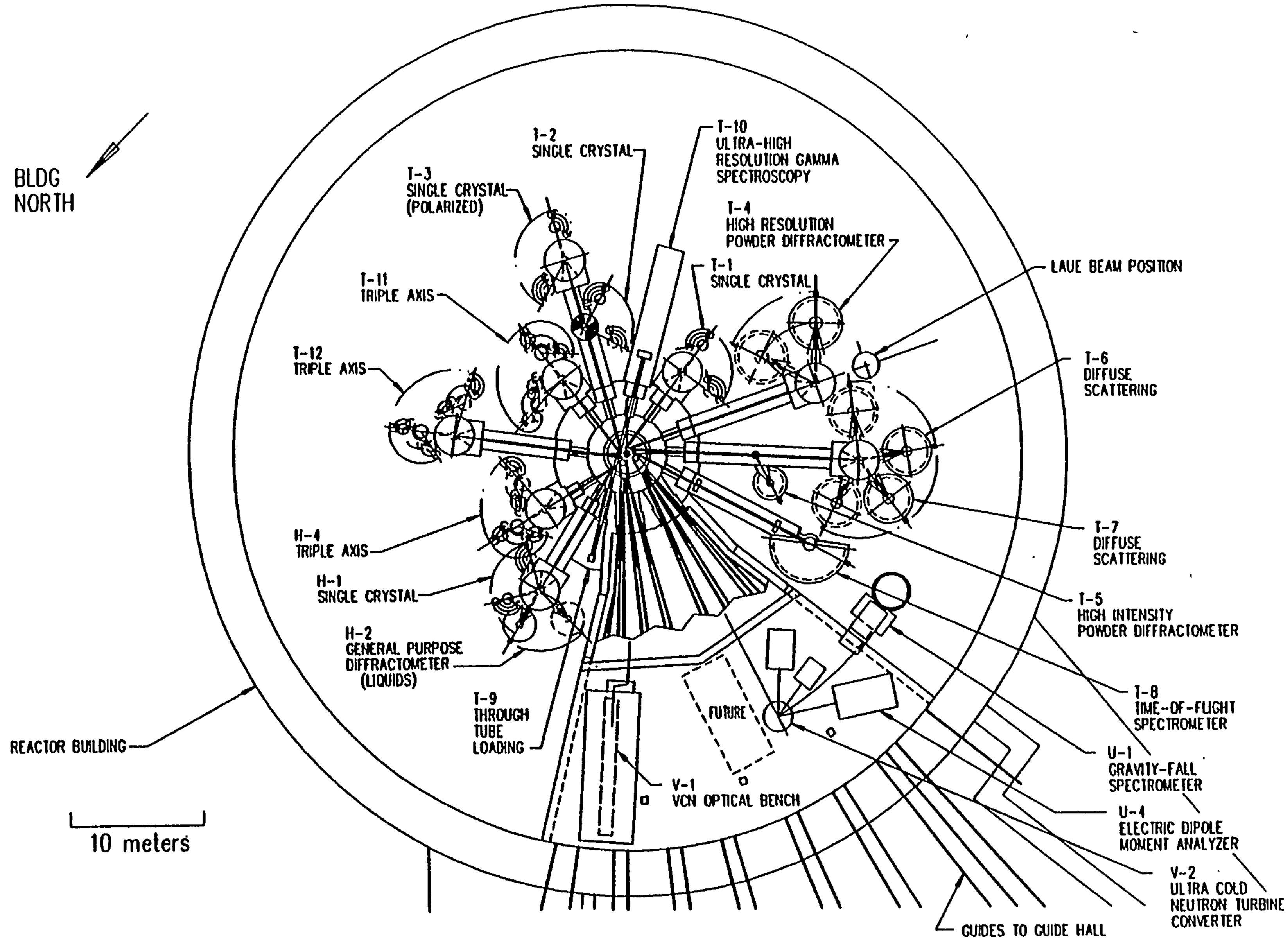


Fig. 3.37. Plan view, ground floor beam room.

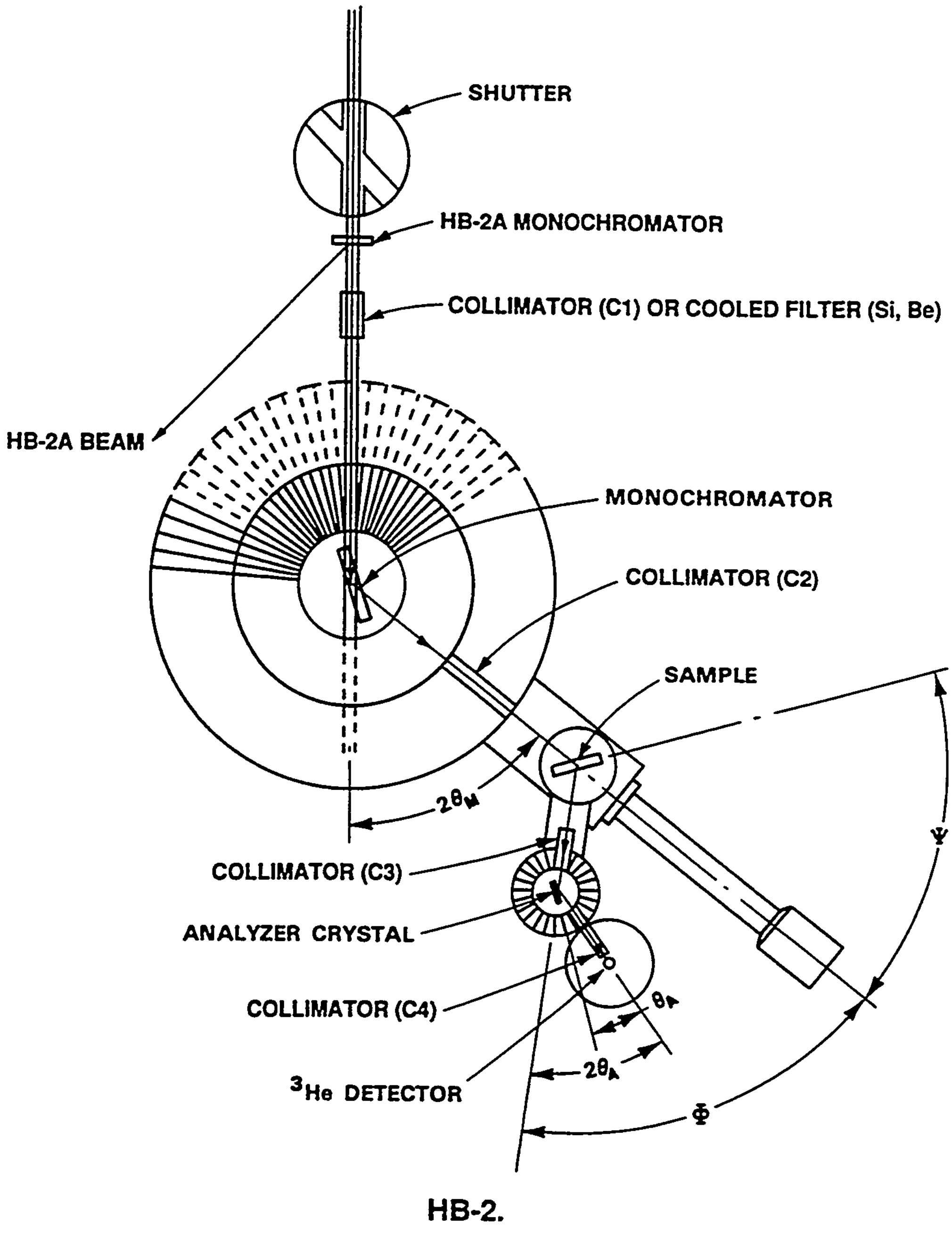


Fig. 3.38. Monochromator drum.

3.11.3.4.14 Typical thermal beam installation beam stop

Downstream of the beam emerging from the monochronomator's exit collimator, there may be one or more sets of scattering experiments. Downstream of the last experiment, some sort of beam stop (shield) will be needed to protect workers in this area. The thickness and type of material to be used is yet to be determined. Before this task begins, the beam intensity emerging from the monochronomator's exit collimator must be known. Some shielding analyses (or assumptions) must also be used to represent the effect of any experiment in the beam downstream of the monochronomator. Simple 1-D analyses will suffice to determine the required thickness of the beam stop material. The space-energy-angular distribution of the neutron and gamma fluxes backscattering off the beam stop should be archived for possible later use in making a general area map of the fluxes and dose rates in the beam room.

3.11.3.4.15 Typical cold instrument shielding

This subtask refers to shielding for the shallow angle neutron scattering (SANS) experiments (L-1, L-2, D-3, D-4, D-5, and D-6) and to the backscattering experiment (D-10) and other experiments (L-4, L-5, L-6, L-8, L-9, L-10, D-9, D-11 and D-12) that would be located in the rear of the guide hall, some distance from the reactor building, as shown in Fig. 3.39. While the fluxes in this area are generally believed to be quite small, this is regarded as a relatively clean area, and researchers will be working in proximity to the experimental equipment during normal operation. Since the fluxes here will be relatively small, the detectors used in these SANS experiments will have to be quite sensitive. For these experiments to be feasible, therefore, it is important that the detectors be adequately shielded against low-level background radiation coming from other experimental apparatuses in the guide hall.

3.11.3.4.16 Loading station shielding

The through-tube loading station is located in the reactor building, close to the biological shield. Figure 3.36 shows a crude schematic diagram of the loading station (a) with the interlock open for manual loading of the specimens in the carrier and (b) during the insertion or withdrawal of the carrier into the through-tube. Various irradiation targets (steel specimens, plastic specimens, or almost anything) may be placed inside a graphite carrier (also called a sample irradiation assembly) while the movable shield plug leading to the through-tube is closed. Once the movable shield plug is removed, the graphite carrier is inserted into the through-tube to the desired location.

It is envisioned that the types of shielding analyses required for the loading station will be similar to those previously described for the horizontal thermal beam tube shield plugs. If the final design calls for a single movable shield plug, or a rotating shield plug with just a single penetration, the analyses may be slightly simpler. The most difficult part of this analysis may be determining the source impinging on the through-tube loading station shield plug during normal operation, since the through tube does not dead-end in the reflector (like the horizontal thermal beam tubes).

3.11.3.4.17 ISOL station on LSBT shielding

The ISOL facility (T-13) is located beyond the LSBT shield plug in a special room on the second floor of the reactor building as shown in Fig. 3.40. After the irradiation of a (fissile) target material in the beam, neutron-rich ionized fission fragments are collected and studied at the ISOL. Eventually an accelerator may be added to this facility, creating unique research opportunities with this type of charged particle. While the neutron and gamma fluxes and dosage rates downstream of

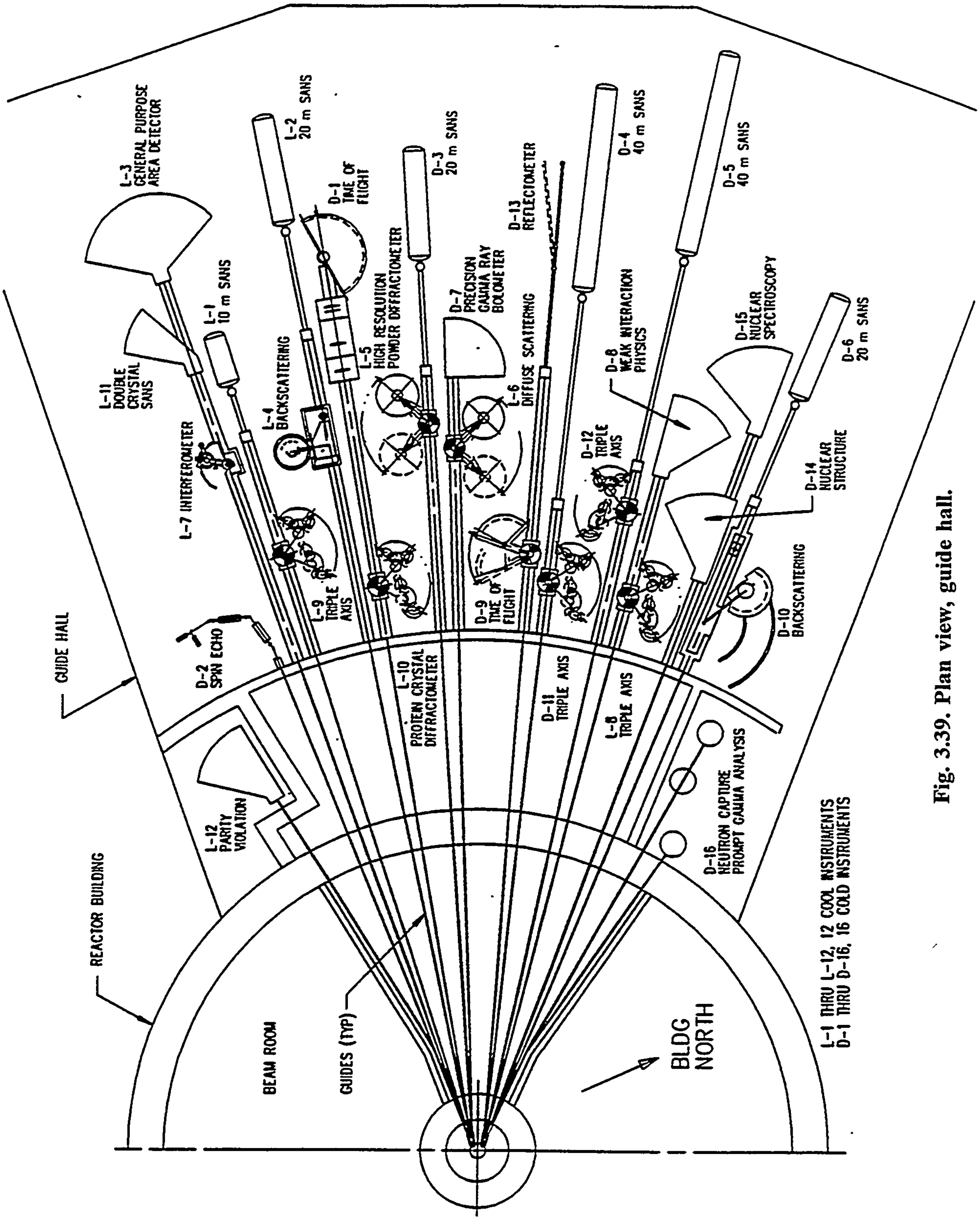


Fig. 3.39. Plan view, guide hall.

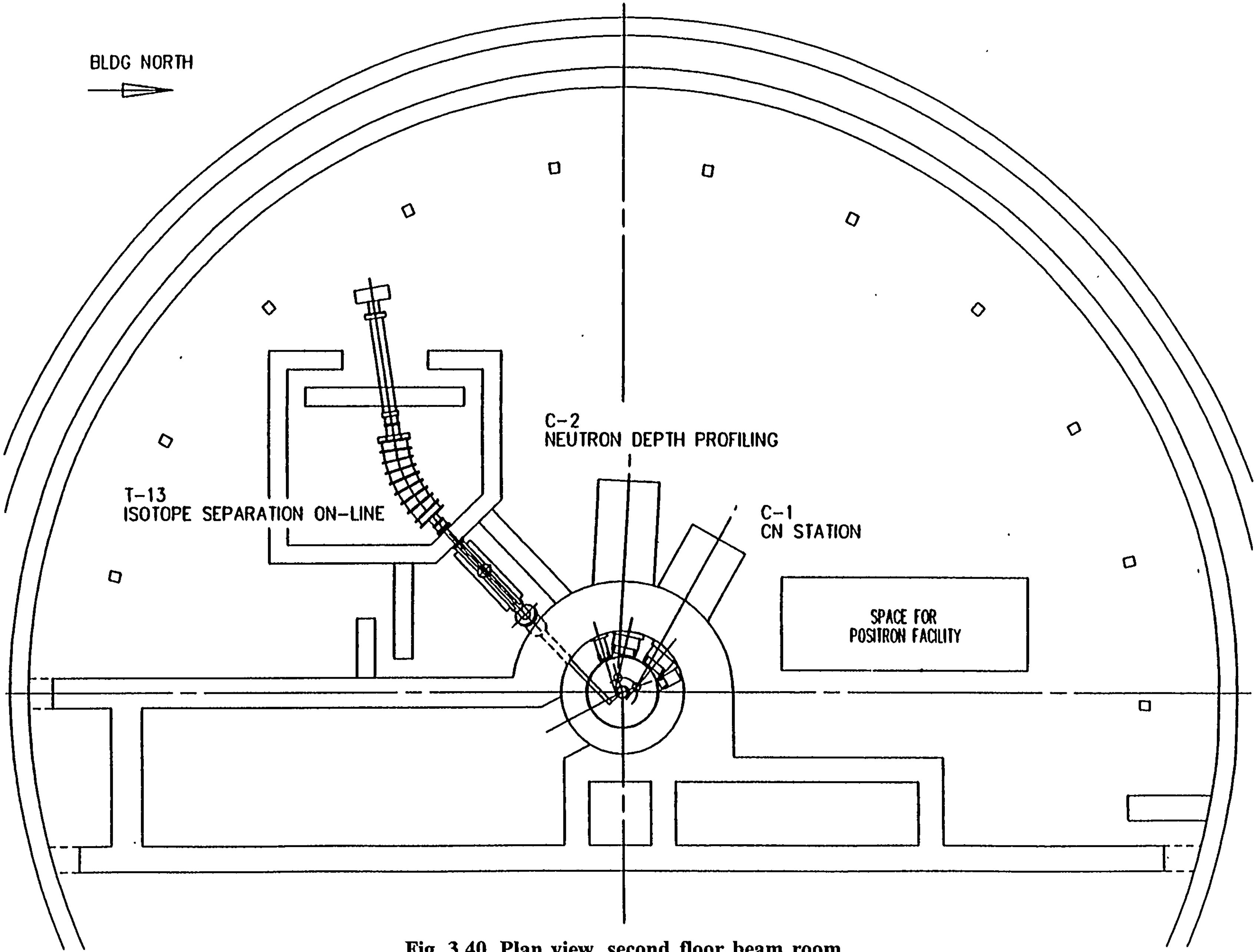


Fig. 3.40. Plan view, second floor beam room.

the LSBT shield plug (due to radiation streaming up the tube) will have been determined under the LSBT subtask, the fission products and fission fragments in the ISOL facility itself will require additional neutron and gamma shielding to protect personnel working in this area. This is certainly true for the material in the region of the ion source and for the material that accumulates on the condenser plates (downstream of the bending magnets). Shielding analyses would also have to be performed for the beam dump (stopping plate) downstream of the bending magnet.

A simple 1-D analysis of the beam stop should be adequate to determine its required thickness and the distribution of neutron and gamma radiation backscattering from the beam stop. As for the ion source and condenser regions of the ISOL facility, it is currently believed that ordinary 2-D RZ analyses (for each component separately) would probably be adequate to determine the neutron and gamma flux fields and dosage rates around these components and to determine the amount of additional shielding that might be required. Combining the dosage rate from these components with that due to radiation scattering off the beam stop should fully define the neutron and gamma dose rates in and around this facility.

3.11.3.4.18 Shielding for transuranic transfer

This subtask refers to determining shielding requirements for the transuranic rod assemblies as they are moved under water from one portion of the ANS facility to another. These small-diameter transuranic rods are ~0.9-m long. Up to 30 of these transuranic rods (similar to those used in HFIR) may be placed in a circular target cage assembly that will slip down over the outside of the lower fuel element (inside the CPBT) during irradiation, where they may be left for one or more irradiation cycles. These transuranic rods are initially filled with curium and, collectively, can produce up to 1.5 g of ^{252}Cf annually and up to 40 μg of ^{254}Es annually. During one of the refueling cycles, the circular target array with up to 30 such rods will be lifted up through the D_2O-filled transfer shaft, moved laterally through the D_2O-filled transfer canal, and placed in the D_2O-filled interim fuel storage canal. Subsequently, it will be lowered through a D_2O/H_2O lock into a light water pool. From there, it will be moved to an underwater handling station where the target cage assembly will be dismantled and a fresh array of transuranic target rods assembled. The irradiated transuranic rods will be moved in and out of the reactor building through the spent fuel pools and transfer tunnels. Eventually these transuranic rods will be loaded into casks in the fuel handling area of the reactor support building for shipment back to the Radiochemical Engineering Development Center (REDC).

Basically, one or two generic shielding calculations will be performed to verify that the amount of D_2O or H_2O covering the transuranic rod assemblies is always sufficient to reduce the neutron and gamma fluxes and dosage rates to acceptable levels on the floor of the refueling deck, where personnel will be actively working during the refueling operation. Separate calculations should be performed for the rods in D_2O and H_2O. In the first case, because of high-energy gammas coming from the irradiated transuranic rods, the shielding calculations should explicitly account for photoneutron production in the heavy water. In both cases, a simple 1-D fine-mesh discrete ordinates shielding calculation (with the rods modeled as a simple sphere) should suffice; although follow-up 2-D RZ calculations with the rods modeled first as a flat disk and then as an annular source might also be desirable for verification purposes.

3.11.3.4.19 In-core materials irradiation capsule transfer

Five instrumented and five noninstrumented materials irradiation capsules are distributed around a circular support assembly that, during normal operation, is located directly above the core's lower fuel assembly, in the D_2O-filled region just inside the upper fuel assembly element. Instrumented capsules are 48 mm in diameter, and the noninstrumented capsules are 16 mm in diameter. Both types of capsules are 0.5-m long. During one of the refueling operations, all 10 irradiation capsules and the circular support assembly are removed as one assembly by the refueling system, which lifts the circular support assembly up through the D_2O-filled transfer shaft and moves it laterally through the D_2O-filled transfer canal. From there it is then hoisted up into the 10-m × 10-m nitrogen-filled equipment transfer and manipulator cell located above the spent fuel storage pool. Then the assembly is lowered into a light water pool, from which it is moved to the storage pool experiment handling facility where the irradiation capsules are removed from the circular support assembly and disassembled using long-handled tools.

At least three different types of shielding calculations must be performed to verify/ensure that the dosage rates on the refueling deck are below acceptable levels.

1a. As the irradiation capsules are moved through the D_2O-filled transfer canal, calculations must be performed to verify that the dosage rates at the surface of the pool are acceptable. If any of the gammas emanating from the capsules are known to be above 2.25 MeV, photoneutron production in the D_2O-filled pool must be explicity accounted for. One-dimensional calculations may suffice for this analysis.

1b. As the 10-capsule array is being hoisted into the equipment transfer and manipulator cell, 2-D calculations must be done to ensure that the operation is being carried out far enough from the concrete walls of the cell that the radiation shining up through the D_2O will be adequately attenuated by either the D_2O or the concrete walls of the cell. This is to ensure that dosage rates on the refueling deck will remain at or below acceptable levels.

2a. While the array is in the nitrogen-filled equipment transfer ANS manipulator cell, calculations must be done to ensure that the concrete walls of the cell are thick enough that dosage rates on the refueling deck will remain at or below acceptable levels. If only gamma radiation is involved, then the QAD point gamma shielding code[50] would be the recommended approach for the calculation. It could account for the sources in the 10 irradiation capsules in the cell, as well as geometry associated with the flat 0.61-m-thick concrete walls of the cell. (Alternately, the point kernel option in MCNP could be used for this calculation, although it would offer no particular advantage over QAD in this particular situation.)

2b. In addition, a 3-D MORSE or MCNP calculation should be performed to evaluate the effect of a 1-m × 1.65-m hole in one of the concrete walls of the cell (above water level), which is presently needed to admit a rail-guided remote hoist trolley into the cell (above water level). This large hole in the cell wall for the trolley is expected to allow unacceptably large amounts of radiation to escape from the cell into the refueling area where people would be working concurrently. In any event, a 3-D Monte Carlo analysis of this area will probably be needed to account for the 3-D geometry, as well as reflection of radiation off the surface of the water and other surfaces near the opening.

3. Finally, a set of calculations similar to (1a) and (1b) above must be performed for the case when the 10-capsule array is being lowered from the nitrogen-filled cell back into the light water side of the spent fuel storage area. These calculations, however, will be somewhat simpler than those in 1a and 1b because there will be no need to account for photoneutron production in this area. Because this assembly will necessarily be lowered into the light water pool at a point close to the cell wall, the primary focus will be on calculating radiation shining up from the water and shining out underneath the concrete wall of the cell as the 10-capsule assembly is first lowered into the light water. If only gamma radiation is involved, the QAD point kernel code should be adequate for this situation.

3.11.3.4.20 Slant hole capsule water shielding for transfer

The slant hole facility is located on the third floor of the reactor building, across from the top of the light water pool. Various irradiation capsules are mechanically hoisted into this water-filled area. From there, they may be transferred to adjacent hot cells. One of these cells is equipped to cut off the sample end of a material irradiation capsule and transfer the sample to a second hot cell for shipping in a cask or through the canal system. Note that the stainless steel irradiation capsules brought to the top of the slant hole facility are typically very radioactive, since they previously have been irradiated near the bottom of the slant hole (SH-1), which is located directly across from the top fuel element in the core, just 330 mm from the centerline of the system. Presumably, 1-D (spherical) or 2-D RZ shielding analyses will be adequate for analysis purposes once the source terms emanating from the capsule(s) are known. While the shielding task could calculate the activation levels associated with the stainless steel body of the irradiation capsules themselves, the contents of the capsules (and hence the radiation levels associated with the contents) are generally not known in advance. Before the "worst case" shielding analysis, therefore, the experimental task designers would have to define a baseline reference sample material that could be used in activation analyses, and/or directly provide the shielding task with the neutron and gamma source terms associated with some hypothetical "worst case" material.

3.11.3.4.21 Water shielding for facility tube transfer

Slant hole facilities SH-1 and SH-2 allow irradiation capsules to be exposed to a relatively hard neutron spectrum just 50 or 60 mm away from the CPBT, near the midplane of the core. Note that these aluminum tubes terminate just 330 mm from the centerline of the system. The aluminum tubes themselves must be replaced periodically because of radiation damage. During that process, they must be removed through the light water above the reflector vessel and stored in an underwater storage facility. Because these aluminum tubes will contain only activation gammas and will be transported only through light water, a fairly simple QAD point kernel shielding analysis should be adequate. ORIGEN-type activation analyses must be performed to determine the activation levels along the length of the tube. Because the entire length of the tube inside the reflector vessel (not just that portion close to the core) will be exposed to very high thermal flux levels, it is expected that a fairly long portion of the tube will be highly activated. Thermal flux levels required for this activation analysis may be obtained from the global 2-D DORT analysis of the core and reflector vessel performed at ORNL, or from the 3-D MCNP analysis performed at INEL.

3.11.3.4.22 Water shielding for isotope target transfer and pneumatic tube transfer shielding

A series of hydraulic and pneumatic rabbit tubes are provided for the production of industrial and medical isotopes, as well as for material irradiation studies. After irradiation, capsules in these rabbit tubes will be moved to various underwater handling facilities before shipment. These subtasks are concerned with verifying that the capsules are always covered with sufficient light water to ensure that the dosage rates above the respective pools and transfer channels are acceptable and pose no hazard to personnel working in the area. Standard 1- and 2-D shielding codes will suffice for these analyses.

3.11.3.4.23 Transfer and storage of spent fuel

This subtask refers to determining shielding requirements for the spent fuel elements as they are moved (under water) from one portion of the ANS facility to another. At the end of each cycle, the spent fuel elements are lifted up through the D_2O-filled transfer shaft, moved laterally through the D_2O-filled transfer canal, and placed in the D_2O-filled interim fuel storage canal. Subsequently, they will be lowered through a D_2O/H_2O lock into a transfer cask in the light water storage pool.

Basically, one or two generic shielding calculations should be performed to verify that sufficient heavy water is always covering the spent fuel elements to reduce the neutron and gamma fluxes and dosage rates to acceptable levels on the floor of the refueling deck, where personnel frequently will be working. At present, the closest approach (i.e., minimal heavy water coverage) would appear to occur when the spent fuel elements are being moved through the 5.8-m-deep transfer canal to the interim fuel storage canal. Because of high-energy gammas coming from the spent fuel, it is imperative that the shielding calculations explicitly account for photoneutron production in the heavy water. Preliminary calculations for the spent fuel assemblies in this region have shown that the neutron dosage rate due to photoneutron production may be 30 to 35 times greater than the gamma dosage rate and several orders of magnitude greater than the neutron dosage rate if photoneutron production were not accounted for. Preliminary shielding calculations for the spent fuel in the transfer canal have shown that the dosage rates at the surface of the pool were unacceptably high by two or three orders of magnitude.

3.11.3.4.24 Cask for the spent fuel

Preliminary shielding calculations will be necessary before a spent fuel shipping cask can be designed for the transport of spent fuel off-site in accordance with all Nuclear Regulatory Commission and U.S. Department of Transportation regulations.

3.11.3.4.25 Transfer and storage of irradiated control rods and the CPBT

The following comments apply to both the irradiated control rods and the activated CPBT, both of which must be replaced periodically. The irradiated control rods (inner and outer) and the activated CPBT must be removed periodically by the refueling system, which will lift them through the D_2O-filled transfer shaft (silo) and move them laterally through the D_2O-filled transfer canal. From the canal, they will be hoisted up into the 10-m × 10-m nitrogen-filled equipment transfer and manipulator cell located above the spent fuel storage pool. After possible inspection, these components will be lowered into the light water storage pool.

At least three different types of shielding calculations must be performed for each component to verify/ensure that the dosage rates on the refueling deck are at or below acceptable levels.

1a. As these components are moved through the D_2O-filled transfer canal, calculations must be done to verify that the dosage rates at the surface of the pool are acceptable. If any of the gammas emanating from these components is known to be above 2.25 MeV, photoneutron production in the D_2O-filled pool must be explicitly accounted for. (This may not be a problem for the activated CPBT after shutdown, but it should be considered for the hafnium control rods.) In either case, 2-D RZ calculations will likely be required for these rather large (long) components.

1b. As these components are being hoisted into the equipment transfer and manipulator cell, another set of 2-D calculations must be performed to ensure that the operation is being carried out far enough from the concrete walls of the cell that the radiation shining up through the D_2O will be adequately attenuated by either the D_2O or the concrete walls of the cell. This is to ensure dosage rates on the refueling deck will remain at or below acceptable levels.

2a. While the components are in the nitrogen-filled equipment transfer and manipulator cell, calculations must be performed for both the CPBT and the irradiated rods to ensure that the concrete walls of the cell are thick enough that dosage rates on the refueling deck will remain at or below acceptable levels. If only gamma radiation is involved, then the QAD point kernel code would be the recommended approach for this calculation. It could account for the spatial distribution of the activated sources in these components inside the cell, as well as the geometry associated with the flat 2-ft-thick concrete walls of the cell.

2b. In addition, a 3-D MORSE or MCNP calculation should be performed to evaluate the effect of the 1-m × 1.65-m hole in one of the concrete walls of the cell (above water level), which is needed to admit a rail-guided remote hoist trolley into the cell (above water level). This large hole in the cell wall for the trolley is expected to allow unacceptably large amounts of radiation to escape from the cell into the refueling area where people would be working concurrently. A 3-D Monte Carlo analysis of this area will probably be needed to account for the 3-D geometry, as well as reflection of radiation off the surface of the water and other surfaces near the opening in the cell wall for the trolley.

3. Finally, a set of calculations similar to (1a) and (1b) must be done for the case when these components are being lowered from the nitrogen-filled cell into the light water side of the spent fuel storage area. These calculations, however, will be somewhat simpler than those in 1a and 1b because there will be no need to account for photoneutron production in this area. Because this assembly will necessarily be lowered into the light water pool at a point close to the cell wall, the primary focus will be on calculating radiation shining up from the water and out underneath the concrete wall of the cell as these components are first lowered into the light water (but are still relatively close to the surface). If only gamma radiation is involved, the QAD point kernel code should be adequate for this situation.

3.11.3.4.26 Biological shield composition

Early scoping calculations for the 2.44-m-thick concrete biological shield surrounding the reflector vessel vault were performed using different amounts of steel rebar (0 to 20 vol%) in the concrete. With no steel rebar, the gamma dosage rate on the outside of the homogeneous biological shield (with no penetrations) was above the allowable limit (0.25 mrem/h). To reach that limit, 10 to 15 vol% steel rebar was found to be necessary. While there has been some concern as to whether that amount of rebar is practical, there is little question that heavier concretes such as barytes concrete (or other concrete mixes with iron-bearing ores such as hematite or limonite) could be used in conjunction with lesser amounts of steel rebar to achieve an effective biological shield design.

3.11.3.4.27 Beam room shielding

Dosage rates in the beam room will be determined to a very modest extent by radiation coming through the biological shield, and to a much larger extent by radiation coming into the beam room through the various experimental facilities. To that extent, this general reference to "beam room shielding" is somewhat redundant. Eventually, after all the subtasks described previously are complete, it may be possible to manually assemble a composite map of the neutron and gamma dosage rates throughout the beam room. Because each of these components would have to be analyzed separately and would require the use of many diverse calculational techniques whose results cannot be superimposed "automatically," the level of effort required to generate a single dosage map for the entire beam room manually may not be justified. It would certainly have to be considered a lower priority task.

3.11.3.4.28 Subpile room shielding

The subpile room shielding subtask initially referred to determining neutron and gamma fluxes and dosage rates in the subpile room, which corresponds to the concrete vault directly below the one containing the reflector vessel. Neutron and gamma radiation enters this area through the lower portion of the 0.506-m-diam CPBT. The CPBT contains the lower portion of the inner control rod and inner control rod drive mechanism, but it is largely filled with heavy water (plus a small amount of light water in the bottom portion of the assembly near the bellows region). Originally, the objective was to determine if maintenance operations could be safely performed in the subpile room during normal operation or after shutdown. Subsequent to the initial full power calculations, the primary focus shifted and is now on minimizing radiation damage (helium production, displacements per atom, and subsequent embrittlement) to the large primary coolant supply adaptor (PCSA) located between the bottom of the reflector vessel and its concrete vault. Initial shielding calculations have shown that the neutron fluences in this region could severely compromise the planned 30-year lifetime of this permanent structural component, which cannot easily be removed or replaced.

Shielding calculations in this area can generally be performed using a 2-D RZ DORT model of the system. The starting point for this analysis would be a space-energy-angle-dependent boundary flux across a horizontal plane halfway between the bottom of the reflector vessel and the bottom of the lower fuel element in the core. These data can be and have been generated from a global 2-D RZ analysis of the entire 2-element core and reflector vessel. Because of the D_2O in the reflector vessel and the lower portion of the CPBT extending down to the subpile room, it is extremely important to account for photoneutron production during normal operation. Even after shutdown, photoneutron production in the D_2O (during the determination of the boundary source and in the subpile or PCSA analysis) would need to be accounted for. This is true as long as the spent fuel remains in the core. After the fuel and other high-energy activation sources are removed, it may not be necessary.

3.11.3.4.29 Second floor shielding—increased pool size and third floor shielding—increased pool size

At present, the 2.44-m-thick concrete walls of the biological shield extend up to the top of the light water pool on the third floor of the reactor building. The objective of these proposed studies would be to investigate (from a shielding viewpoint) the feasibility of decreasing the thickness of these concrete walls at the second and third floor levels, thus expanding the effective inside diameter of the vault as it gets farther from the reactor. This decrease would have the advantage of providing extra light water storage areas on the ledges formed by these two axial offsets. While the required thickness of the concrete vault walls at these axial locations will most likely be set by structural considerations, some shielding calculations should also be performed to ensure that dosage rates on the opposite side of the concrete vault walls do not exceed allowable limits under normal operating conditions.

Ordinarily, the many feet of water above the reflector vessel would provide more than adequate shielding against the neutrons leaking from the vessel; and simple QAD-type point kernel calculations could be used to evaluate the gamma attenuation through the water and the required thickness of the concrete vault walls at different axial levels, once the boundary source at or slightly above the reflector was determined from the large global analysis of the core and reflector. Alternately, a more costly but more accurate 2-D RZ DORT analysis could be performed. If the boundary source just above the reflector vessel showed a large number of high-energy gammas above 2.25 MeV, then photoneutron production in the D_2O-filled refueling silo, which will always be present, would have to be accounted for. Because of the long mean free path of high-energy gammas in either heavy or light water, these high-energy neutrons have the potential to create secondary high-energy photoneutron sources in the D_2O-filed silo at distances far above the reflector vessel. This effect could not be represented in any point-kernel analysis and would therefore require a complex 2-D RZ discrete ordinate shielding analysis in which photoneutron production was explicitly accounted for. On the other hand, if it can be shown that the boundary source some distance above the reflector vessel does not contain any gammas above 2.25 MeV, then the simplistic QAD-type point kernel analysis should be adequate. In that case, even if a 2-D RZ DORT analysis were performed to confirm the QAD results, significant calculational economies could be realized since photoneutron production would no longer have to be accounted for.

3.11.3.4.30 Shielding to allow draining light water

After the fuel and the inner portion of the CPBT have been removed from the core, it may be necessary to drain the D_2O from the refueling silo and CPBT and drain the light water from the reactor vault to perform some remote maintenance operations. At such times, the reflector vessel will still contain many activated components such as the vessel itself; the beam tubes, slant tubes, and other rabbit tubes; as well as the outer CPBT, the control rods, the cold source, and the highly activated Zircaloy shells and support structures associated with the hot source. With the reflector vessel vault completely drained, the dosage rates at the refueling deck on the third floor are expected to exceed allowable limits by several orders of magnitude. Before the reflector vessel concrete vault could be drained, therefore, temporary shielding would have to be installed to protect personnel working on the refueling deck. The amount of temporary shielding that would be required under those conditions is still to be determined. Given the absence of any neutron sources and/or any D_2O, simple QAD-type point kernel analyses should be adequate.

3.12 INSTRUMENTATION AND CONTROLS SYSTEM DEVELOPMENT

The ANS operational objectives place unusual demands on the operating regimes that require unique control and plant protection system capabilities. The instrumentation and control systems development activity examines the capability of different controls and reactor protection systems (RPSs) to meet operational and safety needs. The extent to which special control and RPS capability can be developed will affect the reactor design concepts and its eventual operational capabilities.

This WBS element contains one major project milestone:

> Complete prototype tests of the RPS by the end of June 1999. This subtask provides the final confirmation of the performance of the RPS in time for design adjustments, if necessary.

The instrumentation and control system development activity is divided into three WBS level four tasks summarized in Table 3.26. 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 8-year period covered by this R&D plan are shown in Table 3.27 and the associated schedules in Fig. 3.41. The initial development work would be performed using expense money, and the later work would be performed using line-item money. Capital equipment money would be used to construct test facilities and to purchase test equipment. Subsections 3.12.1 through 3.12.3 provide more detailed information on the WBS level four tasks under this activity.

Table 3.26. 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 systems. The primary purpose of the model is to aid in the development of safety and control features, but the model also is useful for aiding in design decision of thermal-hydraulic parameters. Although the basic model has already been developed, this modeling is expected to continue throughout the project and will be modified as changes in the reactor system design occur. This task also provides support for 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.
1.1.12.2	Protection and control development—This task provides the logical arrangement and integration of reactor control and protection features into the plant design. It includes the following activities: reactor control—developing the essential features and performance requirements for the reactor and heat removal systems; reactor protection system—developing the requirements and analyzing the adequacy of ANS protection systems and developing requirements for components that are not commercially available (e.g., release magnets, rod latches, control rod performance requirements); experiment systems—developing requirements for the performance and interface of experiment facilities, including the cold sources, with the reactor control and protection systems; diagnostics and surveillance—developing requirements and methods for a reactor monitoring system to provide computation, diagnostic, and display capability for timely, concise, and directed information for operator aids, engineering diagnostics, maintenance, and administration; subcriticality monitoring—developing the requirements and identifying the techniques necessary for subcriticality monitoring of the reactor core, both installed and during refueling operations; and control integration—developing the philosophy, requirements, and techniques for overall integration of plant control features, protection systems, and human factors of plant operation.
1.1.12.3	RPS component development—This task establishes requirements for sensors, instruments, and actuators for protection and control. It includes the following activities: sensor development—establishing requirements and initiating development of nuclear and process sensors to meet the special needs of the ANS where direct application of commercial products is not possible, establishing qualification requirements; instrumentation development—defining systems, establishing requirements, and developing special instrumentation for the protection and control systems, evaluating suitability or adaptability of commercially available instrumentation; actuator development—developing requirements and details of special actuators that may be required for protection or control, (e.g., scram rod release magnets, position indicators, hydraulic actuators).

Table 3.27. WBS level four breakdown of costs for the instrumentation and controls system development activity

WBS Level 3	Level 4	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.12		Instrumentation and Control System Development										
	1.1.12.1	Dynamic model development and analysis	Exp.	161	97							258
			Line		59	108	67	67	67	34		402
			Cap.									0
	1.1.12.2	Protection and control development	Exp.	244	361	183	152					940
			Line	101	101	101	52	26	26	10		417
			Cap.		81	325	271					677
	1.1.12.3	Reactor protection system component development	Exp.	134	162	96	76					468
			Line	46	45	63	67	67	67	34		389
			Cap.									0
		Subtotals	Exp.	539	620	279	228	0	0	0	0	1666
			Line	147	205	272	186	160	160	78	0	1208
			Cap.	0	81	325	271	0	0	0	0	677
		Contingency	Exp.	27	62	28	23					140
			Line	7	41	54	37	32	32	16		219
			Cap.		16	65	54					135
		Total	Exp.	566	682	307	251	0	0	0	0	1806
			Line	154	246	326	223	192	192	94	0	1427
			Cap.	0	97	390	325	0	0	0	0	812

Fig. 3.41. Schedule for WBS 1.1.12 instrumentation and control system development.

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 is important to advanced conceptual design as the reference design is modified. It also is important for studying control system concepts for start-up, normal operation, shutdown, transients, and accidents. A summary of the uses of the dynamic model for this task is included in Table 3.26.

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

3.12.1.2 Description of the Dynamic Model Task

A dynamic model of important characteristics of the ANS systems is being designed, updated, and employed. The model is primarily to aid in developing 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 involves 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 includes a continuing effort to validate the dynamic model.

This task also supports 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 focus 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.

3.12.1.2.1 Status

The dynamic model has been used to determine the response and motion requirements of the RPS and the control and safety rods. The requirements are based on a range of potential accidents identified in the CSAR and the reference conceptual design data from other project tasks. The mechanical response characteristics required of the safety rods have been specified for the mechanical designers using the dynamic model data. The model is revised from time to time as new or refined information becomes available regarding core reactivity, rod worths, and thermal-hydraulic margin considerations. The model also has been used to evaluate control characteristics of the primary and secondary coolant loops. The dynamic model has been used interactively with the thermal-hydraulic designers to evaluate design options and to assess the importance of small and large pipe breaks.

3.12.2 Protection and Control Development

3.12.2.1 Justification for the Protection and Control Development Task

Control of the ANS requires strategies different from those for commercial plants and for HFIR. Analysis of the 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 this task is analyzing the capability of the control system to avoid accidents or avoid the necessity to initiate safety systems. Because the ANS thermal-hydraulic and neutronic dynamics are faster than those of commercial plants, the response times and data throughput requirements must be analyzed to determine what is adequate for the ANS control system.

Special components in the RPS 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 RPS task is included in the R&D effort because it requires development of nonstandard components with requirements that commercial reactors do not have.

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

A reactor monitoring system will use plant computers [the Plant Control and Data Acquisition System (PCDAS)] as in commercial power plants, but it will be advanced in implementation and different because of special ANS requirements. R&D is needed to integrate it into the plant control system and to differentiate its requirements from those of a commercial system.

A control integration task provides bases for meeting the high performance specifications for the plant control systems. Safety and nonsafety issues will be examined to provide information for the PSAR and FSAR. This information is important for meeting anticipated regulatory requirements.

3.12.2.2 Description of the Protection and Control Development Task

A summary of the protection and control tasks is included in Table 3.26. Details of these tasks are presented in the following paragraphs.

A control system task is 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' needs 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. This network will be developed in more detail as the control system develops.

A protection system task is to develop the requirements for and to analyze the adequacy of ANS protection systems. In addition, this task will provide the development of requirements for components that are not commercially available (e.g., release magnets, rod latches, and ANS control rods). The RPS 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 is to develop a protection system that is acceptable to the regulating agency; therefore, some emphasis will be on identifying those features most likely to cause problems because of their uniqueness. The performance of the RPS 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 RPS.

This task includes providing the requirements for the performance of the experiment facilities, including the cold sources, and their interfaces 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.

A 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 ANSR 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 systems 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 ANSR monitoring system will be identified and compared with the requirements of the Advanced Light Water Reactor monitoring system.

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 integrated systems being considered for commercial nuclear plants upgrading their instrumentation and control systems and for those proposed for the Advanced Light Water Reactor will be evaluated in this task. Commercial experience will be used to improve the design of the ANS instrumentation and control systems. The safety issues concerning the potential failure modes will be evaluated. 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.

3.12.2.2.1 Status

Detailed RPS requirements have been developed and documented in the system design description (SDD) 33. Operational modes and control requirements have been evaluated and documented in SDD 33 and in technical specifications prepared for the RPS and for the PCDAS, which includes the integrated control features for the plant. It is recognized that commercial systems are not available that will satisfy the response requirements of the RPS for the ANS. It is recognized also that software-coded digital systems with protective functions are difficult to license within the current philosophy of the Nuclear Regulatory Commission. Work continues to develop application-specific integrated circuits (ASICs) for use in the RPS, which have the potential for solving both of these problems.

3.12.3 Reactor Protection System Component Development

3.12.3.1 Justification for the RPS Component Development Task

The flux detectors require special consideration because they are mounted underwater; they are in extremely high neutron flux and high delayed gamma flux; and they require fast time response. This application is different 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.

The nuclear industry has very special protection and control requirements, but the 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.

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

3.12.3.2 Description of the RPS Component Development Task

The sensor development task will establish requirements for and initiate development of nuclear and process sensors 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 (e.g., 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 is 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. Its accuracy and repeatability have a direct effect on the flux delivered by the reactor.

A subtask will define systems, establish requirements, and develop special instrumentation for the protection and control systems. One subtask is 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 are 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.

Another task is to develop the requirements and details of special actuators that may be required for protection and control (e.g., 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.

3.12.3.2.1 Status

Locations have been identified for the application of commercially available ionization chambers and fission chambers to monitor reactor power. These detectors are to be located in the light water pool in special waterproof housings that provide protection and permit appropriate movement and positioning of the detectors. These locations compete for space with the beam tubes and experiment facilities, and some adjustments are being made to avoid interferences. Investigations have determined that self-powered neutron detectors can be applied effectively using thimbles in the reflector vessel. These detectors will provide diversity, improved response, and independence from other interferences.

Studies have been made, in conjunction with instrument vendors, to evaluate the limitations of resistance thermometer response in thermowells. It has been determined that the original specifications proposed for the ANS cannot be met with thermowells. Direct immersion sensors, as used originally in the HFIR, are extremely undesirable in the ANS because of the potential for heavy water leakage. Careful evaluation of control and protection responses has resulted in relaxing the temperature response requirements so that specially adapted thermowells and resistance thermometers can be used with response in the order of 5 s instead of the original 2-s response that was sought.

Studies of flow measuring devices have concluded that a venturi is needed to achieve both the response and absolute accuracy required for the ANS. Interaction with the conceptual design architect/engineer has resulted in piping layouts that provide the necessary straight sections of pipe, free of valves or obstacles, both upstream and downstream of the venturi to permit achieving the needed accuracy. Acoustic flowmeters have been investigated and may be usable for applications where reduced accuracy can be tolerated. This type of flowmeter is desirable because pipe penetrations are not needed.

The space available in the inner control rod scram release latch assembly is very limited and calculations indicated that it may not be possible to achieve adequate holding force. Two proof-of-principle magnets were fabricated in the maximum size that would fit the conceptual inner rod design, and the magnets were tested for force and release response. The tests confirmed that the magnetic core material used for HFIR magnets (50% ferro-nickel) would not have adequate force margin, but that an alternative material (2% silicon-steel) commonly used in transformers would have adequate holding force margin and acceptable release time characteristics.

3.13 FACILITY CONCEPTS

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

This task contains one major milestone:

Complete the reactor component evaluation tests by December 1999.

The facility concepts activity is divided into five WBS level four tasks summarized in Table 3.28. Most of this work would be performed at ORNL. The total estimated costs for this activity over the 8-year period covered by this R&D plan are given in Table 3.29, and the associated schedules are shown in Fig. 3.42. 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 3.13.5 provide more detailed information on the WBS level four tasks under this activity.

3.13.1 Reactor Components Tests

3.13.1.1 Justification for the Reactor Components Tests

The failure of certain reactor system components to meet functional requirements could have significant impacts on plant safety and availability. 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 requirements.

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.

Table 3.28. Summary description of the reactor concepts work breakdown structure level four tasks

WBS	Task description
1.1.3.1	Reactor components tests—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, inner control rod tests, and beam tube thimble collapse tests.
1.1.3.2	Reflector vessel and core flow tests—These tests will provide flow distribution and velocity data for the reflector vessel and internal component design configuration. It will also include the CPBT. Of primary interest is the flow distribution through each fuel element, the control rod cavity, and the by-pass between the CPBT and the fuel element side plates.
1.1.3.3	Refueling components tests—This activity will test the capability 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.
1.1.3.4	Special tests—These tests are required for the irradiation capsule disconnect assembly, the 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.
1.1.3.5	Combined electrolysis and catalytic exchange demonstration tests—This activity will test certain key components of the ANS heavy water upgrade and detritiation facility at as close to actual operating conditions as practical. These components are the electrolysis cell, the catalyst, the trickle bed recombiners, and the combined electrolysis and catalytic exchange column control scheme.

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

Thimble tests are required to verify the capability of the various thimbles and beam tubes to survive pressure surges, and to investigate the pressure limits of operation. Installation and sealing methods will be developed and verified. Flow through thimble cooling shrouds will be characterized.

Tests of inner control rod components are necessary to characterize and confirm the performance of individual components that contribute to the overall performance of the inner control rod system. These include mechanical latch component wear, latch magnet response, scram deceleration, scram position indication, and lower high pressure seal performance.

3.13.1.2 Description of the Reactor Components Tests

This task provides those tests not covered by other R&D tasks that are necessary to develop or confirm the operational performance of reactor components. They include tests of seals, locking ring/bolt torque devices, outer shutdown rods, thimble tests, and inner control rod components. Descriptions of these tests are provided in the following paragraphs.

Table 3.29. WBS level four breakdown of costs for the facility concepts activity

WBS Level 3	Level 4	Description	Type	1996	1997	1998	1999	2000	2001	2002	2003	Total
1.1.13		Facility Concepts										
	1.1.13.1	Reactor components test facility	Exp.	2414	1275	239	236	87				4251
			Line	146	418	469	225					1258
			Cap.	223								223
	1.1.13.2	Reflector vessel and core flow tests	Exp.	415	666	630	187					1898
			Line									0
			Cap.									0
	1.1.13.3	Refueling components test facility	Exp.		79	53	157	131	91			511
			Line			222	692	156				1070
			Cap.		39	438	216					693
	1.1.13.4	Special test facility	Exp.	881	760	691	469	1356	1177	1159	253	6746
			Line			371	529	114				1014
			Cap.									0
	1.1.13.5	CECE Demonstration	Exp.									0
			Line	165	1617	976	301					3059
			Cap.									0
		Subtotals	Exp.	3710	2780	1613	1049	1574	1268	1159	253	13406
			Line	311	2035	2038	1747	270	0	0	0	6401
			Cap.	223	39	438	216	0	0	0	0	916
		Contingency	Exp.	186	278	161	105	157	127	116	25	1155
			Line	16	407	408	349	54				1234
			Cap.	45	8	88	43					184
		Total	Exp.	3896	3058	1774	1154	1731	1395	1275	278	14561
			Line	327	2442	2446	2096	324	0	0	0	7635
			Cap.	268	47	526	259	0	0	0	0	1100

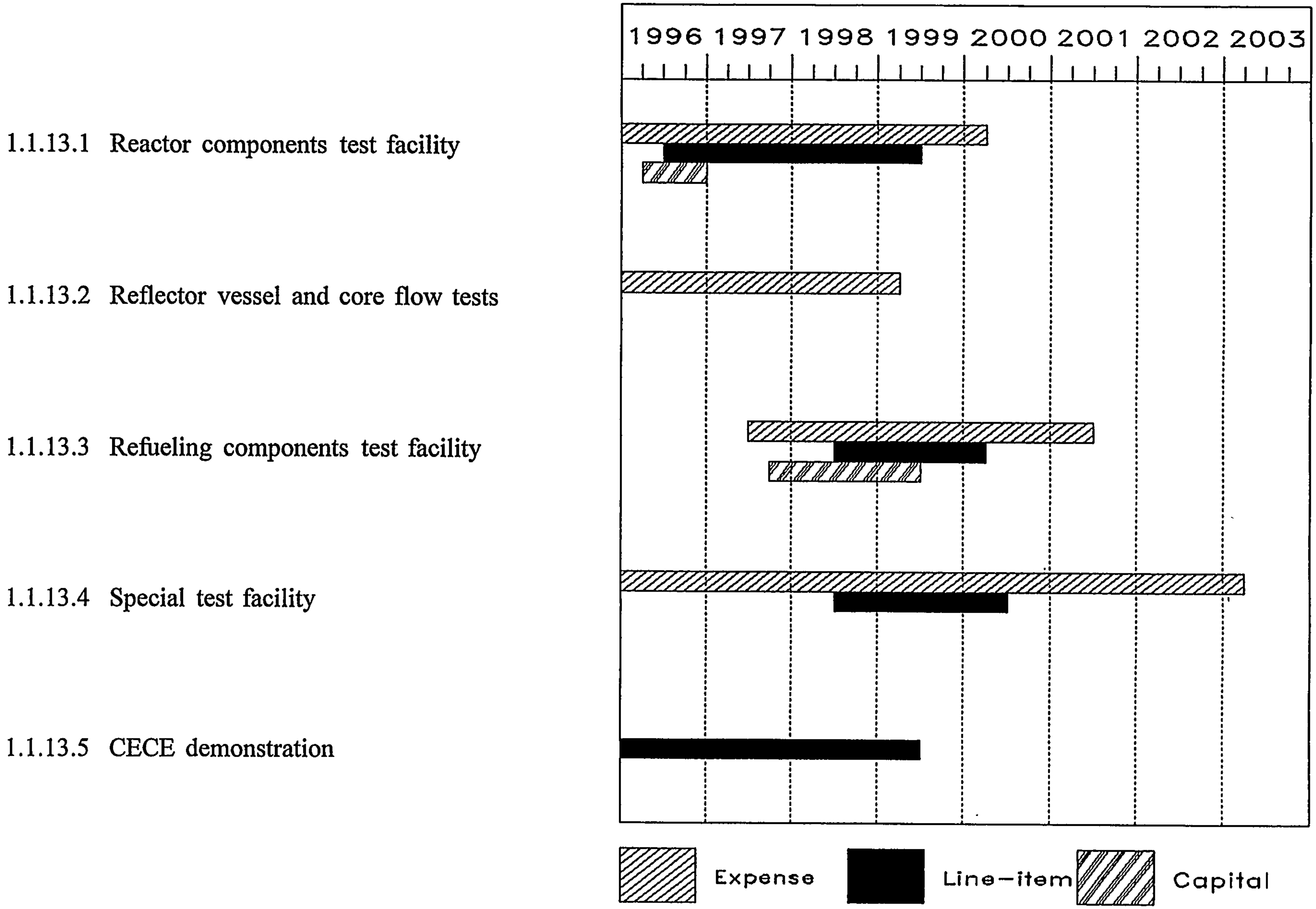


Fig. 3.42. Schedule for WBS 1.1.13 facility concepts.

3.13.1.2.1 Reactor components test facility

The reactor component tests, excluding the inner control rod latch magnet test described below, have similiar test facility requirements:

1. water at pressures of the primary and secondary cooling system;
2. simulated full-scale reflector vessel interfaces, in particular the upper primary coolant outlet flange and thimble flanges; and
3. data acquisition and control system.

To accommodate these requirements, a common-use facility will be used to perform the reactor components tests. This facility will consist of pressurized water supply systems to provide primary and secondary coolant system pressures, a common vessel with provisions for simulating the reflector vessel openings for thimbles, the upper primary coolant outlet, and the lower CPBT seal. Smaller secondary vessels may be necessary for performing high pressure thimble collapse tests. A data acquisition and control system will provide control of the water systems, experiments, and test hardware, and will provide data acquisition for sensors. Seal tests, outer shutdown rod tests, locking ring/bolt torque tests, thimble tests, and possibly some of the inner control rod tests will be performed in this facility.

3.13.1.2.2 Seal tests

The seal tests will be performed in the reactor components test facility and will be used for two different functions. First, they will evaluate different seal configurations and seals from different vendors. Second, they will be used to verify the seal design chosen for the final design. Screening tests will be performed under conditions that simulate the actual operation and maintenance conditions expected in the reactor. The tests 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 tests 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 (i.e., bolted vs 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 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 operational 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 leakage characteristics of new seals, and seals that have been installed more than once, will be developed. Tests will include pressure, temperature, and relative motion cycling, as appropriate.

3.13.1.2.3 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. This test will require a simulation of the top of the reflector vessel and the CPBT. The test will be performed in the reactor components test facility. This same type of closure will be used to secure the closure elbow.

3.13.1.2.4 Outer shutdown rod tests

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

These tests will be performed in the reactor components test facility. Both phases could be performed in the same vessel. However, the vessel that would be used for the eight-rod tests is expected to be used for other reactor component tests; therefore, it may be desirable to use a separate vessel for the single-rod phase to save time.

3.13.1.2.5 Outer shutdown single-rod test

The single-rod test will provide early verification that a hydraulically controlled shutdown rod system is a possible secondary shutdown system alternative. Investigation of reaction times, cooling flow rates, and hydraulic pressures for a single rod will provide required input data for design of the hardware, hydraulic supply, and control system of the full eight-rod design.

This test 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; the necessary structure inside the tank to support the shutdown rod assembly; a water pumping system; minimal valving to operate the rod; and appropriate instrumentation and controls. Investigating temperature effects and water density effects is not planned for the one-rod tests.

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 ΔP;
- the operability of the position indicator system as a function of the flow rate throughout the tubing; and
- the response time of a single rod.

3.13.1.2.6 Full-scale eight-rod test

The full-scale eight-rod test is necessary to verify the hydraulic effects of momentum, flow distribution, and flow instabilities when several rods are supplied from a common manifold. These interactive effects cannot be investigated in a single-rod test. Additionally, leak rates and installation techniques will be investigated relative to the design of the hydraulic remote disconnect.

This test would consist of a tank in the reactor components test facility, 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 with a liquid that would be the same density as the D_2O at the normal operating temperature of the reflector vessel.

As a minimum, the test 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;
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 multiple piston seals supplied by a common manifold;
4. provide a means for determining the leakage at the seals of the manifold plug-in remote disconnect units;
5. provide for determining the cooling water flow rate distribution for the neutron absorbers 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 determining the suitability of the latches that retain the shutdown rod assembly on the support structure;
10. provide for determining the scram impact effects of the rods on the manifold, hydraulic cylinders, and support structures;
11. 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.

3.13.1.2.7 Thimble tests

Tests of the various thimbles and beam tubes will eliminate uncertainties related to the design and operation of these components. Analysis methods will be verified for predicting thimble collapse pressure, cooling, and pressure surge survivability. These tests will be conducted in a vessel that is part of the reactor components test facility. External pressure will be rapidly increased to investigate survivability in pressure surges. Various design configurations will be pressurized to collapse for verification of thimble strength. Strain gages will monitor structural conditions during these pressure excursions. Flow rates through thimble cooling shrouds and methods of installation and sealing to a simulated reflector vessel interface will be investigated and verified.

3.13.1.2.8 Inner control rod component tests

This set of tests is designed to test component parts of the inner control and shutdown rod system. Overall operation and response of the complete full-scale system will be tested in the control element test facility which is part of WBS 1.1.7.

Several component parts of the inner control rod system have design and performance uncertainties that need resolution prior to the design and fabrication of a full system test. Verifying performance of these individual components is paramount to continuing the design of the present baseline control rod concept. These tests can be performed in parallel in small test stands and will provide required data for input into the final design effort. Performing these tests as part of a full-scale system in the control element test facility could result in major design and testing delays. Inner control rod component tests to be performed are: latch release magnet test, latch component wear test, lower high pressure seal test, inner control rod damper test, and control rod scram position indicator test.

3.13.1.2.9 Latch release magnet test

This test was completed in the first quarter of FY 1994. This test characterized performance of the latch release magnet and determined its load carrying capacity and response time. Analytical tools used in the latch magnet design were verified. Due to the small available space for this component in the ANSR, a very compact winding design is necessary. Analysis had shown that the baseline latch release magnet design was marginally able to support the pressure and gravity loads applied to it.

Candidate magnet core, winding, and armature materials were selected and full-scale electromagnets were fabricated. These magnets were mounted in a standard tensile testing machine. Various current levels were applied to the magnet windings to establish current/load relationships. With fixed applied loads, a square wave current was applied to the magnet windings. The current wave period was increased until the magnet released. This established the response time between loss of current and magnet release.

Results of the test are available in report ORNL/ANS/INT-54.[51] Figure 3.43 shows the test configuration. Figures 3.44 and 3.45 are photographs of the test magnet components and the test stand arrangement.

3.13.1.2.10 Latch component wear test

While the ANS scram latch is, in principle, similar to the HFIR scram latch, it uses disks rather than spheres and will be constructed of different materials. This test will develop the wear vs time characteristics of the ANS inner control rod latch mechanical components.

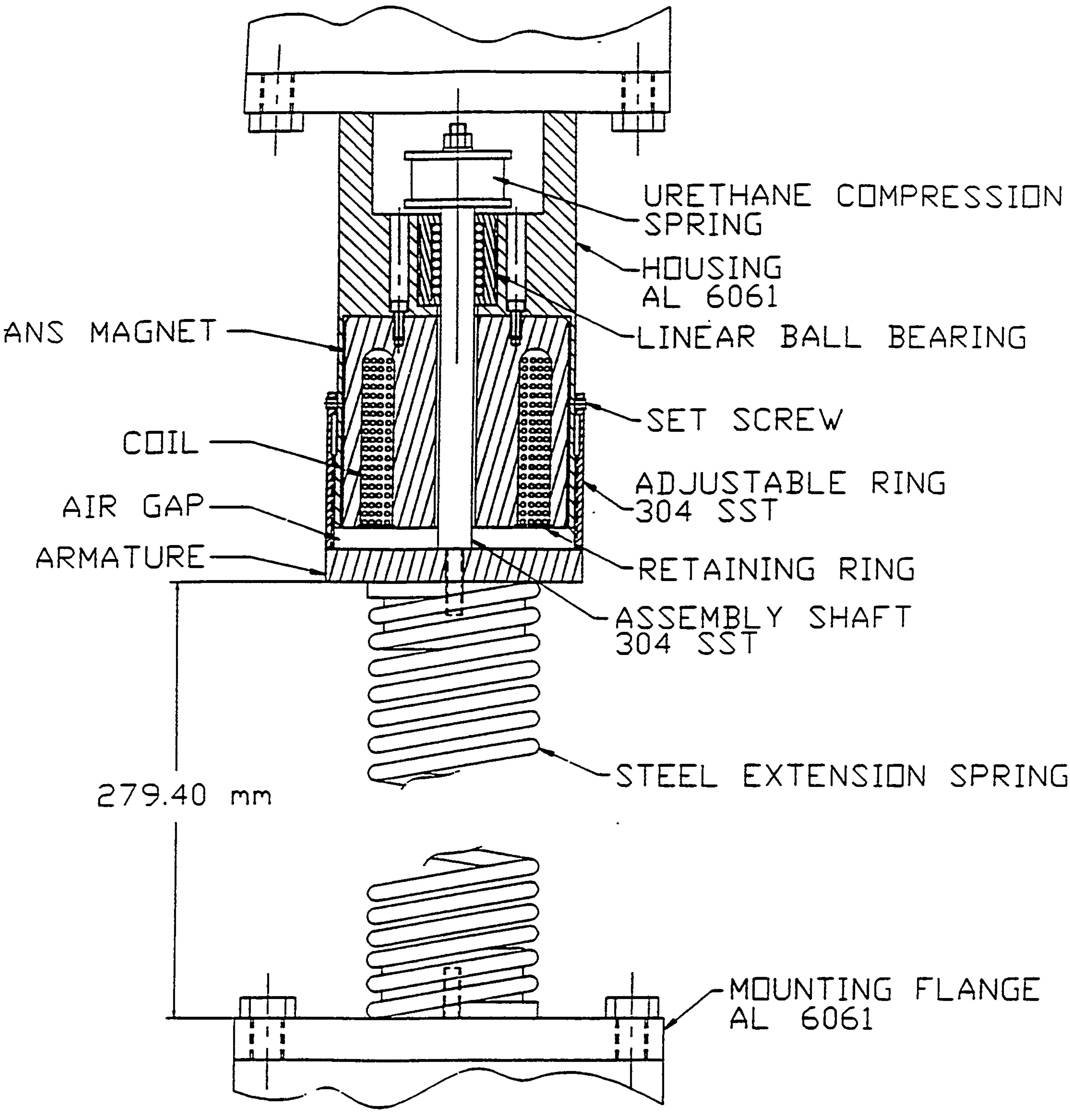


Fig. 3.43. ANS magnet test assembly.

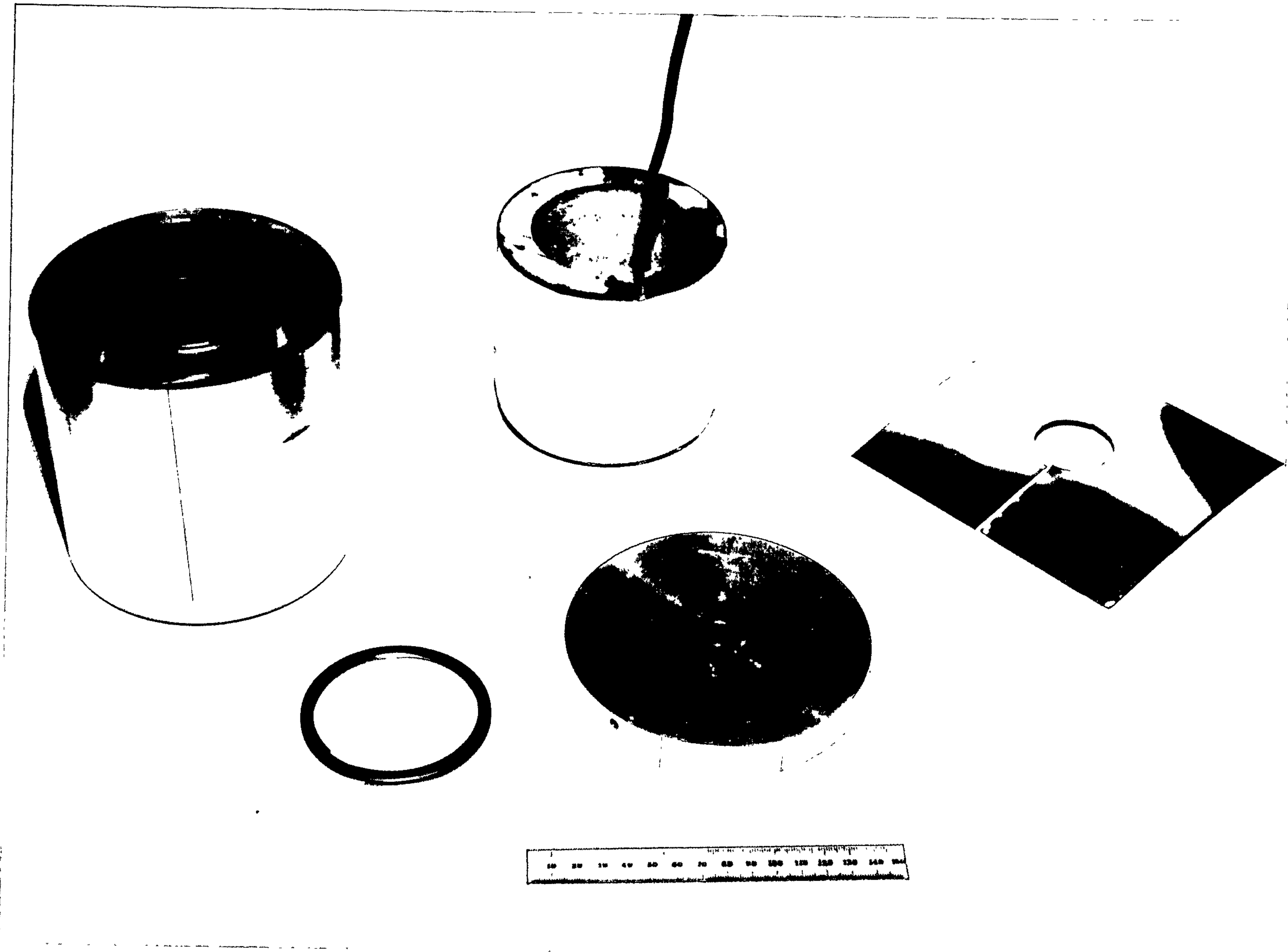


Fig. 3.44. Magnet test components.

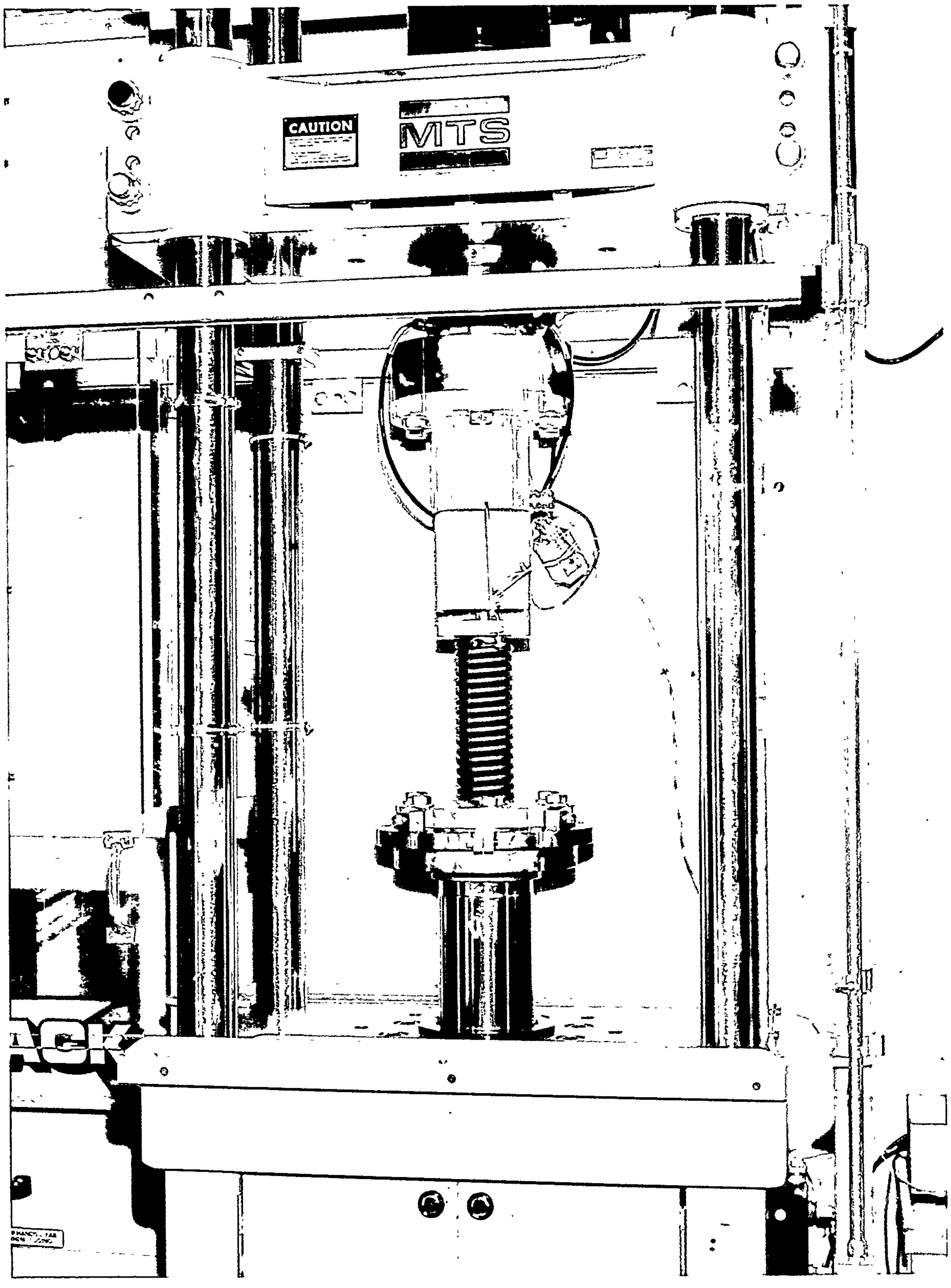


Fig. 3.45. Photograph of the magnet test stand arrangement.

The latch release rod, disks, slotted drive shaft tube, and latch seat are all components that experience high localized stresses and frictional forces during normal operation, scram, and relatching. These components require accurate interface geometry and surface finishes to operate properly. Wear could cause inadvertant unlatching or the inability to relatch after a scram. Additionally, the use of traditional hard surface materials such as stellite, which contains high levels of cobalt, has been restricted due to activation levels.

Latch components will be fabricated from selected candidate materials. A test device will apply loads and motion that simulate the normal latched, scram, and relatching scenarios. This test device will have the capability of cycling the hardware through many operation and scram cycles and performing life cycle tests. The test device will utilize a standard computer controlled tensile machine to apply specified loads and motions onto the latch hardware. The latch hardware will be immersed in water of the appropriate pH to simulate the reactor environment and its effect on friction.

A conceptual design of the latch wear test device has been prepared and is shown in Fig. 3.46.

3.13.1.2.11 Lower high pressure seal test

The inner control rod drive shafts pass through a dynamic seal between the subpile room at atmospheric pressure and the heavy water primary cooling system. Leaks through this seal into the subpile room are unacceptable. This leak would represent a breach of the primary pressure boundary and result in a difficult clean up of heavy water containing tritium. The ANS seal geometry consists of three seals in a tight cluster that share buffer fluids, buffer gasses, and drain tubes. High neutron flux levels at the seal location exclude organic elastomer sealing materials. The tight geometry, nonorganic seals, and zero leak requirements make the ANS seals unique when compared to HFIR and other standard seal designs for this component.

This test will investigate the designability, reliability, wear behavior, and overall effectiveness of the lower high pressure seal. Seal vendors will initially participate in developing a conceptual seal design. One or more candidate designs will be selected for testing. Tests will then be performed that will test the seals ability to effectively seal against leaks while subjected to typical and atypical conditions. This would include dynamic shaft motion through the seal, misalignment, leak detection and monitoring, maintainability, and lifetime. The test will establish the ability of industry to design and build a seal that meets the rigid and unique ANS requirements for this component.

3.13.1.2.12 Inner control rod deceleration damper test

During a scram, the inner control rods are accelerated into the core and stopped at the proper scram location. Decelerating and stopping the rods in a very short time span could possibly result in buckling of the control rod drive tubes. This could impact our ability to shut the reactor down and would affect our ability to restart after a scram. The conceptual deceleration technique is a combination of hydraulic displacement damping and mechanical springs. The hydraulic component consists of displacing D_2O through a pattern of holes in the guide tube as the rods move downward. The hole pattern will be designed to tailor the deceleration curve to avoid forces that would cause buckling of the control rod drive tubes.

This test will characterize the hydraulic damping and be a test bed for developing the required hole pattern. Test data will be used to provide validation of analytical techniques and models. The test geometry will consist of a simulation of the drive tube piston, the guide tube with holes, and springs. Other components of the inner control rod system, which do not contribute to the performance of the damper, will not be modeled in this test. The present test concept uses static water. However, the effects of flowing water over the damper section will be evaluated early in the

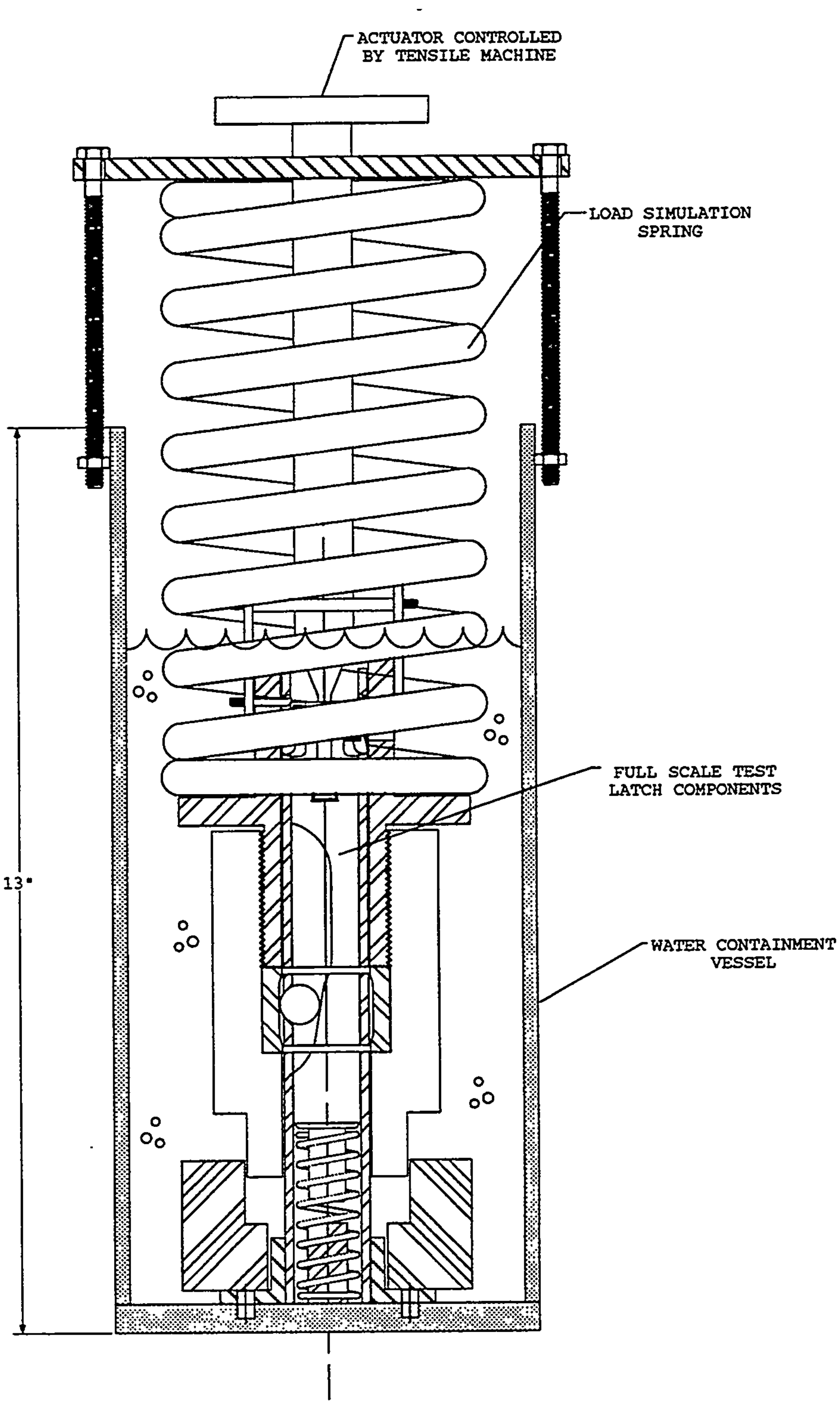


Fig. 3.46. ANS inner control rod latch wear test stand.

test design to insure all relevant factors are provided in the test stand. Springs will initially accelerate the simulated drive tube piston to typical velocities, and then the deceleration profile will be measured.

The finalized hole pattern and full-scale damper scheme will be tested at full, primary system flow in the control element test facility which is part of WBS 1.1.7.

3.13.1.2.13 Inner control rod position indicator test

Reactor control requires the ability to detect the position of the inner control rod in the fully inserted (scram) position. Neutron levels restrict the use of electrical devices such as switches and proximity sensors. The conceptual baseline design for this sensor is a hydraulic system that senses pressure and flow changes when a flow orifice in the deceleration spring housing is either covered or uncovered by the absorber drive tube. The reliability and sensitivity of this hydraulic switch scheme is not well known.

This test is to determine the suitability of the hydraulic position indicator. Various orifice hole cross sections will be tested to determine the best shape for this feature. Multiple hole orifice patterns could be investigated to determine the feasibility of gaining position indication over a range of positions.

Results of this test will also apply to the design of the outer control rods. A similar pressure/flow switch scheme is proposed as the baseline design for position indication for the outer control rod system.

The finalized position indicator system will be tested in full, primary system flow and pressure in the control element test facility, which is part of WBS 1.1.7.

3.13.2 Reflector Vessel and Core Flow Test

3.13.2.1 Justification for the Reflector Vessel and Core Flow Test Task

In the thermal-hydraulic design and safety analyses of the core and reflector components, certain flow patterns and flow splits are assumed based on computational fluid dynamics codes. Flow tests are needed to validate the computational fluid dynamics analyses, and, in turn, to validate key assumptions used in design and safety analyses. The results of these tests could require modification of the designs of components in both the reflector and the fuel element regions; therefore, they should be completed as early in the program as practical.

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.

3.13.2.2.1 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, including 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.

3.13.2.2.2 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 region around the control rods or through the region separating the inner CPBT wall 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. 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

Failures in operation of the refueling components could have safety and potentially large availability impacts; therefore, they must be tested to provide performance confirmation and to develop reliability data. 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 lead to 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 dry and completely accessible. The facility will be representative of actual equipment, but the test components may be fabricated without the extensive QA 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 head test. Separate descriptions of these tests follow.

3.13.3.2.1 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 determine that the fuel element is properly attached to the handling system.

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

3.13.3.2.3 The tool head test subtask

Typical tool heads will be tested in the refueling components test facility. Both simple and 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 Tests

3.13.4.1 Justification for the Special Tests Task

The special tests task will provide feedback to the design team on certain refueling cycle uncertainties 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 Tests 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 will be performed in a new or existing facility or in small special-purpose test fixtures. Four tests have been identified so far and are described in the following paragraphs.

3.13.4.2.1 The irradiation capsule disconnect test subtask

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

3.13.4.2.2 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.

3.13.4.2.3 The flush/purge/lock test subtask

Because of high internal heating loads, some components cannot be allowed to dry in air before being transferred to light water storage. Transferring these highly activated ANS components from D_20 to storage in light water requires (1) flushing the component with heavy water to reduce the levels of tritium entrained in heavy water and (2) flushing heavy water out of the component with light water so the heavy water can be reclaimed. The goal is to minimize the amount of tritium transferred to the light water. The analytical models that predict the amount of mixing will be verified.

The flush/purge test facility will simulate the flushing procedure for several typical components and determine the flushing efficiency of the operation. Dye will be used to simulate the D_20. Dye levels will be measured and monitored to determine flushing efficiencies. The necessity to approximate the D_2O density will be evaluated. Potential components to be tested in this facility are the upper and lower fuel assemblies, transuranium target rods, inner control rod components, and irradiation capsules. Other components may be identified as detailed reactor maintenance and disassembly procedures are developed.

The facility consists of a flushing tank, dye insertion and detection equipment, pneumatic mixing equipment, fill and drain holding tanks, interconnecting piping, valves, circulation pumps, and heat exchangers as required to simulate the design concept. Testing on the upper and lower ANS fuel elements will be simulated with existing dummy HFIR fuel elements.

3.13.4.2.4 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 before it is installed at the actual cell. It will be necessary to program the manipulators on site to optimize the process.

3.13.5 Combined Electrolysis and Catalytic Exchange (CECE) Pilot Plant Test Facility

3.13.5.1 Justification for the CECE Pilot Plant Test Facility

Failure to identify potential weaknesses in the CECE system could cause serious schedule delays during the heavy water processing period before initial reactor fill. If this testing were not done and problems with the CECE system did occur, then portions of this work would have to be completed using the ANS heavy water upgrade and detritiation facility (HWUDF) plant equipment either late in the commissioning phase of the project or during the heavy water processing period.

This would be technically more difficult to do, would have to be done with less experienced personnel, and could delay startup of the ANS facility.

Other facilities of this type do exist. These include the Mound and the Chalk River facilities. The Mound facility is dedicated to another mission, and the Chalk River facility would require a costly reconfiguring from the liquid phase catalytic exchange mode of operation into the CECE mode. Portions of the demonstration program could also be done at Electrolyser Corporation (testing of the electrolysis cell) or the entire program, as now defined, could be done at the Chalk River Laboratories. However, a portion of the demonstration objectives (testing the catalyst with ORNL water and developing an ORNL experience base) could be accomplished only at ORNL. An evaluation of alternative sites will be performed early in the test design phase.

3.13.5.2 Description of the CECE Pilot Plant Test Facility and Tests

The purpose of the CECE pilot plant test facility is to test certain key components of the ANS HWUDF at as close to actual operating conditions as possible. These components are the electrolysis cell, the catalyst, the trickle bed recombiners, and the CECE column control scheme. In doing these tests, an experience base in the operation of the equipment will be started with ORNL personnel.

The flowsheet for the test system is shown in Fig. 3.47. The suggestion to test the electrolytic cell originally came from Electrolyser Corporation and was expanded to include the operation of a catalyst column. The only cell built to date using this design was built as a 7.5-kA prototype to be scaled up later. The test electrolytic cell is a full-size HWUDF cell that operates at a current level of 25,000 A at 2.5 V. It has an electrolyte volume of about 3 L per 1000 A, for a total of about 75 L. The cell uses a potassium hydroxide electrolyte. Deuterium from the cells is passed through the CECE columns and exchanged with protium, and the protium is vented. The electrolysis cell will produce sufficient pressure that a compressor is not needed. The catalyst columns are equivalent in length to one HWUDF column and scaled in diameter to duplicate the fluxes in those columns. Demineralized process water is fed through the top of the column. To simulate product withdrawal to the cryogenic distillation system from the electrolytic cells, a stream of gaseous deuterium will be removed from the stream flowing to the CECE column. Removal of the gaseous stream in this manner is required to test the control system, as well as to simulate a feed stream to the column and test the trickle bed recombiner technology. As it is used in the HWUDF product recombination system, this recombiner will be a 10-in.-diam trickle bed unit, which is appropriately sized for the stream flowrate.

Heavy water is available at the Oak Ridge Y-12 Plant in the form of "half heavy water" (50% D_2O–50% H_2O), which costs about 1/3 as much as the equivalent volume of reactor-grade heavy water. For this test program, since the column is not long enough to maintain a complete concentration gradient, the use of this material is preferred. The tritium content of this water is about 500 μCi/L. Tritium spiking may also be required in some tests, but the quantities will be in the millicurie range, which should pose minimal safety concerns. The deuterium concentration of the hydrogen vented from the column top would be natural background.

A ventilation system will be required to ventilate the equipment where it is installed, along with a room hydrogen monitor. In addition, utilities such as cooling water, electrical power, and ventilation will be required, along with the necessary permits, safety reviews, and environmental documentation.

The facility would be maintained in a standby mode at the conclusion of testing and would be available to simulate problems that might occur in the full-scale HWUDF after start-up. The facility could also be used to provide very limited backup upgrading capacity in the event of an unplanned shutdown of the HWUDF. At the conclusion of the program, the cell would be available to the

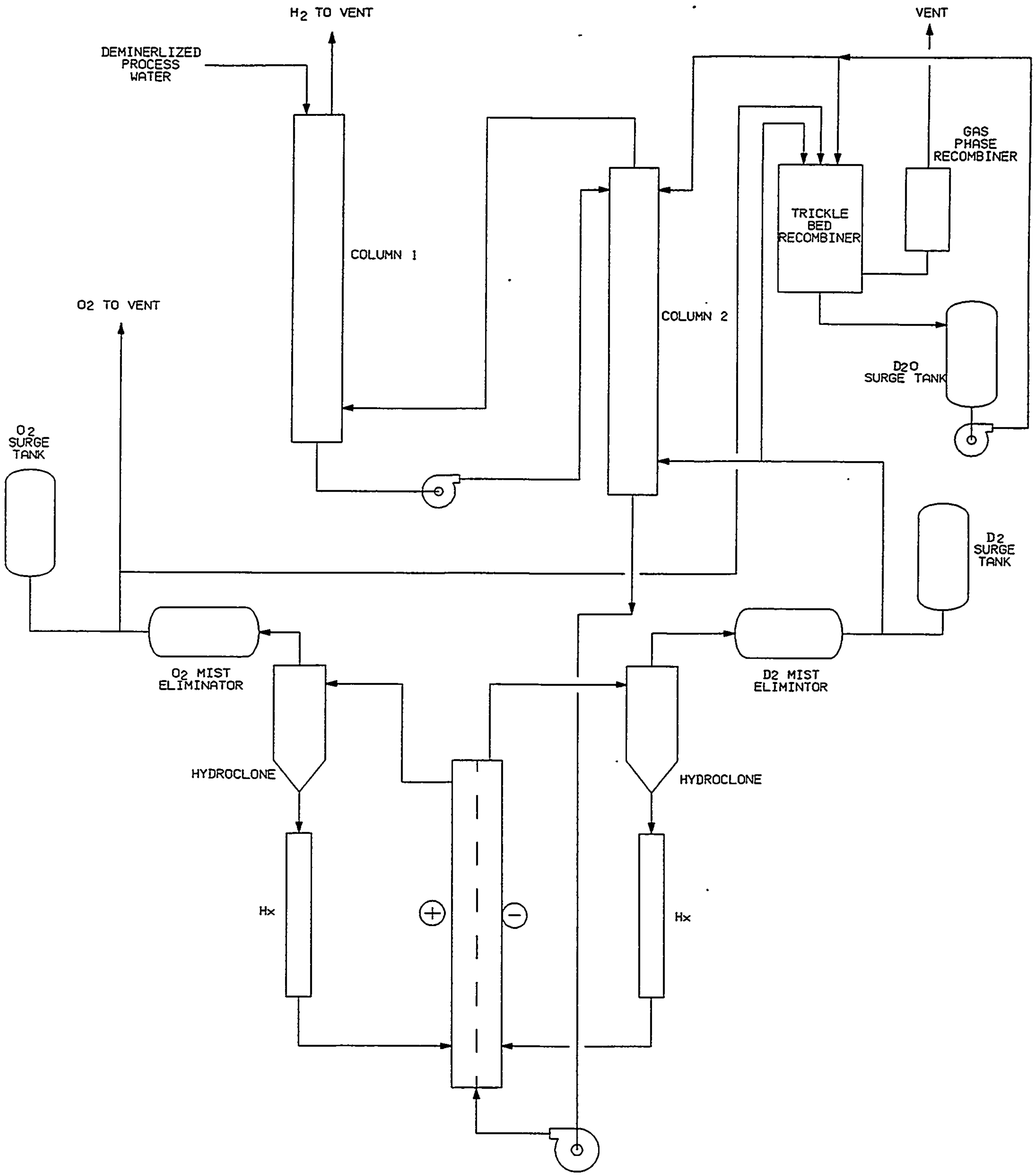


Fig. 3.47. Flowsheet for the CECE test system.

HWUDF as a spare. The catalyst would be removed from the column and retained as spare material, and the rest of the equipment would be removed for disposal. The heavy water would be returned to Y-12 inventories.

3.13.5.2.1 Test descriptions and objectives

The electrolysis cell will be operated over a period of time to demonstrate the design. Data on power consumption, throughput, efficiency, leakage occurrences, reliability, and ease of maintenance will be generated and used to refine the design of the actual cell for the facility.

The CECE column computer model predicts that precise feed and withdrawal rates are critical to maintain low column effluent discharges of deuterium and tritium, and to maintain isotopic purity of the heavy water product. The control system will be demonstrated by controlling the gaseous deuterium removed from the electrolytic cells. This will allow determination of the sensitivity of the system to load and/or to set point changes and will be used to verify the computer model calculations.

The trickle bed recombiner will be tested to demonstrate the design, efficiency, and reliability, as well as to identify any problem areas.

The catalyst will be tested over the life of the CECE pilot facility to determine if some unknown component is present in ORNL water that could impair catalyst function. Although extensive catalyst testing has been done, the operation of the catalyst is critical to the success of the plant, and the identification of any special water treatment requirements is warranted.

Perhaps the most important goal is to develop an experience base at ORNL in the operation of this type of equipment. Very few personnel at ORNL have had experience in handling heavy water or tritium, and there is no experience base at all for CECE technology. Therefore, what works very well at Chalk River Laboratories might not work as well here because of the lack of this technical expertise.

The facility will be operated over a period of 1 year in a batch mode. Both to facilitate operations and avoid undue resource requirements, the operation will be conducted using 15 shift/week runs (5-day continuous) with operation during off-shifts conducted by automatic control.

The following test objectives are to be accomplished with this facility.

1. Test Electrolyser Corporation's low-inventory electrolysis cell at HWUDF operating conditions. This product has no experience base in the size required. The program will test a full-size cell. The full-scale HWUDF will have 22 cells of the same size to produce the flows required.
2. Test the CECE column control strategy. The CECE column computer model predicts that maintaining precise gas and liquid feed and withdrawal rates is critical to maintain low column effluent discharges of deuterium and tritium, and to maintain isotopic purity of the heavy water product. Therefore, the control of the system is potentially a critical parameter that has not been demonstrated.
3. Test the CECE catalyst with the water available at ORNL. The catalyst will be tested over the life of the CECE pilot facility to determine if some unknown component is present that could impair catalyst function. Although extensive catalyst testing has been done, the operation of the catalyst is critical to the success of the plant, and the identification of any special water treatment requirements is warranted.
4. Test the trickle bed reactor. To test the control system, a product stream of gas from the electrolysis cells is required. This stream will be rich in deuterium and should not be discarded but returned to the system, requiring a recombiner. This recombiner can be a small trickle bed unit. This situation also allows the opportunity to simulate a feed stream to the CECE column.

5. Develop an experience base with ORNL personnel in the operation of this type and size of equipment.

Evaluation of the proposed facility suggests that it will perform all of the required functions satisfactorily. The facility would lend itself to modification if other testing is desired at a later date. The facility would be maintained in a standby mode at the conclusion of testing and would be available to simulate problems that might occur in the full-scale HWUDF after start-up. The facility could also be used for training the HWUDF operators prior to their certification and could provide very limited backup upgrading capacity in the event of an unplanned shutdown of the HWUDF.

4. REFERENCES

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